

REQUEST FOR CONFIDENTIALITY

OF MATERIAL SUBMITTED IN RELATION TO CMD # 24-H3.1

IMPORTANT NOTE:

The purpose of the confidentiality request process is to seek a decision from the Commission as to whether specific information being presented to a Commission proceeding can be protected. Generally, material received as part of a matter before the Commission is made available to the public by default. The rule of confidentiality (i.e., Section 12 of the [CNSC Rules of Procedure](#)) is applied only if the Commission decides in favour of a request for confidentiality.

Restricted access to proceedings and related material is exceptional, proportional, and minimal, and is not imposed lightly. Therefore, and to minimize the possibility of a challenge to a confidentiality ruling, the Commission weighs any request for confidentiality against the criteria set out in Section 12 to confirm that:

- the importance of protecting the information outweighs the public interest in public hearings and disclosure of evidence; and
- the confidentiality measures would affect the public nature of the proceeding only to the extent necessary to adequately protect the given information.

In the interest of enabling a timely decision, any request for confidentiality must be accompanied by redacted versions of all documents named in the request, and/or adequately informative summaries that can be made available to participants and the public. **Please provide the appropriate versions, as applicable.**

It is the responsibility of the person making the request to provide an adequately detailed explanation as to how and why subrule 12(1) applies.

In the matter of:

OPG Application for a Licence to Construct a Single BWRX300 Reactor at the DNNP Site

With regard to CMD: **CMD 24-H3.1 “OPG Written Submission in support of the Hearing on the Application for the Darlington New Nuclear Project Power Reactor Construction License”**

This request has been prepared in Canada, in the province of Ontario in the matter of **OPG Application for a Licence to Construct a Single BWRX300 Reactor at the DNNP Site**, scheduled for consideration in a Public Hearing scheduled for October 2024 and January 2025.

I, Mark Knutson, of 889 Brock Rd, Pickering, Ontario L1W 3J2, am an authorized representative of Ontario Power Generation. I understand that:

- documents and information (“the material”) provided to the Canadian Nuclear Safety Commission (“the Commission”) as part of a public proceeding may be made publicly available;
- the material is considered confidential only if it is prescribed information under the [Nuclear Safety and Control Act](#) (NSCA), as defined in section 21 of the [General Nuclear Safety and Control Regulations](#), or if the Commission takes measures to protect the information; and
- regardless of any request for confidentiality or approval of same, the material may be disclosed if the Commission is required by law to disclose it (for example, after a request under [Access to Information Act](#)).

I hereby request that the Commission take measures to protect the following information, pursuant to rule 12 of the [Canadian Nuclear Safety Commission Rules of Procedure](#):

Note: Where the request for confidentiality applies only to part of the submission, the portions to be deemed confidential must be clearly identified to distinguish them from any content that is non-sensitive.

TABLE 1: MATERIAL TO BE DEEMED CONFIDENTIAL

	Item Name	Portion(s) to be Deemed Confidential	Reason for Request (details to be provided below)
1.	NK054-CORR-00531-10737, OPG Confidential – DNNP – Submission Package #2(b) Design and Safety Analysis Deliverables in Support of the Licence to Construct Application for the CNSC Review	<input type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown **The letter is not required to be treated as confidential and can be made available, however the attachments to this document are confidential as identified in rows 2-19**	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential.
2.	NK054-REP-01210-00140, BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Design Description Qualification and BWR Fuel Licensing	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Section 4.2, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
3.	NK054-REP-01210-00147, BWRX-300 Darlington New Nuclear Project (DNNP) Preliminary Fire Safe Shutdown Requirement and Analysis Document	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input checked="" type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify),

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	Item Name	Portion(s) to be Deemed Confidential	Reason for Request (details to be provided below)
			and is consistently treated as confidential. <u>** OPG's PSAR, Section 9A.6.3, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
4.	NK054-REP-01210-00167, BWRX-300 Darlington New Nuclear Project (DNNP) Preliminary Fire Hazards Assessment Report	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input checked="" type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Section 9A.6, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
5.	NK054-REP-01210-00168, BWRX-300 DNNP Fire Protection System Preliminary Code Compliance Review Report	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input checked="" type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Section 9A.6, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
6.	NK054-REP-01210-00169, BWRX-300 Darlington New Nuclear Project (DNNP) Independent Third-Party Review Report of Preliminary Fire Protection Design	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input checked="" type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial,

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	Item Name	Portion(s) to be Deemed Confidential	Reason for Request (details to be provided below)
			<input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Section 9A.6, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
7.	NK054-DP-01210-00002, BWRX-300 Darlington New Nuclear Project (DNNP) Human Factor Engineering Program Plan	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Chapter 18, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
8.	NK054-REP-01210-00163, BWRX-300 Darlington New Nuclear Project (DNNP) Probabilistic Safety Assessment Summary	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input checked="" type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Chapter 15 is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
9.	NK054-REP-01210-00158, BWRX-300 Darlington New	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input checked="" type="checkbox"/> The information is a matter of national or nuclear security

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	Item Name	Portion(s) to be Deemed Confidential	Reason for Request (details to be provided below)
	Nuclear Project (DNNP) Hazard Analysis Results		<input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR Chapter 2 is proposed as a sufficiently descriptive publicly-accessible summary**</u>
10.	NK054-REC-08130-1049591, DNNP SMR1 Level 0 Schedule	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's LTC Application Figure 1.1-2 is proposed as a sufficiently descriptive publicly-accessible summary**</u>
11.	NK054-REP-01210-00145, BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Bundle Information Report for Equilibrium 12-Month Cycle	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or

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	Item Name	Portion(s) to be Deemed Confidential	Reason for Request (details to be provided below)
			<input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Chapter 4, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
12.	NK054-REP-01210-00146, BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Equilibrium 12-Month Cycle Nuclear Design Report	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Chapter 4, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
13.	NK054-REP-01210-00159, BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Chapter 4, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
14.	NK054-REP-01210-00160, BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Assembly Mechanical Design Report	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial,

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			<input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Chapter 4, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
15.	NK054-REP-01210-00161, BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Assembly Thermal-Mechanical Design Report	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Chapter 4, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
16.	NK054-REP-01210-00162, BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Assembly Pressure Drop Characteristics	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Chapter 4, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
17.	NK054-REP-01210-	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security

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	Item Name	Portion(s) to be Deemed Confidential	Reason for Request (details to be provided below)
	00164, BWRX-300 Darlington New Nuclear Project (DNNP) TRACG Application		<input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>** OPG's PSAR, Chapter 4, is proposed as a sufficiently descriptive publicly-accessible summary.**</u>
18.	NK054-REP-01210-00172, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Compliance Matrix Report	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>**Redacted version enclosed</u>
19.	NK054-REP-01210-00170, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential.

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			<u>**Redacted version enclosed</u>
20.	NK054-CORR-00531-10766, DNNP – Submission of Package #5(b) Core Control Processes and Operations Aspects Confidential Deliverables in Support of the Licence to Construct Application for the CNSC Review	<input type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown <u>**The letter is not required to be treated as confidential and can be made available, however the attachments to this document are confidential as identified in rows 21/22**</u>	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential.
21.	NK054-REP-03420-00001, BWRX-300 Occupational Dose Assessment Report	<input checked="" type="checkbox"/> Entire content <input type="checkbox"/> Redacted content as shown	<input checked="" type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input checked="" type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or <input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>**OPG's written CMD, Section 4.7.4, is considered to provide a sufficient publicly-accessible summary of the material presented in this document.**</u>
22.	NK054-REP-03500-00001, Independent Peer Review of the Preliminary Safety Analysis Report (PSAR) for the Darlington New Nuclear Project	<input type="checkbox"/> Entire content <input checked="" type="checkbox"/> Redacted content as shown	<input type="checkbox"/> The information is a matter of national or nuclear security <input type="checkbox"/> Disclosure of the information would likely endanger the life, liberty, or security of a person or person(s) The information is of a: <input type="checkbox"/> financial, <input type="checkbox"/> commercial, <input type="checkbox"/> scientific, <input checked="" type="checkbox"/> technical, <input type="checkbox"/> personal (and the person has not consented to disclosure), or

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			<input type="checkbox"/> other nature (specify), and is consistently treated as confidential. <u>**Redacted version enclosed</u>

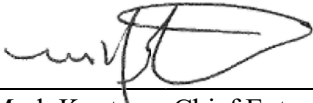
Detailed reason(s) for request:

- The above-noted material should be protected for the following reasons:
The documents identified contain information that is a matter of national or nuclear security and are of commercial, technical or scientific value due to the novel nature of both the design and the project management structure.
- I attest that the above-noted material is not available through any public sources.
- MANDATORY:** I have included a **summary** or **redacted** version of the material that provides adequate detail to satisfy the public interest in public hearings and disclosure of evidence.
- I understand that if this request is not approved by the Commission, I may withdraw the associated material within five business days of receiving written notice of the Commission's decision from the Commission Registrar (except as noted in items 5 and 6, below).
- Notwithstanding item 4, above, I understand that if submission of the material is required pursuant to reporting requirements under the [NSCA](#) or the regulations under the NSCA, or pursuant to a licence issued under the NSCA, or if the material is specifically requested by the Commission, it may **not** be withdrawn.
- I understand that upon receipt of this request, the Commission Registrar will treat the material that is subject to this request as confidential unless and until the Commission makes a ruling to deny this request.

Attachments:

- CMD 24-H3.1 (Submission from OPG) – Application for a Licence to Construct one BWRX-300 Reactor at the Darlington New Nuclear Project Site (DNNP)
- OPG Report, "Ontario Power Generation Inc. Darlington New Nuclear Project: BWRX-300 Preliminary Safety Analysis Report" NK054-SR-01210-00001 R01, Chapters 2, 3, 4, 9A, 12, 13, 15.
- OPG Document, Darlington New Nuclear Project – Application for a Licence to Construct a Reactor Facility
- NK054-REP-01210-00172, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Compliance Matrix Report
- NK054-REP-01210-00170, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report
- NK054-REP-03500-00001, Independent Peer Review of the Preliminary Safety Analysis Report (PSAR) for the Darlington New Nuclear Project

Authorized signature:



Mark Knutson, Chief Enterprise Engineer and

2024/07/26

Chief Nuclear Engineer

Date



HITACHI

GE Hitachi Nuclear Energy

Ontario Power Generation Inc. Darlington New Nuclear Project:

BWRX-300 Preliminary Safety Analysis Report



ONTARIOPOWER GENERATION	
ACCEPTED	✓
ACCEPTED AS NOTED	
REVISE AND RESUBMIT	
	15MAR2023
Signature	Date
Name: Karim Osman	
Dept: DNNP Design Engineering	
This acceptance does not relieve the contractor from responsibility for errors or omissions or from any obligations or liability under this contract.	
Notes: OPG Document Number: NK054-SR-01210-00001 R001 OPG Security Classification: OPG Proprietary	

Revision 1

March 7, 2023

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HITACHI

GE Hitachi Nuclear Energy

NEDO-33951

Revision 2

March 7, 2023

Non-Proprietary Information

**Ontario Power Generation Inc.
Darlington New Nuclear Project
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 2
Site Characteristics**

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

NEDO-33951 REVISION 2
NON-PROPRIETARY INFORMATION

REVISION SUMMARY

Revision #	Section Modified	Revision Summary
0	All	Initial Release
1	Section 2.1.2 Section 2.2.3 Section 2.5.1 Section 2.6.4 Section 2.6.5 Section 2.6.8 Section 2.6.9 Section 2.7.1 Section 2.7.2 Section 2.7.3 Section 2.8.2 Section 2.11.4 Section 2.12.1 Section 2.12.5 Section 2.12.8	Incorporated corrections per customer acceptance review
2	All	Edited to improve readability, streamline the text, and ensure consistency across all sections of Chapter 2
	All	Several paragraphs are deleted for they became irrelevant, outdated, or obsolete due to the incorporation of recent (2022 and 2023) information generated in works involving DNNP site-specific investigations, analyses, and assessments.
	All Summary Tables	The tables at the beginning of each section are updated to reflect the edited and added contents of corresponding texts in that section.
	All Other Tables	Other tables are updated or replaced with inputs from new characteristics and parameters generated in the site-specific studies completed in 2022 and 2023.
	All Figures	Updated or replaced to reflect the new information resulted from several 2022 and 2023 assessments, investigations, and analyses
	Acronym List	Updated to include added acronyms

NEDO-33951 REVISION 2
NON-PROPRIETARY INFORMATION

Revision #	Section Modified	Revision Summary
	Section 2.0	Chapters 7, 19 and 20 added to the list of key chapters, and edits involving details are made to previously listed chapters
	Section 2.1.1	Paragraphs added on Site Topography regarding the different grade elevations at and around the Darlington Nuclear site
	Section 2.1.1	Edited to incorporate information in Reference 2.1-7
	Section 2.1.2	Edited to reflect current contents of Chapter 9B, and to incorporate the information in the 2022 Environmental Impact Assessment in Reference 2.1-4
	Section 2.1.2.1	Edited to incorporate information in the 2022 References 2.1-4, 2.1-5, and 2.1-6
	Section 2.1.2.3	A new bullet added to reflect information on the heavy haul routes described in Reference 2.1-4
	Section 2.1.2.4	Added bullets number 6 And 7 regarding not using the cooling towers and combing the primary and secondary heat transport systems
	Section 2.1.10	Added seven new References 2.1-4 to 2.1-9
	Section 2.2.2	Added a paragraph on the 2022 DNNP Hazard Analysis Methodology (Reference 2.2-10)
	Section 2.2.3.2	Edited to incorporate information in the 2022 assessments reported in Reference 2.2-11 and Reference 2.2-12
	Section 2.2.5.2	Edited to incorporate information in the 2022 PNGS re-assessment documented in Reference 2.2-13
	Section 2.2.11	Three references added: 2.2-13, 2.2-14 and 2.2-15
	Section 2.4.2	Paragraphs added to reflect information in the 2022 EIS in Reference 2.4-2
	Section 2.4.3	Two references added: the 2022 Reference 2.4-2 and the 2009 Reference 2.4-3
	Section 2.5.2.1	Edited and updated to incorporate information in the 2022 Flood Hazard Assessment documented in Reference 2.5-18
	Section 2.5.3 and associated subsections	Edited and updated to incorporate information in the 2022 Reference 2.5-18, Reference 2.5-19
	Section 2.5.3.1	Deleted DNGS information that became irrelevant

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Revision #	Section Modified	Revision Summary
	Section 2.5.3.3	A new Table 2.5-2 is added
	Section 2.5.3.4	Deleted DNGS information that became irrelevant and edited to incorporate information in the 2022 Reference 2.5-18
	Section 2.5.4	Edited and updated to incorporate information in the 2022 Reference 2.5-18 and the 2023 Climate Change Impact Strategy documented in Reference 2.5-20
	Section 2.5.5 and associated subsections	Edited and updated to incorporate information in the 2022 References 2.5-18 and 2.5-21
	Section 2.5.6 and associated subsections	Edited and updated to incorporate information in the 2022 Reference 2.5-18
	Section 2.5.7	Added four new References: 2.5-18 and 2.5-21
	Section 2.6.2	Edited and updated to incorporate information in the 2022 Flood Hazard Assessment in Reference 2.6-17
	Section 2.6-4	Edited and updated to incorporate information in the 2022 Reference 2.6-17
	Section 2.6-5	Edited and updated to incorporate information in the 2022 Wind Gust Analysis in Reference 2.6-14
	Section 2.6-5	Added new Table 2.6-3 and Table 2.6-4
	Section 2.6.9	Edited and updated to incorporate information in the 2022 Winter PMP Validation in (Reference 2.6-15)
	Section 2.6.9	Added new Table 2.6-7
	Section 2.6.12	Edited to incorporate information in the 2023 Climate Change Impact Strategy in Reference 2.6-19
	Section 2.6.13	Added six new References: 2.6-14, 2.6-15, 2.6-16 and 2.5-18
	Section 2.7	<p>The entire Section 2.7 is re-configured to incorporate new information documented in:</p> <ol style="list-style-type: none"> 1. The 2023 DNNP Foundation Interface Analysis (FIA) Report (Reference 2.7-38). 2. The 2022 DNNP geotechnical investigations and test results, Phase-1 Power Block (Reference 2.7-39) 3. The 2023 offshore geotechnical investigations (Reference 2.7-40)

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		<p>4. The 2022 DNNP-specific Probabilistic Seismic Hazard Assessment (PSHA) (Reference 2.7-41)</p> <p>5. The 2022 DNNP seismically-induced soil liquefaction assessment (Reference 2.7-42)</p> <p>Added referencing to the 2022 and 2023 completed DNNP/BWRX-300 investigations, analyses, and assessments.</p>
	Section 2.7.1	Deleted irrelevant DNGS information and outdated information
	Section 2.7.2.4	Added information based on the 2023 offshore investigations (Reference 2.7-40), and deleted outdated information
	Section 2.7.3.1	Updated relevant figures, and added information based on the 2022 BWRX-300 Power Block geotechnical investigations (Reference 2.7-39)
	Section 2.7.3.2	Edited and added new information, including Table 2.7-1, Table 2.7-2, and Table 2.7-3, documented in the results and figures from the 2022 Power Block geotechnical investigations (Reference 2.7-39)
	Section 2.7.3.3	Deleted outdated information, made edits, and added new information, per in the results and figures in the 2023 FIA Report (Reference 2.7-38), and the 2022 Power Block geotechnical investigations (Reference 2.7-39)
	Section 2.7.4.1	Introduced the 2022 DNNP PSHA (Reference 2.7-41)
	Section 2.7.4.3	Added information and updated relevant figures based on information in the 2022 PSHA (Reference 2.7-41)
	Section 2.7.4.4	Added information and updated relevant figures based on the 2022 PSHA (Reference 2.7-41)
	Section 2.7.4.6	<ul style="list-style-type: none"> This subsection is currently dedicated to present the results of the work performed in the 2022 PSHA report (Reference 2.7-41) <p>Outdated information deleted</p>
	Section 2.7.4.7	<ul style="list-style-type: none"> Added information under “Surface Faulting” based on findings reported in the 2022 geotechnical investigations (Reference 2.7-39) Added information on potential liquefaction based on the results reported in the 2022 Soil Liquefaction Assessment report (Reference

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		2.7-42) and the 2022 PSHA (Reference 2.7-41) Added new figures
	Section 2.7.4.8	<ul style="list-style-type: none"> This subsection is discontinued <p>Previous information in Subsection 4.7.4.8 pf Revision 1 was merged into other Subsections of Section 2.7</p>
	Section 2.7.5	<ul style="list-style-type: none"> Added referencing to the 2022 NK054-REP-01210-00175 Phase I Geotechnical Investigations (Reference 2.7-39) and the 2023 DNNP FIA report (Reference 2.7-38) Focus is on providing DNNP and BWRX-300 characteristics and parameters <p>Information on “Bounding Design” is deleted as such information is detailed in Chapter 3, Section 3.3.1.1</p>
	Section 2.7.5.1	<p>New information added and updates made based on the 2023 FIA (Reference 2.7-38) and the 2022 DNNP Power Block geotechnical investigations (Reference 2.7-39);); including:</p> <ul style="list-style-type: none"> 2.7.5.1.2 Bearing Capacity Evaluation for Proposed Foundations 2.7.5.1.3 Earth Pressure <p>2.7.5.1.4 Time-Dependent Deformation for Proposed Foundations</p>
	Section 2.7.5.2 and associated Subsections	<p>Outdated information deleted, new information added, and updates made based on the 2023 FIA (Reference 2.7-38) and the 2022 DNNP Power Block geotechnical investigations (Reference 2.7-39); including:</p> <ul style="list-style-type: none"> 2.7.5.2.1 Subgrade Profiles Stratigraphy 2.7.5.2.2 Equivalent Linearized Static Properties of Soil and Engineered Fill Materials 2.7.5.2.3 Equivalent Linearized Static Properties of Rock 2.7.5.2.4 Dynamic Subgrade Properties 2.7.5.2.5 Seismic Design Parameters <p>2.7.5.2.6 Groundwater Level</p>
	Section 2.7.5.3	<p>New information added and updates made based on the 2022 DNNP Power Block geotechnical investigations (Reference 2.7-39)</p>

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	Previous Section 2.7.5.4	Information in Revision 1, Subsection 2.7.5.4 titled "Site Response Analysis" is deleted, since it is covered in Chapter 3, Subsection 3.3.1.1.2
	Previous Section 2.7.5.5	Information in Revision 1, Subsection 2.7.5.5 titled "Design Response Spectra for BWRX-300 at DNNP Site" is deleted and replaced with new information in Subsection 2.7.5.2.5.1 on Ground Motion Spectra
	Previous Section 2.7.5.6	Information in Revision 1, Subsection 2.7.5.6 on "Strain-Compatible Subgrade Profiles for BWXR-300 at DNNP Site" is deleted and replaced with new information in Subsection 2.7.5.2.5.2 on Strain-Compatible Soil properties
	Section 2.7.6	<ul style="list-style-type: none"> Due to the reconfiguration of Section 2.7, several references in Revision 1 are deleted since they are not referenced anymore in Revision 2. The previous identifying numbers of such Revision 1 references were 2.7-20, -22, -23, -27, -28, -29, -33, -38, -39, -40, -42, -43, -44, -45, -46, -47 <p>New references added, from the current Reference 2.7-31 to Reference 2.7-43, inclusive</p>
	Section 2.8	Added Bullet number 6 for and edited the text based on the information in the 2022 DNNP EIS (Reference 2.8-10)
	Section 2.8.7	Added (Reference 2.8-10) regarding the 2022 DNNP EIS
	Section 2.9	Added a new bullet for and edited the text based on the information in the 2022 DNNP EIS (Reference 2.9-16)
	Section 2.9.3	Added (Reference 2.9-16) regarding the 2022 DNNP EIS
	Section 2.10	Introduced and added Table 2.10-1 titled Summary of DNNP Site Relevant Characteristics and Parameters
	Section 2.11.3	Introduced the work completed on FIA (Reference 2.11-19) and the Geotechnical investigations in (Reference 2.11-20)

ACRONYM LIST

Acronym	Explanation
3D	Three-Dimensional
AOO	Anticipated Operational Occurrence
BDBA	Beyond Design Basis Accident
BDBE	Beyond Design Basis Earthquake
BL-AOO	Baseline Abnormal Operational Occurrence
BWR	Boiling Water Reactor
BWRX-300	Boiling Water Reactor, 10 th Design – 300 MWe
CANDU	CANada Deuterium Uranium
CAV	Cumulative Absolute Velocity
CB	Control Building
CEUS	Central Eastern United States
CGD	Canadian Geodetic Datum
CNEP	Consolidated Nuclear Response Plan
CNSC	Canadian Nuclear Safety Commission
CWS	Circulating Water System
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DEC	Design Extension Condition
D-in-D	Defence-in-Depth
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
DRL	Derived Release Limit
DSA	Deterministic Safety Analysis
DWMF	Darlington Waste Management Facility
EA	Environmental Assessment
EIS	Environmental Impact Statement
EME	Emergency Mitigating Equipment
EMP	Environmental Monitoring Program
EPRI	Electric Power Research Institute
ERA	Environmental Risk Assessment
FHA	Fire Hazards Assessment

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Acronym	Explanation
FIA	Foundation Interface Analysis
FPC	Fuel Pool Cooling and Cleanup System
HCSC	Hazard-Consistent, Strain-Compatible
HU	Hydrostratigraphic Unit
HVAC	Heating, Ventilation, and Air Conditioning
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICC	ICS Pool Cooling and Cleanup System
ICS	Isolation Condenser System
INPO	Institute of Nuclear Power Operations
LOCA	Loss-of-Coolant Accident
LOPP	Loss-of-Preferred Power
LTC	Licence to Construct
MCA	Main Condenser and Auxiliaries
MCR	Main Control Room
NHS	Normal Heat Sink
NSCA	Nuclear Safety and Control Act
OPG	Ontario Power Generation
PCW	Plant Cooling Water System
PEOC	Provincial Emergency Operations Centre
PIE	Postulated Initiating Event
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PNERP	Provincial Nuclear Emergency Response Plan
PNGS	Pickering Nuclear Generating Station
POSAR	Pre-Operational Safety Analysis Report
PPE	Plant Parameter Envelope
PSA	Probabilistic Safety Assessment
PSHA	Probabilistic Seismic Hazard Assessment
RB	Reactor Building
RPV	Reactor Pressure Vessel
RWB	Radwaste Building

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Acronym	Explanation
SA	Severe Accident
SAA	Severe Accident Analysis
SAM	Severe Accident Management
SAMG	Severe Accident Management Guideline
SCR	Secondary Control Room
SMR	Small Modular Reactor
SPT	Standard Penetration Test
SRA	Site Response Analysis
SSI	Soil-Structure Interaction
SSC	Structures, Systems, and Components
TB	Turbine Building
TLD	Thermoluminescent Dosimeter
UCS	Uniaxial Compression Stress
UHS	Uniform Hazard Response Spectrum
USNRC	United States Nuclear Regulatory Commission
WPCP	Water Pollution Control Plant

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2.0 SITE CHARACTERISTICS

Information in Chapter 2 details the site characteristics and their evaluation in support for the design, safety assessment and periodic safety review (Reference 2.0-4) of the Boiling Water Reactor, 10th Design – 300 MWe (BWRX-300) facility (also known as BWRX-300 facility). Over the planned design life (refer to Chapter 1, Table 1.5-1) of the BWRX-300 facility, the information in Chapter 2 will periodically be updated (Reference 2.0-4) to risk-inform the evaluation and implications of any such updates on safety.

Chapter 2 includes the following characteristics of Ontario Power Generation's (OPG) Darlington New Nuclear Project (DNNP) site and the surrounding region:

- Geography and Demography (Section 2.1)
- Evaluation of Site-specific Hazards (Section 2.2)
- Proximity of Industrial, Transportation and Other Facilities (Section 2.3)
- Plant Site Activities Influencing Plant Safety (Section 2.4)
- Hydrology (Section 2.5)
- Meteorology (Section 2.6)
- Geology, Seismology, and Geotechnical Engineering (Section 2.7)
- Potential Effects of Nuclear Power Plants in the Region (Section 2.8)
- Radiological Conditions due to External Sources (Section 2.9)
- Site-related Issues in Emergency Preparedness and Response, and Accident Management (Section 2.10)
- Monitoring of Site-related Parameters (Section 2.11)

Chapter 2 also includes Section 2.12 which describes OPG's disposition plans to finalize remaining DNNP site-specific characterization work including, for example, Foundation Interface Analysis (FIA), confirmatory site geological and seismic hazard investigations, and climate change effects on-site hydrological and meteorological parameters.

The following key chapters should be referred for additional information relevant to the material reported in Chapter 2:

1. Chapter 1: Introduction and General Considerations

Information in Chapter 1, Sections 1.4 and 1.5 describes the DNNP site layout, as well as the BWRX-300 facility footprint, key parameters, and basic dimensions of key buildings in the Power Block.

2. Chapter 3: Safety Objectives and Design Rules for Structures, Systems, and Components

Chapter 3, Section 3.3 includes information on the BWRX-300 design approach to prevent and mitigate the effect of external hazard on safety-classified structures, systems, and components (SSCs). Also, Chapter 3, Subsection 3.5.5.2 describes the design loads and load combinations on the deeply embedded Reactor Building (RB) structure.

3. Chapter 6: Engineered Safety Features

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Information is provided in Chapter 6, Section 6.2 on the design of the Isolation Condenser System; and in Section 6.4 on the BWRX-300 control room habitability features including missile protection, radiation shielding, radiation monitoring, air filtration and ventilation systems, lighting, and fire protection.

4. Chapter 7: Instrumentation and Control

Measures for fire protection and qualification for electromagnetic compatibility are described in Chapter 7.

5. Chapter 9A: Auxiliary Systems

Chapter 9A presents information on the BWR-X-300 fuel storage and handling system in Subsection 9A1.2, Fuel Pool Cooling and Cleanup System (FPC) in Subsection 9A1.3, Plant Cooling Water System (PCW) in Subsection 9A.2.1, Normal Heat Sink (NHS) in Subsection 9A.2.5, Isolation Condenser System Pool Cooling and Cleanup System (ICC) in Subsection 9A.2.6, Heating, Ventilation, and Air Conditioning (HVAC) Systems in Section 9A.5, Fire Protection Systems, in Section 9A.6.

6. Chapter 9B: Civil Engineering Works and Structures

General design requirement information is provided in Chapter 9B, Section 9B.2 on the integrated RB, and Section 9B.3 on other structures including other buildings in the Power Block, the Pumphouse/Forebay as well as the intake and discharge tunnels.

7. Chapter 10: Steam and Power Conversion Systems

In Chapter 10, information related to equipment functions, design basis, operation, and maintenance is presented in Section 10.5 for the Main Condenser and Auxiliaries (MCA) system, and in Section 10.8 for the Circulating Water System (CWS).

8. Chapter 15: Safety Analysis

Chapter 15, Subsection 15.5.3 documents the Deterministic Safety Analysis (DSA) of bounding Baseline Abnormal Operational Occurrences (BL-AOOs), while 15.5.4 evaluates the bounding BWRX-300 Design Basis Accidents (DBAs) involving Loss-of-Coolant Accidents (LOCA) and non-LOCA. Also, Subsections 15.5.5 and 15.5.6 present analyses of Design Extension Conditions (DECs) with and without core damage, respectively. Furthermore, Subsection 15.6.1 described the general approach to the Probabilistic Safety Analysis (PSA) while Section 15.7 includes results of analyzed DSA and PSA bounding events. Finally, Appendix 15A demonstrates implementing Defence-in-Depth (D-in-D) provisions ensures protection against unacceptable radiation releases

9. Chapter 19: Emergency Preparedness and Response

The development of the DNNP nuclear emergency response plan is presented in Section 19.1, the emergency response facilities are described in Section 19.2, and the accident assessment techniques are detailed in Section 19.3.

10. Chapter 20: Environmental Aspects

Chapter 20 describes OPG's Environmental Monitoring Program in Subsection 20.11.2, Effluent Monitoring Program in Subsection 20.11.3, and Groundwater Monitoring Program in Subsection 20.11.4.

11. BWRX-300 Security Annex

The prescribed information in the Security Annex documents the analysis of a large commercial aircraft crash.

Scope

Chapter 2 scope includes the establishment of site characteristics that comprise information such as:

1. The site location, the area under control of OPG, and the area surrounding the DNNP site including activities which impact BWRX-300 facility operation, population distribution and density (Section 2.1), and the locations and transport routes that present potential risk for the facility (Section 2.3).
2. The site-specific external hazard evaluation (Section 2.2) for events of natural and human-induced origin during the planned lifetime of the facility, and any process or activity at the site that affects the operation of the facility (Section 2.4).
3. The collection of DNNP site-specific baseline data such as hydrological (Section 2.5); meteorological (Section 2.6); as well as geological, seismological, geotechnical (Section 2.7) information.
4. The description of the site and the surrounding environment (Sections 2.8), and of external sources related to the dispersion of radioactive material in air, water, and soil (Section 2.9).
5. The feasibility of emergency preparedness as related to accessibility and transport of any pertinent equipment to the DNNP site and the BWRX-300 facility (Section 2.10).
6. The arrangements for monitoring site-related parameters (Section 2.11) throughout the lifetime of the facility.

Relevant Legislations and Regulations

The following provisions of the Nuclear Safety and Control Act (Reference 2.0-1), the General Nuclear Safety and Control Regulations (Reference 2.0-2) and the Class I Nuclear Facilities Regulations (Reference 2.0-3) are relevant to Chapter 2.

- Subsection 44(1) of the NSCA (Reference 2.0-1) states that “[t]he Commission may, with approval of the Governor in Council, make regulations.
(e) Respecting the location, design, construction, installation, operation, maintenance, modification, decommissioning, abandonment and disposal of a nuclear facility or part of a nuclear facility.
(o) Establishing requirements to be complied with by any person who possesses, uses, packages, transports, stores, or disposes of a nuclear substance or prescribed equipment or who locates, designs, constructs, installs, operates, maintains, modifies, decommissions, or abandons a nuclear facility or nuclear-powered vehicle.
- Section 3 of the Class I Nuclear Facilities Regulations (Reference 2.0-3) states that “[a]n application for a licence in respect of a Class I nuclear facility, other than a licence to abandon, shall contain the following information in addition to the information required by Section 3 of the General Nuclear Safety and Control Regulations (Reference 2.0-2):
 - a. A description of the site of the activity to be licensed, including the location of any exclusion zone and any structures within that zone
 - b. Plans showing the location, perimeter, areas, structures, and systems of the nuclear facility
 - c. Proposed management system for the activity to be licensed, including measures to promote and support safety culture

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- d. Name, form, characteristics, and quantity of any hazardous substances that may be on the site while the activity to be licensed is carried on
- e. Proposed worker health and safety policies and procedures
- f. Proposed environmental protection policies and procedures
- g. Proposed effluent and environmental monitoring programs
- Section 5 of the Class I Nuclear Facilities Regulations (Reference 2.0-3) states that: “[a]n application for a licence to construct a Class I nuclear facility shall contain the following information in addition to the information required by Section 3:
 - a. Description of the proposed design of the nuclear facility, including the manner in which the physical and environmental characteristics of the site are considered in the design
 - b. Description of the environmental baseline characteristics of the site and the surrounding area
 - c. Effects on the environment and the health and safety of persons that may result from the construction, operation and decommissioning of the nuclear facility, and the measures that will be taken to prevent or mitigate those effects
 - d. Proposed location of points of release, the proposed maximum quantities and concentrations, and the anticipated volume and flow rate of releases of nuclear substances and hazardous substances into the environment, including their physical, chemical, and radiological characteristics

References

- 2.0-1 Government of Canada, “Nuclear Safety and Control Act (S.C. 1997, c. 9).”
- 2.0-2 Government of Canada SOR/2000-202, “General Nuclear Safety and Control Regulations.”
- 2.0-3 Government of Canada SOR/2000-204, “Class I Nuclear Facilities Regulations.”
- 2.0-4 CNSC Regulatory Document REDGOC-2.3.3, “Operating Performance - Periodic Safety Reviews.”

2.1 Geography and Demography

Section 2.1 details the geographical and demographical baseline characteristics of the DNNP site and the surrounding regions. It contains the following information:

- Darlington Nuclear site context and surrounding land uses - Subsection 2.1.1
- BWRX-300 facility layout and the exclusion zone - Subsection 2.1.2
- Population distribution and density - Subsection 2.1.3
- Municipal services - Subsection 2.1.4
- Site access and transportation networks - Subsection 2.1.5
- Public transit – Subsection 2.1.6
- Active hiking and cycling trails - Subsection 2.1.7
- Parks spaces and waterbodies - Subsection 2.1.8
- Industrial facilities - Subsection 2.1.9

Table 2.1-1 lists key geographic and demographic characteristics and parameters within a 10-km survey area surrounding the Darlington Nuclear site.

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Table 2.1-1: Site Layout, Geographic, and Demographic Characteristics and Parameters

Characteristic	Value/Description	
Land Size	Darlington Nuclear site	Approximately 4.9 km ²
	DNNP	Approximately 1.8 km ²
	DNGS	Approximately 3.1 km ²
Exclusion Zone	BWRX-300	350 m (radius) from the RB outside wall
	DNGS	914 m
Topography	<ul style="list-style-type: none"> Current parking and storage areas east of the DWMF is at approximately 88 m (Canadian Geodetic Datum of 1928 (CGVD28), or simply CGD)) Further east, the terrain rises to 102 m CGD close to the Darlington Creek watershed Extreme berm of elevation from 100 to 110 characterize the north boundary of the southern portion of the site to the railway tracks The northern portion of the site is bounded the north by Energy Road and to the south by the Railway tracks East of Holt Road, the terrain peaks at 120 m CGD and slopes down to the east to roughly 86 m CGD 	
Grade Elevation	Plant (BWRX-300 Facility)	88 m CGD (Refer to Subsection 2.7.1)
Population Distribution and Density (2021), for the Municipality of Clarington	Courtice	28,545
	Bowmanville	47,176
	Orono	2,476
	Newcastle	11,933
	Total	90,130
Municipal Service within the 10-km Survey area	Fire Emergency Stations	6 (Excluding DNGS site fire station)
	Regional Police Station	One (plus one administrative police department)
	Hospitals	One (Lakeridge Health in Bowmanville)
Directly Adjacent Industrial Facilities	East	St. Marys Cement Group
	West	<ul style="list-style-type: none"> Darlington Nuclear Energy Complex CoPart, Vehicle Auction Facility Covanta Durham York Energy Centre Courtice Water Pollution Control Plant (WPCP) East Penn, Batteries warehouse facility Future Anaerobic Digester facility
Transportation network within 10 km	Highways	401, 407, 418
	Railways lines	<ul style="list-style-type: none"> Canadian National, south of Highway 401 and bisects the site Canadian Pacific, north of Highway 401
	Airports	Oshawa Executive Airport
	Naval Ports	Port of Oshawa East Pier

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Characteristic	Value/Description	
	Private Dock	Private dock on St. Marys facility
Public Transit	Bus (902A King bus line)	One stop at Old Holt Road and King Street
	Transit-on-demand	Request pick up to nearest transit stop
	Rural-on-demand	Request pick up at current location
	88 GO Bus	Multiple stops along Bowmanville Avenue and King Street
	GO Transit's Lakeshore East Rail Service (planned for operation in 2026)	Courtice GO Station
		Bowmanville GO Station
Hiking and Cycling Trails	Darlington Waterfront Trail	Pedestrian and cyclists trail
Parks Spaces and Waterbodies (Note: A complete list is provided in Appendix C)	Provincial Parks	One – Darlington Provincial Park
	Recreational Facilities	Darlington Hydro Soccer Field and Bowmanville Baseball Fields
	Conservation Areas	Five in Bowmanville and two in Oshawa
	Beaches	Three – Two in Bowmanville and one in Oshawa
Industrial Facilities within 10 km	<ul style="list-style-type: none"> • Directly adjacent industrial facilities, refer to Subsection 2.1.1 • A complete list of industrial facilities falling within the surveyed area is found in Appendix A. • Pickering Nuclear Generating Station, about 25 km west of DNNP 	

2.1.1 Darlington Nuclear Site Context and Surrounding Land Uses

Site Topography

The Darlington Nuclear site topography is briefly described in Subsection 2.7.1. The 2022 Flood Hazard Assessment NK054-REP-02730-00001 (Reference 2.1-9) provides in this Subsection 2.1.1 additional information on the site topography including key detailed terrain elevations, as briefly recapped in the following paragraph.

The Darlington Nuclear site is situated in an undulating to moderately rolling limestone till plain, although its natural contours have been extensively graded. The existing 4-unit Darlington Nuclear Generating Station (DN GS) is located at elevation of about 78 m CGD. This is the lowest elevation area of the southern portion of the Darlington Nuclear site. From this location, the site slopes upward to the northwest, north and east. To the east, the terrain steadily slopes upward along the Lake Ontario shoreline, forming a bluff. The DNNP site, currently a parking and storage area southeast of the Darlington Waste Management Facility (DW MF), is just north of shoreline bluff, at approximately 88 m CGD. Farther east, the terrain rises to elevation 102 m CGD at the boundary of the Darlington Creek watershed before sloping down to its main branch near the eastern boundary of the site. The north boundary of the southern portion of the Darlington Nuclear site is characterized by an extensive berm that ranges in elevation from 100 m CGD to 110 m CGD and separates the southern portion of the site from the transecting Canadian National Railway tracks. The northern portion of the site is bounded to the north by Energy Drive and to the south by the Canadian National Railway tracks. Between Crago Road and Park Road, there is a large ridge rising to 132 m CGD. Between Park Road and Holt Road, the terrain ranges from 98 m to 130 m CGD. East of Holt Road, the DNNP terrain peaks at 120 m CGD and slopes downward to the east to roughly 86 m CGD.

Area and Bounding Roads

The Darlington Nuclear site is approximately 4.9 km² in size and located within the Municipality of Clarington, Regional Municipality of Durham, Province of Ontario, Canada. OPG also owns and operates the eight-unit Pickering Nuclear Generating Station (PNGS) (refer to Subsection 2.2.5.2) within the City of Pickering which is located approximately 25 km to the west of the Darlington Nuclear site, as shown in Figure 2.1.1-1.

The Darlington Nuclear site encompasses both the DN GS and the DNNP lands as shown in Figure 2.1.1-2. The Darlington Nuclear site is bounded by Crago Road to the west, Energy Drive to the north, St. Marys Cement to the east and Lake Ontario to the south. The existing DN GS site is approximately 3.1 km² in size and is located west of Holt Road on the western portion of the Darlington Nuclear site, whereas the DNNP land of approximately 1.8 km² is located east of Holt Road. Figure 2.1.1-2 shows also the 914-meter DN GS exclusion zone, which partly overlaps the location where the BWRX-300 first unit is to be built in the southwestern corner of the DNNP site as shown in Chapter 1, Figure A1.1-2.

Industrial Facilities

The major industrial facilities in the vicinity of the Darlington Nuclear site, as shown in Figure 2.1.1-3, include:

1. St. Marys Cement Group which is located directly east of the DNNP site on Bowmanville Avenue, and is an active quarry for resources servicing the aggregate and concrete industry
2. The lands designated as Clarington Energy Business Park which is located directly west of the DN GS and includes:

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- a. Covanta Durham York Energy Centre which manages household waste from the regions of Durham and York
 - b. OPG's Darlington Energy Complex, an approximately 27,900 m² multi-use building that provides offices and services supporting the Darlington Refurbishment project
 - c. CoPart, a vehicle auction and recycling facility
 - d. East Penn, a warehousing facility for batteries
 - e. Courtice Water Pollution Control Plant (WPCP), a wastewater treatment facility commissioned in late 2007, with an average day rated capacity of 68.2 million liters per day with a peak flow capacity of 180 million liters per day (Reference 2.1-7)
 - f. Planned location for a project that is being evaluated involving an Anaerobic Digester facility (Reference 2.1-7) to treat raw sludge collected from Courtice WPCP
3. OWASCO RV, which is a recreational vehicle sale and service centre, located north of Highway 401

There are some industrial developments in the Courtice Employment Area located northwest of the Darlington Nuclear site, including warehousing and automobile dealerships. All of the industrial facilities falling within the surveyed area are listed in Appendix A.

Developmental Activities

OPG actively reviews planning applications in the Municipality of Clarington to monitor sensitive land use developments within 3 km of the DNGS and DNNP facilities. Additionally, OPG reviews planning applications within 10 km of the Darlington Nuclear site in the Municipality of Clarington and the City of Oshawa. These applications include official plan amendments, zoning by-law amendments, draft plans of subdivision and condominium, and other miscellaneous planning related documents.

OPG completes an annual development activity report detailing all proposed developments in the municipalities of Clarington and Oshawa within 10 km of the Darlington Nuclear site. In such a report, OPG reviews the:

- a. Type and location of proposed application
- b. Date on which the application was submitted
- c. Details of the proposed application
- d. Status of the application

Urban Communities and Rural Areas

The urban communities of Oshawa and Courtice are located northwest of the Darlington Nuclear site, while the urban community of Bowmanville is located to the northeast of the DNNP site. A rural area separating the Clarington urban areas of Courtice and Bowmanville is located immediately north of the DNNP site. The community of Newcastle is also located east of the DNNP site within the survey area; albeit only a portion is included in the survey area. For the purposes of Section 2.1 and Section 2.3, the geographic limits defined for the survey area are approximately 10 km from the site and include Taunton Road to the north, Simcoe Street to the west, an approximate border of Darlington Clarke Townline Road to the east, and Lake Ontario to the south (refer to Figure 2.1.1-4).

Land Use Assessment for Environmental Effects

The 10 km survey area is consistent with the Land Use Assessment Zone, which was the furthest distance that measurable effects on planned land use structure as well as impacts on sensitive land uses are identified in the proximity to the Darlington Nuclear site. The Land Use Assessment of Environmental Effects Technical Support Document completed in 2009 identified the Regional Study Area as being approximately 50 km from the Darlington Nuclear site as shown in Figure 2.1.1-4. The DNNP Land Use Environmental Assessment Follow-Up Monitoring Plan / Methodology Report was developed in 2022 NK054-CORR-00531-10635 (Reference 2.1-3) to fulfill the requirement of OPG Commitment D-P-12.7 in the 2021 NK054-REP-01210-00078 (Reference 2.1-2). As per the 2022 NK054-CORR-00531-10635 (Reference 2.1-3), OPG will continue to monitor planning development in land use in proximity to the DNNP site, and regularly consult with the Municipality of Clarington, City of Oshawa and the Regional Municipality of Durham on proposed land use changes. The effects on implementation of emergency plans will be investigated throughout the site preparation and construction phases.

2.1.2 BWRX-300 Facility Layout and Exclusion Zone

The layouts of the DNNP site and BWRX-300 Unit 1 as well as associated infrastructures are described in Chapter 1, Section 1.4, and Section 1.5 satisfy the regulatory requirements of Sections 4.5.4 and 4.5.5 of REGDOC-1.1.2 (Reference 2.1-1). The selected location, in the southwestern corner of the DNNP area, limits the amount of spoilage to remove and avoids encroachment on the Bank Swallow habitat. This location is also in proximity to DNGS ensuring effective connections to DNGS available infrastructure. The DNNP site also incorporates considerations that support a total of four BWRX-300 units, as conceptually shown in Figure 5 of the 2022 DNNP BWRX-300 Environmental Impact Statement (EIS) NK054-REP-07730-00055 (Reference 2.1-4).

The deployment of the BWRX-300 facility does not require expanding the DNGS switchyard. Rather, a new 230 kV switchyard is to be located East of the Extended Holt Rd and South of the Canadian National Railway tracks, adjacent to the BWRX-300 facility buildings, as shown in Chapter 1, Figure A1.1-2 and Figure A1.4-2 for one unit and conceptually shown in Figure 7 of the 2022 NK054-REP-07730-00055 (Reference 2.1-4) for four units.

Existing roads are being used to the maximum extent practicable and no new off-site roadways are required.

The Pumphouse/Forebay structure is positioned outside the northwestern corner of the protected area. As described in Chapter 9B, Subsection 9B.3.5.2, onshore vertical shafts are designed to facilitate the operation of up to four BWRX-300 units and the construction of the intake and discharge tunnels. The intake tunnel conveys cooling water from the lakebed intake structure to the onshore intake vertical shaft. The discharge tunnel conveys the discharge water from the onshore discharge vertical shaft to the discharge tunnel and diffusers. The discharge structure is located near the lakeshore and does not require lake infill.

2.1.2.1 Required Exclusion Zones

The exclusion zone is established at 350 m from the RB outside wall. For the BWRX-300 first unit, the exclusion zone partly overlaps the eastern portion of the DNGS site, as shown in Chapter 1, Figure A1.1-2. The exclusion zone of 350 m for the conceptual layout of four units shown in Figure 5 of the 2022 NK054-REP-07730-00055 (Reference 2.1-4) is within the DNNP eastern boundary with St. Mays Cement industrial facility.

The rationale for determining the exclusion zone is discussed in Section 8 of the 2022 NK054-REP-01210-00142 (Reference 2.1-5), and considers the security requirements, evacuation

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needs, land usage needs, and environmental conditions, in accordance with Section 6.5 of REGDOC-2.5.2 (Reference 2.1-8). Note the BWRX-300 Small Modular Reactor (SMR) is built within the DNNP site boundary with a smaller footprint of approximately 9,800 m², per the 2022 NK054-REP-01210-00142 (Reference 2.1-5), compared with the original application involving much larger nuclear power plants, per the 2010 NK054-REP-01200-10000 (Reference 2.1-6).

Chapter 15, Section 15.7 includes tabulated summaries listing the DSA results for bounding BWRX-300 AOO and DBA event sequences. Also, Chapter 15, Appendix 15A demonstrates implementation of the D-in-D provisions ensures protection against unacceptable radiation releases. Chapter 15, Section 15.7 thus concludes all BWRX-300 analyzed bounding AOOs, DBAs or DECAs without core damage have met the dose acceptance criteria for the 350 m exclusion zone.

2.1.2.2 Security Requirements

The security requirements for the DNNP site and the BWRX-300 facility and how such security requirements are met are described in the Security Annex, which is an OPG Confidential Protected Security document.

2.1.2.3 Description of Site Layout

The high-level description of the DNNP site layout includes:

- The Power Block that encompasses several buildings and a plant services area (refer to Chapter 1, Figure A1.5-1)
- Locations of the site vehicle entrance (sally port) as well as roads to allow access of trucks and individuals to Power Block buildings, with the Protected Area Access Building located west of the sally port (refer to Chapter 1, Figure A1.4-1)
- Locations of the irradiated fuel dry storage (which is regulated under a separate licence), Pumphouse/Forebay, intake shaft and tunnel, discharge structure and tunnel, and switchyard and transmission lines (refer to Chapter 1, Figure A1.1-2)
- Heavy haul routes for the construction phase of Unit 1 as shown in Chapter 1, Figure A1.1-2, and for the construction phases of Units 2, 3, and 4, as shown in Figure 5 of the 2022 EIA (Reference 2.1-4).

2.1.2.4 Minimizing Environmental Impacts

Measures are included in the DNNP site layout and BWRX-300 design to minimize the impact on the surrounding region and the environment, per the 2022 NK054-REP-07730-00055 (Reference 2.1-4), for example:

1. The location and placement of the lakebed intake structure regarding the commitment for fish entrainment and impingement as well as the discharge diffusers to meet the commitment for effluent plume in the 2021 NK054-REP-01210-00078 (Reference 2.1-2)
2. Consideration of sensitive land features, such as shoreline bluffs and Bank Swallows, habitat to the extent practicable
3. A smaller BWRX-300 footprint which does not need any additional land area that could be obtained from lake infill
4. Designing into the site storm water management provisions for the construction and post construction phases

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5. Minimizing the area of disturbance for permanent structures as well as the areas for spoils on the DNNP site by optimizing the BWRX-300 footprint
6. Cooling towers are not used for the BWRX-300 for either the normal or ultimate heat sinks, per Table 3 of the 2022 EIS NK054-REP-07730-00055 (Reference 2.1-4); thus, the adverse effects associated with cooling towers (e.g., effects on the visual landscape and socio-economic conditions) are not applicable
7. The primary and secondary heat transport systems are combined, and use is made of natural circulation and passive safety systems resulting in an optimized size of the facility and contributing to lowering the risk of normal and abnormal operating conditions

2.1.3 Population Distribution and Density

The Municipality of Clarington and the City of Oshawa have both experienced steady growth over the last ten years.

According to recently released Statistics Canada data, Clarington's population was 101,427 in 2021, which is an increase of 10.2% from that in 2016 when the population was recorded at 92,130. The rural area of Clarington had a population of 11,297 in 2021. The Municipality of Clarington Official Plan forecasts that Clarington will have a population of 140,340 by 2031, with 124,685 in its urban areas and 15,655 in its rural areas. The 2021 population data listed in Table 2.1-2 for the Municipality of Clarington is distributed amongst four urban areas including Courtice, Bowmanville, Orono, and Newcastle as shown in Figure 2.1.3-1.

Table 2.1-2: Population Data for the Municipality of Clarington for 2021

Urban Area	Population
Courtice	28,545
Bowmanville	47,176
Orono	2,476
Newcastle	11,933
Total	90,130

The population of the City of Oshawa was 149,607 in 2011 and grew to 159,458 in 2016, which was a 6.6% increase. The City of Oshawa's Official Plan provides population forecasts of 174,695 in 2021, 184,460 in 2026 and 197,000 in 2031.

Refer to Subsection 2.8.4 for detailed 2016 population data that is broken into sectors by distance and direction for use in air dispersion modeling within a 30 km radius of the Darlington Nuclear site.

2.1.4 Municipal Services

Within the 10 km survey area, there are 17 education institutions available for students: 12 primary schools and five secondary schools. As well, there are six fire emergency stations (excluding OPG's on-site Darlington fire station) and one regional police station (plus one administrative police department). Additionally, there is one hospital - Lakeridge Health in Bowmanville.

2.1.5 Site Access and Transportation Networks

The Darlington Nuclear site can be accessed via two roads. Holt Road runs north to south and allows for direct access to the site. Energy Drive runs west to east and connects to Park Road for access to the site. Multiple parking lots are present on the site.

Within 10 km of the site, there are many arterial roads, minor arterial roads, highways, residential roads, and rural roads. These roads fall within the borders of the 10 km survey area defined in Subsection 2.1.1. A complete list of roads falling within the surveyed area can be found in Appendix B.

Transportation networks of significance are listed in the following:

1. Three 400-series highways are located within 10 km of the site - Highways 401, 407, and 418 (refer to Subsection 2.3.1(b) for supplementary information on Highway 401).
2. Two railway lines are located within 10 km of the site which converge and run adjacent to one another east of Lakeshore Road, Newcastle:
 - a. The Canadian Pacific line runs west east, which is located just north of Highway 401, and is used for trains transporting cargo.
 - b. The Canadian National line runs west east, which is located south of Highway 401 and used for trains transporting people and cargo, and part of which bisects the DNNP and DNGS sites (refer to Subsection 2.3.1 for further information, and Subsection 2.2.3.2(a) for hazards related to potential railway accidents).
3. Oshawa Executive Airport is located at the southeast corner of Taunton Road and Thornton Road North. The airport is located just outside the 10 km survey area (refer to Subsection 2.3.1(c) for additional information).
4. The Port of Oshawa East Pier (at the bottom of Simcoe Street South) is located west of the site and allows cargo ships to receive/deliver shipments.
5. St. Marys Cement has a private dock at its facility to the east of the DNNP site for the shipment of aggregate from its operations.

2.1.6 Public Transit

The closest regional transit stop to the site is located at Old Holt Road and King Street, approximately 5 km north of the site. The stop is part of the 902A King bus line offered by Durham Regional Transit and runs west east through the Durham Region. Additionally, the region introduced two types of on-demand transportation services in the Durham Transportation Master Plan (2017): transit on-demand and rural on-demand. Transit on-demand allows riders to request a ride with pickup located at their nearest transit stop, while rural on-demand allows riders to request a ride with pickup at their current location. The region also has a park and ride station within the survey area located at Courtice Road north of Highway 401.

The closest transit stop to the site is a GO Bus stop located at Bowmanville Avenue and Baseline Road. The stop is part of the 88 GO Bus Route that is running from Oshawa to Peterborough with multiple bus stops located north of the site along Bowmanville Avenue and King Street. Additionally, there are two proposed GO Transit stations within the survey area. GO Transit's Lakeshore East Rail Service will operate on the Canadian Pacific rail line north of Highway 401, which will include service to the two proposed stations: Courtice GO (Courtice Road north of Baseline Road) and Bowmanville GO (Bowmanville Avenue north of Aspen Spring Drive). Per correspondence with Durham Region staff, the Courtice and Bowmanville GO stations are

projected to be operational in 2026. Furthermore, two secondary plans are currently being developed for the areas adjacent for each proposed GO station.

2.1.7 Active Hiking and Cycling Trails

As shown in Figure 2.1.8-1, the Darlington Waterfront Trail, part of the Great Lakes Waterfront Trail, is a multi-use path that forms part of the recently approved Durham Regional Cycling Plan. The trail is used by pedestrians and cyclists for transportation or recreational purposes, provides direct access to the Darlington Nuclear site and falls within OPG owned lands. Additionally, hiking trails are available near Lakeview Park in Oshawa, as the Larry Ladd Harbour Trail connects to Lakeview Beach. The Primary Cycling Network Durham currently provides over 400 km of cycling infrastructure in the region.

2.1.8 Park Spaces and Waterbodies

There is abundance of parks, greenspaces, conservation areas, and waterbodies located within the 10 km survey area, with multiple public recreational spaces directly adjacent to Darlington Nuclear site. As detailed in Subsection 2.1.7, part of the Darlington Waterfront Trail runs through the Darlington Nuclear site. Directly adjacent to the west of the DNGS site is Alijco Beach, a beachfront which can be accessed by users for recreational purposes. Other park spaces and waterbodies are dispersed throughout the rest of the survey area, with places of significance listed below:

1. One provincial park falls within the survey area - Darlington Provincial Park.
2. The Darlington Hydro Soccer Fields facility (owned by OPG and licensed to the Municipality of Clarington) falls within the survey area, as does Bowmanville's Baseball Fields Complex (located at Green Road just north of Highway 401).
3. Five conservation areas fall within the survey area: three are located in Bowmanville (Bowmanville Valley Conservation Area, Bowmanville Westside Conservation Area, Stephen Gulch's Conservation Area) and two are located in Oshawa (Harmony Valley Conservation Area, Oshawa Valleylands Conservation Area).
4. Three beaches fall within the survey area: two are located in Bowmanville (Alijco Beach, Port Darlington Beach) and one is located in Oshawa (Lakeview Beach).

A complete list of park spaces and water bodies falling within the surveyed area can be found in Appendix C.

2.1.9 Industrial Facilities

The industrial facilities that are within the survey area of 10 km and directly adjacent to Darlington Nuclear site are discussed in Subsection 2.1.1.

Other industrial facilities are dispersed throughout the rest of the survey area, with most facilities located west of the site in Oshawa. A complete list of industrial facilities falling within the surveyed area is found in Appendix A.

While not located in the survey area, the PNGS is located approximately 25 km west of the Darlington Nuclear site (refer to Subsection 2.1.1 and Subsection 2.2.5.2).

2.1.10 References

- 2.1-1 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 2.1-2 NK054-REP-01210-00078 R007, 2021, "Darlington New Nuclear Project Commitments Report," Ontario Power Generation.

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- 2.1-3 NK054-CORR-00531-10635, 2022, "DNNP: Submission of Environmental Assessment Follow-Up Monitoring Plans / Methodology Reports and Request for Acceptance and Closure of Their Respective Commitments under D-P-12," Ontario Power Generation.
- 2.1-4 NK054-REP-07730-00055-R000, 2022, "Darlington New Nuclear Project Environmental Impact Statement Review Report for Small Modular Reactor BWRX-300," Ontario Power Generation.
- 2.1-5 NK054-REP-01210-00142-R000, 2022, "Darlington New Nuclear Project – Site Evaluation Update Summary Report," Ontario Power Generation.
- 2.1-6 NK054-REP-01200-10000 R005, 2010, "Use of Plant Parameters Envelope to Encompass the Reactor Designs being considered for the Darlington Site," Ontario Power Generation.
- 2.1-7 Durham Region, "Courtice Water Pollution Control Plant - 2021 Annual Performance Report."
- 2.1-8 CNSC Regulatory Document REGDOC-2.5.2, Version 1.0, "Design of Reactor Facilities: Nuclear Power Plants."
- 2.1-9 NK054-REP-02730-00001, 2022, "Flood Hazard Assessment," Ontario Power Generation.



Figure 2.1.1-1: Darlington Nuclear Site Proximity to Pickering Nuclear Generating Station

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2-18

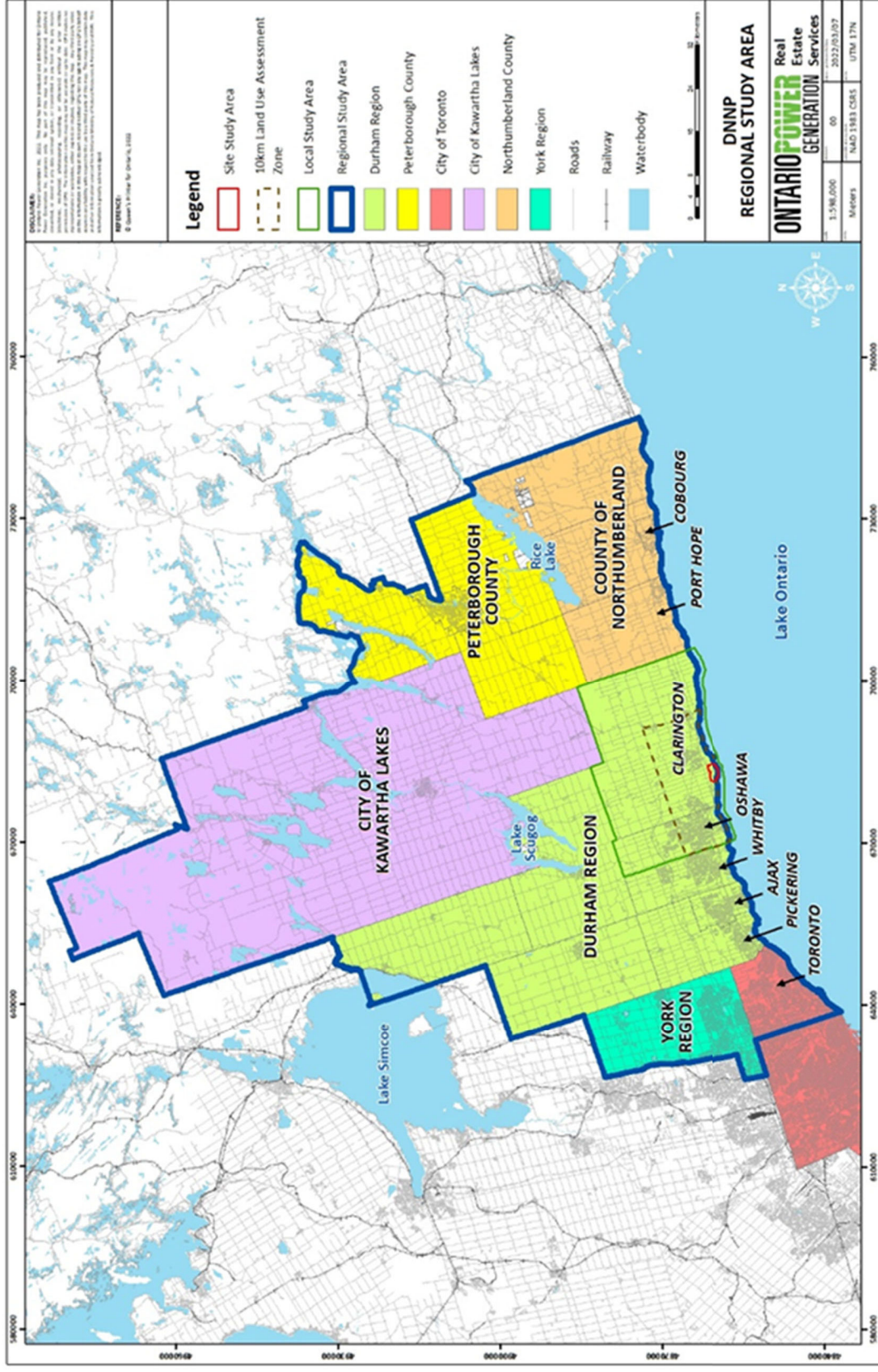


Figure 2.1.1-4: DNNP Regional Study Area

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2.2 Evaluation of Site-Specific Hazards

Section 2.2 characterizes and quantifies site-specific hazards that are used in the design of the BWRX-300 and builds upon the 2022 DNNP Hazard Analysis Methodology NK054-REP-01210-00144 (Reference 2.2-10). As the DNNP and DNGS share the Darlington Nuclear site (refer to Subsection 2.1.1), the DNGS 2019 Hazard Screening Analysis NK38-REP-03611-10043 (Reference 2.2-5) is used in support of Section 2.2 and to inform the DNNP hazard screening analysis. All such site characteristics are validated for the BWRX-300 Unit 1 design and its location on the DNNP site, as shown in Chapter 1, Figure A1.1-2.

2.2.1 Introduction

Section 2.2 includes the methodology used for and the results of the evaluation of site-specific external hazards associated with the DNNP site and the BWRX-300 facility. Such evaluation is derived from previous DNNP hazards assessment work completed in the 2009 NK054-REP-01210-00008 (Reference 2.2-1) and the 2009 NK054-REP-01210-00019 (Reference 2.2-2) as well as from a 2019 DNNP site preparation licence renewal activity report NK054-REP-01210-00108 (Reference 2.2-3). The evaluation addresses specific items relevant to DNNP site-specific external hazards, as identified in the 2020 OPG's application to renew the DNNP site preparation licence NK054-CORR-00531-10533 (Reference 2.2-4).

The methodology used to evaluate external hazards is described in Subsection 2.2.2.

The hazards identified for further evaluation are:

- Subsection 2.2.3: Transportation Accidents, Including Toxic Chemical or Gas Releases / Explosions Hazards
- Subsection 2.2.4: Stationary Non-nuclear Accidents Hazards
- Subsection 2.2.5: Stationary Nuclear Accidents Hazards
- Subsection 2.2.6: Industrial Hazards
- Subsection 2.2.7: Biological, Animal, and Frazil Ice Hazards
- Subsection 2.2.8: Ice Storm Hazard
- Subsection 2.2.9: Electromagnetic Interference Hazard
- Subsection 2.2.10: On-site Methane Hazard

A summary results and follow-up considerations of the hazards listed above are provided in Table 2.2-1.

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**Table 2.2-1: Screening and Validation of CNSC–Identified DNNP
Site-Specific Hazards**

2.2.2 External Hazards Evaluation Methodology			
Methodology	The methodology and criteria used in the 2019 DNGS NK38-REP-03611-10043 (Reference 2.2-5) Comparable methodology and criteria developed in the 2022 BWRX-300 DNNP NK054-REP-01210-00144 (Reference 2.2-10)		
Screening Criteria	Qualitative Criteria – QL-1 to QL-7 Quantitative criteria – QN-1 to QN-5		
2.2.3 Characterization of Hazards from Transportation Accidents, Including Toxic Chemicals or Gas Releases/Explosions			
2.2.3.1 Hazards from Air Transportation Accidents			
Small aircraft	Screened out	QL-1: Equal or lesser damage than similar design basis event	The small aircraft crash is screened out as the BWRX-300 is designed to withstand site-specific automobile tornado missiles, per Subsection 2.6.6.
Large military aircraft	Screened out	QL-3: Cannot occur at or close enough to the site to affect BWRX-300	Large bombers, large cargo planes, fuel tankers, or heavily armed jet fighters do not fly in the vicinity of the Bowmanville airspace
Large civil aircraft	Screened out	QN-5: Frequency of <1.0E-7/yr	NOTE: Malevolent large aircraft crash is analyzed in the Security Annex.
2.2.3.2 Characterization of Hazards from Rail Transportation Accidents			
Release of toxic gases	Screened in as DEC	Hazard frequency is estimated at 1.9E-06 occ./yr. Thus, this hazard is a Beyond Design Basis Accident (BDBA) DEC, as documented in NK054-REP-01210-00150 (Reference 2.2-11)	
Explosions	Screened in as DEC	Hazard frequency is estimated at 9.0E-07 acc./yr Thus, this hazard is a BDBA DEC, as documented in NK054-REP-01210-00149 (Reference 2.2-12)	
2.2.3.3 Characterization of Hazards from Road Transportation and Traffic Accidents			
Release of toxic or asphyxiant material	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant	The location of the Darlington Nuclear site is about 1.0 km away from Highway 401.
2.2.3.4 Characterization of Marine Transportation			
Chemical Leak	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant QL6: Does not cause an initiating event	Commercial shipping is approximately 27 km away for the DNNP. The consequence of a chemical leak from a tanker or a cargo ship, would be mostly an environmental hazard, and would not have an impact on safe operation of the station.
Release of toxic gases	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant	The location of the DNNP is about 27 km away from the general tanker or cargo ship commercial routes in Lake Ontario.
Explosion	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant	The location of the DNNP is about 27 km away from the general tanker or cargo ship commercial routes in Lake Ontario.

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Physical Damage	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant QL1: Bounded by the impact of damage caused by frazil ice described in Subsection 2.2.7.2	Hazards from accidents involving recreational boats or vessels pose no significant threat to the BWRX-300 safe operation, even if the accidents occur near the lake water intake structure. Also, a restricted zone is established around the BWRX-300 offshore structures.
2.2.4 Characterization of Stationary Non-Nuclear Accidents			
2.2.4 Fire – Natural Gas Pipelines	Screened out	QL6: Does not cause an initiating event or relevant safety function	There are no substantial pipelines carrying large quantities of natural gas, close enough to the site.
2.2.4.1 Release of toxic gases or chemical from commercial outlets in the area	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant QL5: The event is slow to develop so there is sufficient time to eliminate the source of adequately respond	There are no industrial toxic gas or chemical storage tanks or pipelines carrying significant quantities of natural gas close enough to the site. Assumed St. Marys toxic release is not close enough to the site to affect the plant
2.2.4.2 Explosion – Shock Waves	Screened out	QL3: Cannot occur on or close enough to the site to affect the plant	Distances between DNNP and: <ul style="list-style-type: none">• Cigas Propane tanks are about 3.6 km far from the DNNP site• St. Marys diesel fuel tanks is greater than 700 m from the Power Block of multi-unit layout (Reference 2.2-16).
2.2.4.2 Explosion - Missiles Hydrogen used for Tritium Removal Facility	Screened out	QL3: Large Missiles - Cannot occur on or close enough to the site to affect the plant QL4: Small Missiles - Bounded by design basis tornado in Subsection 2.6.6	The Tritium Removal Facility is located approximately 1.0 km west of the DNGS vacuum building.
2.2.5 Characterization of Stationary Nuclear Accidents Hazards			
2.2.5.1 Cameco's Port Hope Uranium Conversion Facility	Screened out	The facility is located on the north shore of Lake Ontario, approximately 40 km east of Darlington Nuclear site. The Cameco plant is a chemical processing facility with negligible radioactive releases.	
2.2.5.2 PNGS	Screened out	Any hazard from PNGS irradiated fuel still within an irradiated fuel bay or a dry storage facility is bounded by the much closer event from DNGS. Based on (Reference 2.2-5), PNGS radioactive release event is characterized as a slow developing event, allowing sufficient time for operators to take appropriate actions (if warranted), and can therefore be screened out.	
2.2.5.3 DNGS – Exclusion Zone	Screened in	The DNNP site is partly within the exclusion zone of DNGS.	
2.2.5.4 Characterization of Other Radiological Hazards from DNGS			
2.2.5.4.1 DNGS – Tritium Removal Facility – Tritium Release	Screened out	Evaluations in (Reference 2.2-1) and (Reference 2.2-5) determined that regulatory dose limits at the site boundary apply to all these nuclear events with negligible impact to DNNP.	

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2.2.5.4.2 DNGS – Irradiated Wet Fuel Storage Facility	Screened out		
2.2.5.4.3 DNGS – Irradiated Dry Fuel Storage Facility	Screened out		
2.2.5.4.4 DNGS – Radioactive Waste Storage	Screened out		
2.2.6 Characterization Industrial Hazards (St. Marys)			
St. Marys Cement Plant – Uncontrolled blasts	Screened in	St. Marys Cement commits to carry out blasts with a maximum allowable horizontal, vertical, longitudinal, and radial velocities of less than 3 mm/s measured at the Darlington Nuclear site property boundary with St. Marys.	
2.2.7 Characterization of Biological, Animal and Frazil Ice Hazards			
2.2.7.1 Water-based Biological	Screened out	QL4: Bounded by the impact of damage caused by frazil ice described in Subsection 2.2.7.2	Hazards associated with blockage of intake cooling water resulting in the loss of heat sink
2.2.7.1 Airborne birds or insects	Screened out	QL-1: Equal or lesser damage than similar design basis event	This event is equivalent to outside air damper isolation during off-normal conditions
2.2.7.2 Frazil Ice	Screened in	Frazil ice is considered a potential hazard for causing water intake blockage to DNNP.	
2.2.8 Characterization of Ice Storm Hazard			
Ice Storm	Screened out	QL-1: Equal or lesser damage than similar design basis event	For the DNNP BWRX-300, the loss of the switchyard is part of the Loss-of-Preferred Power (LOPP), an Anticipated Operational Occurrence, which is the Pressure Increase Group and is designated as a BL-AOO event
2.2.9 Characterization of Electromagnetic Interference Hazard			
Electromagnetic Interference	Screened in	Since electromagnetic interference sources (e.g., high-voltage transmission lines and communication towers) are continuously present, the risk of electromagnetic interference at the site must be addressed in the design basis of the BWRX-300	
2.2.10 Characterization of On-site Methane Hazard			
During construction	Screened in	Methane gas is harmful to the health of humans and is combustible. Methane gas must be monitored during excavation, especially for the RB, since the methane is expected to dissipate quicker than what was observed in the boreholes due to the significantly larger air space.	
Post construction	Screened in	Methane in bedrock during operation is added as a hazard to be considered during design	

2.2.2 External Hazards Evaluation Methodology

The 2019 Hazards Screening Analysis reported in the 2019 NK38-REP-03611-10043 (Reference 2.2-5) provides a comprehensive assessment of the hazards associated with the DNGS site. Given that the DNNP site is within the Darlington Nuclear site (refer to Chapter 1, Figure A1.1-2) and in geographic proximity with the DNGS site, this analysis is deemed applicable to support and inform the evaluation of the external hazards listed in Subsection 2.2.1 for the DNNP site. In addition, since the DNGS external hazard screening methodology NK38-REP-03611-10043 (Reference 2.2-5) is aligned with the 2022 BWRX-300 DNNP Hazard Analysis Methodology NK054-REP-01210-00144 (Reference 2.2-10), the results of the 2019 DNGS analysis in NK38-REP-03611-10043 (Reference 2.2-5) are used to supplement and validate the DNNP site-specific external hazards evaluation reported in the 2022 DNNP NK054-REP-01210-00144 (Reference 2.2-10).

In the 2019 Site Preparation Licence Renewal Activity Report NK054-REP-01210-00108 (Reference 2.2-3), detailed DSA and PSA are performed during the BWRX-300 design phase. The DSA and PSA updates are performed in compliance with CNSC REGDOC-2.4.1 (Reference 2.2-14) and REGDOC-2.4.2 (Reference 2.2-15), respectively, and are tracked under the 2021 DNNP Commitment D-C-3 NK054-REP-01210-00078 (Reference 2.2-8). With respect to external hazards, DNNP Commitment D-C-3 also requires “the design of the new plant must demonstrate that it can mitigate the identified hazards to ensure that the required safety goals are met.”

The screening methodology and criteria used to assess hazards are described is found in Section 1.0 of the DNGS 2019 NK38-REP-03611-10043 (Reference 2.2-5). The screening technique involved a systematic approach starting with a qualitative assessment of the impacts of hazards on the safe operation of the station, followed by a quantitative screening of hazards not being screened out qualitatively. The methodology follows OPG’s PSA guides for screening of internal and external hazards.

The 2022 BWRX-300 DNNP Hazard Analysis Methodology NK054-REP-01210-00144 (Reference 2.2-10) builds on the 2019 Darlington screening technique NK38-REP-03611-10043 (Reference 2.2-5) and devises comparable criteria for the BWRX-300 facility. The developed qualitative and quantitative screening criteria are applicable to screening internal and external hazards, as listed in Appendix B of the 2022 NK054-REP-01210-00144 (Reference 2.2-10).

The following criteria are used for qualitative screening of hazards in the 2019 NK38-REP-03611-10043 (Reference 2.2-5):

- QL-1: The event is of equal or lesser damage potential than similar events for which the plant has been designed.
- QL-2: The event has a significantly lower reactor sources likelihood than another event that has been screened out, and yet the event could not result in worse consequences than the other event.
- QL-3: The event cannot occur at the site or close enough to the site to affect the plant.
- QL-4: The event is included in the definition of another event.
- QL-5: The event is slow in developing such that it can be demonstrated that there is sufficient time to eliminate the source of the threat or provide an adequate response.
- QL-6: The event does not cause an initiating event (including the need for a controlled shutdown) as well as safety system function losses needed for the event.
- QL-7: The consequences to the plant do not require the actuation of front-line systems.

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NOTE: QL-1 to QL-5 apply to both the reactor and non-reactor sources. QL-6 and QL-7 apply only to reactor sources and not to the non-reactor sources.

The following criteria are used for quantitative screening in the 2019 NK38-REP-03611-10043 (Reference 2.2-5).

- QN-1: Severe Core Damage Frequency $< 1.0\text{E-}6/\text{yr}$. Applies only to reactor sources and not to non-reactor sources.
- QN-2: Design Basis Hazard Frequency, $< 1.0\text{E-}5/\text{yr}$ and Conditional Core Damage Probability < 0.1 . Applies to reactor sources only and not to non-reactor sources.
- QN-3: Severe Core Damage Frequency $< 10^{-7}/\text{yr}$. Applies to the reactor sources only. An equivalent QN for non-reactor sources of Low Release Frequency (LRF) $< 1.0\text{E-}7/\text{yr}$ is considered.
- QN-4: Design Basis Hazard Frequency, $< 1.0\text{E-}6/\text{yr}$ and Conditional Core Damage Probability < 0.1 . Applies to reactor sources only. An equivalent QN for non-reactor sources is considered as follows: Design Basis Hazard Frequency, $< 1.0\text{E-}6/\text{yr}$ and conditional large release probability (CLRP) < 0.1 .
- QN-5: Initiating Event or Hazard Frequency may be screened out if it can be shown that their frequency is $< 1.0\text{E-}7/\text{yr}$. Applies to both reactor and non-reactor sources.

The application of this methodology results in hazards being “screened out” or “screened in.” “Screened out” implies that the hazard does not pose any safety concerns. “Screened in” implies further assessment is required to address the hazards. Hazards which are neither qualitatively nor quantitatively screened out, are addressed during detailed Probabilistic Safety Assessments (for example, seismic, high winds).

2.2.3 Characterization of Hazards from Transportation Accidents, Including Toxic Chemicals or Gas Releases/Explosions

Evaluations of hazards from transportation accidents are detailed as follows:

- By air - Subsection 2.2.3.1
- By train - Subsection 2.2.3.2
- By road - Subsection 2.2.3.3
- By marine – Subsection 2.2.3.4

Previous assessment results for DNNP hazards associated with transportation events are provided in the 2009 NK054-REP-01210-00008 (Reference 2.2-1) and the 2009 NK054-REP-01210-00019 (Reference 2.2-2). The evaluations presented in Subsection 2.2.3 address the specific issues identified by the CNSC in Subsections 4.6.1, 4.6.2 and 4.6.3 of the 2020 Renewal Application for DNNP Site Preparation Licence NK054-CORR-00531-10533 (Reference 2.2-4).

Aircraft crashes and ship accidents were evaluated for the DNNP site in Section 4.3 and 4.4 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1). The evaluation did not consider the impact from toxic chemicals or gas releases/explosions specific to these accidents. However, the impact from toxic chemicals or gas releases/explosions from transportation accidents were implicitly assessed in Section 4.6 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1).

Further, Section 4.6 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) evaluated the risks associated with hazardous fluids, including toxic clouds from the release of toxic gases, deflagrations (explosions) from the release of liquified petroleum gases and flammable pressure

liquified gases. The evaluation determined toxic gas clouds reaching the DNNP site at high enough concentrations have the potential to impact the Main Control Room (MCR) and Secondary Control Room (SCR) habitability of the proposed plant (that is, the BWRX-300 nuclear facility). Refer to Chapter 6, Section 6.4 for further details on habitability of the MCR and SCR.

With respect to explosions, the evaluation in the 2009 NK054-REP-01210-00008 (Reference 2.2-1) identified potential damage to buildings from missiles resulting from Boiling Liquid Expanding Vapour Explosion (i.e., tanks containing liquified petroleum); when travelling at high velocity, these missiles can damage outdoor and indoor equipment. The evaluation determined that the overpressure effects due to explosion on the building must be mitigated. Mitigation may require the use of an appropriate physical barrier or the physical separation of important safety equipment/systems. The evaluation stated that requirements for this hazard is to be considered during the detailed design phase of the project (that is, BWRX-300). The 2019 DNGS Hazards Screening Analysis NK38-REP-03611-10043 (Reference 2.2-5) also assessed the release of toxic chemicals and gas/release explosions from transportation accidents. The data used for DNGS hazards analysis supplement the DNNP site-specific data that are employed in the design and safety analysis stage of the DNNP BWRX-300, as applicable.

For additional information specific to toxic gas and chemical hazards, refer to Subsection 2.2.3.2 for rail transportation accident hazards, Subsection 2.2.4.1 for release from stationary hazards, and Subsection 2.4.1 for on-site hazards.

2.2.3.1 Characterization of Hazards from Air Transportation Accidents

Two types of aircraft are examined: small and large (both civil and military).

1. The small aircraft crash is screened out qualitatively as not having an impact on the safe operation of the facility, based on the screening criterion QL1 that the event is of equal or lesser damage potential than similar events for which the plant is designed. Per Section 3.1 of the 2019 Darlington hazard screening analysis (Reference 2.2-5), small aircraft impact is bounded by tornado missiles. The small aircraft crash is therefore screened out as the BWRX-300 will be designed to withstand automobile tornadoes missiles (refer to Subsection 2.6.6).
2. Large aircraft (military) aviation accidents are not a concern for the Darlington Nuclear site, as there are no large bombers, large cargo planes or fuel tankers, or heavily armed jet fighters flying in the vicinity of the Bowmanville airspace, per the 2020 NK054-CORR-00531-10533 (Reference 2.2-4).
3. Large aircraft (civil) accidents are screened out under screening criterion QN5 (refer to Subsection 2.2.2) based on a hazard frequency of $<1.0E-7/\text{yr}$.

2.2.3.2 Characterization of Hazards from Rail Transportation Accidents

As described in Subsection 2.1.5, two railway lines run within the 10 km study area surrounding the Darlington Nuclear site. Of particular relevance is the Canadian National Railway line which bisects the Darlington Nuclear site and passes approximately 600 m north of the DNNP site. This railway line has potential hazards associated with assumed derailment accidents involving one or more cargo cars.

Rail transportation accidents are assessed in the 2019 DNGS Hazards Screening Analysis NK38-REP-03611-10043 (Reference 2.2-5), the 2022 DNNP Rail Transportation – Toxic Gas/Chemical Release Hazard Assessment NK054-REP-01210-00150 (Reference 2.2-11), and the 2022 DNNP Rail Transportation – Explosion Hazard Assessment NK054-REP-01210-00149 (Reference 2.2-12). The objective is to address hazards associated with train derailment and crash, including

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cold or hot toxic gas releases, as well as Vapour Cloud Explosions, Boiling Liquid Expanding Vapour Explosion, and other types of explosions.

The assessments considered the two rail lines running “east-west” directly north of Darlington Nuclear site. Of particular interest is the Canadian National Railway Toronto to Montreal main line which passes through the OPG DNGS and DNNP sites, as shown in Figure 2.1.1-2.

One of the hazards analyzed in the 2022 NK054-REP-01210-00150 (Reference 2.2-11) is the possibility of a large toxic gas/chemical release. A consequential harm from this hazard could be a toxic gas/chemical release that would be airborne toward the DNNP site with the capacity for widespread and distant impact. Another hazard is the potential of large explosion, analyzed in NK054-REP-01210-00149 (Reference 2.2-12), involving explosive commodities being transported by the railway line, occurring in the vicinity of DNNP BWRX-300 structures and components. The following toxic gas release and explosion scenarios are assessed in the 2019 NK38-REP-03611-10043 (Reference 2.2-5), the 2022 NK054-REP-01210-00150 (Reference 2.2-11), and the 2022 NK054-REP-01210-00149 (Reference 2.2-12), for applicability to DNNP:

1. Cold Toxic Gases Release: Release and dispersion of airborne toxic chemicals or asphyxiants toward the BWRX-300 HVAC intakes that could expose the station staff to toxic chemicals and result in challenging the habitability of work areas.
2. Hot Toxic Gas Release: Similar to cold toxic gas releases, if the train derailment accident involves fire, it could result in hot toxic gas releases. Combustible chemicals could result in releasing an intense heat, causing secondary combustion of other materials (e.g., insulations, containers and covers), and such releases usually involve other chemicals that can have a wide range of toxicities. Heavy hydrocarbons produce a significant amount of carbon dioxide, carbon monoxide and soot when they catch fire. Some chemicals may produce toxic byproducts while burning, such as hydrazine (combustion byproducts include nitrogen dioxide, which is highly toxic).
3. Hydrocarbon Explosions: Release of light hydrocarbons with high vapour pressures (flammable), when transported under high pressure (e.g., liquefied petroleum gas), can produce two types of explosions:
 - a. Boiling Liquid Expanding Vapour Explosion: Boiling Liquid Expanding Vapour Explosions could generate missiles, fireballs, and blast waves. Missiles could travel hundreds of meters from the source. Blast waves from Boiling Liquid Expanding Vapour Explosions are normally localized.
 - b. Vapour Cloud Explosion: With Vapour Cloud Explosions, vapour cloud ignition is delayed after the cloud has dispersed somewhat and mixed with air. Vapour Cloud Explosions produce blast waves that could damage buildings and equipment.
 - c. Confined Explosions: A flammable fluid can produce a confined explosion if it becomes airborne, mixes with air, and is ignited in a confined space. This would produce a so-called Confined Explosion. Such an explosion could arise in a building, a room, or the vapour space of a storage tank. Blast waves from confined explosions are localized.

The hazard from the release of toxic gases resulting from Canadian National Railway assumed transportation accidents close to the DNNP site have an estimated frequency of 1.9E-06 occ./yr, per the 2022 NK054-REP-01210-00150 (Reference 2.2-11). Thus, it is screened out from design basis input since it is assessed as a Beyond Design Accident (BDBA) DEC, per REGDOC-2.4.1 (Reference 2.2-14).

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Similarly, the explosion hazard from a Canadian National Railway derailment accident near the DNNP site has an estimated frequency of $9.0\text{E-}07$ occ./yr, per the 2022 NK054-REP-01210-00149 (Reference 2.2-12). Consequently, it is screened out from design basis input based on the assessment that it is a BDBA DEC, per REGDOC-2.4.1 (Reference 2.2-14).

2.2.3.3 Characterization of Hazards from Road Transportation and Traffic Accidents

Road transportation and traffic accidents are assessed in 2019, and results for DNGS are reported in Subsection 3.2.3 of per the 2019 NK38-REP-03611-10043 (Reference 2.2-5). The assessment considered the location of the Darlington Nuclear site, also encompassing the DNNP site, which is about 1.0 km away from the Macdonald–Cartier Freeway (also known as Highway 401) and one of the busiest highways in Canada.

The event scenario considered involves two tractor trailers crash (or rollover), such that multiple containers are damaged, consequential toxic or asphyxiant materials are released into the atmosphere, and the wind (2 m/s) disperses the airborne chemicals toward the BWRX-300 HVAC systems intakes (refer to Chapter 9A, Section 9A.5 for information on BWRX-300 HVAC systems).

Highway 401 is about 1.0 km north of the DNNP site. The impact of two tractor trailer crash is therefore screened out based on distance. Explosion or release of toxic/asphyxiant materials from the colliding two tractor trailers depends on the size of insuring breaks and the consequential amount of material released (via leaking or 100% break), wind direction and speed, and the degree of dilution due to dispersion. This scenario is therefore screened out based on distance and low impact without performing confirmatory assessment.

2.2.3.4 Characterization Hazards from Marine Transportation

The cargo vessels move along shipping lanes which are designated by the Ministry of Transport, and the nearest approach is about 27 km from the Darlington Nuclear site, per Section 3.3 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5). Therefore, scenarios involving tankers or cargo ships are, in general, screened out based on distance, per screening criterion [QL3]

The consequences of a chemical leak from a tanker or a cargo ship would be mostly an environmental hazard. Depending on the exact nature, severity, and progression time of the accident as well as the consequential amount of leaked material, lake current and degree of dilution, such scenarios would have negligible impact on the quality, or the quantity of the cooling water supplied to the BWRX-300. A tanker or cargo ship accident resulting in chemical leak is screened out based on screening criterion [QL6].

The hazard of an explosion onboard a cargo ship and subsequent release of toxic gases is screened out based on screening criterion [QL3], that is, the event cannot occur at the site or close enough to the site to affect the plant.

As described in the 2019 NK38-REP-03611-10043 (Reference 2.2-5), a large number of small or large recreational boats or vessels travel across Lake Ontario, Winter conditions limit this traffic to about 8 months of the year. Hazards from accidents involving such recreational boats or vessels pose no significant threat to the BWRX-300 safe operation, even if the accidents occur near the lake water intake structure. St. Marys Cement Company Limited owns a pier that is about 700 m to the east of the DNNP site. Bulk carriers may load cement or unload gypsum or coal at this dock. Also, a restricted zone is established around the BWRX-300 offshore structures. The consequence of such accidents is bounded by a frazil ice hazard, and therefore, associated hazards are screened out based on criterion [QL1]

2.2.4 Characterization of Stationary Non-Nuclear Accidents Hazards

The evaluation of hazards associated with stationary non-nuclear accidents is based on the results of the assessment reported in the 2019 NK38-REP-03611-10043 (Reference 2.2-5) for DNGS. Since DNNP is also located within the Darlington Nuclear site boundary, the results of the DNGS assessment are relevant to DNNP.

Event scenarios that can result in an accidental fire, explosion, or a release of hazardous material from stationary sources have been assessed in Section 3.5 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5). The locations of the initiating mechanism for these sources are constrained to tank farms and forest fires.

The main stationary sources of external hazardous material near the Darlington Nuclear site are:

1. Regional Water Treatment Plants which generally have a large inventory of Chlorine for treatment of water.
2. Cigas Propane, which is located 3.6 km away from Darlington Nuclear site, where a large inventory of propane gas is stored.
3. St. Marys Cement plant located about 1.5 km east of the DNGS site and approximately 700 m from the DNNP site. The plant stores large inventories of a variety of hazardous chemicals on-site. The main toxic and hazardous materials are as follows (Reference 2.2-16):
 - Aqueous (19%) ammonia (NH_4OH) tank with capacity of up to 38 000 L
 - Diesel fuel storage tanks with capacity of up to 50 000 L used for heating and fueling mobile equipment.
4. The DNGS Tritium Removal Facility where chemicals and fuel stored could potentially pose hazards to DNNP BWRX-300 resulting from the release of toxic chemicals, hydrocarbon explosions (Boiling Liquid Expanding Vapour Explosions and Vapour Cloud Explosions), or confined explosions (refer to Subsection 2.2.5.4 for additional information on DNGS potential hazards).

Substantial pipelines carrying large quantities of natural gas do not run close enough to the Darlington Nuclear site. Therefore, the risk of fire due to pipelines ruptures poses a negligible incremental risk to the DNNP site and, thus, it was screened out based on screening criterion [QL-6] (Subsection 2.2.2).

2.2.4.1 Characterization of Toxic Chemicals Releases from Stationary Hazards

As described in the 2019 NK38-REP-03611-10043 (Reference 2.2-5), the event scenario assessed involves a local accident in one of the regional water treatment plants (for example, Courtice WPCP) or in the St. Marys Cement plant, resulting in the release of chlorine gas (Cl_2) or gas/aqueous ammonia ($\text{NH}_3/\text{NH}_4\text{OH}$), respectively. Combustion of NH_3 in the air could result in NO or NO_2 , in the presence of appropriate catalysts. Nitrogen dioxide is toxic by inhalation, but it is easily detectable by smell at low concentrations. The combustion of ammonia in air is difficult in the absence of a catalyst, as the temperature of the flame is usually lower than the ignition temperature of the ammonia-air mixture.

The accident is assumed to include multiple containers. As such, the airborne toxic material, chlorine, or ammonia, released into the atmosphere could disperse toward the BWRX-300 HVAC intakes. Depending on the size and nature (i.e., severity and time frame) of the release, wind direction and wind speed, the concentration of toxic chemicals varies.

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For the chlorine hazard, the nearest water treatment plant, the Courtice WPCP, is approximately 5 km west of the BWRX-300 HVAC intakes. Thus, this hazard is screened out under screening criterion [QL-3] (Subsection 2.2.2) as the event cannot occur at the site or close enough to the site to affect the plant.

With respect to the ammonia hazard associated with accidents at the St. Marys Cement plant which is located approximately 700 m east of DNNP site boundary (Reference 2.2-16) and considering the total low-level of inventory of ammonia at the St. Marys plant, the toxic release is screened out from further assessment under screening criterion [QL-1] (Subsection 2.2.2).

2.2.4.2 Characterization of Explosions from Stationary Sources

The event scenario involves the explosion of multiple propane tanks at the Cigas Propane storage facility, or the explosion of multiple diesel fuel tanks located at the St. Marys Cement plant as per Subsection 3.5.2 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5). As multiple tanks are damaged, there are missiles potentially generated by the explosions, as well as shockwaves, which can damage SSCs several hundred meters away.

The screening distances for different types of explosions, per the 2019 NK38-REP-03611-10043 (Reference 2.2-5) are estimated at 1600 m for Boiling Liquid Expanding Vapour Explosion, 700 m for explosions equivalent to 61.5 Mg trinitrotoluene, and 460 m for Vapour Cloud Explosion. For the DNNP, considering the distances of the hazardous sites (3,600 m for Cigas Propane, and greater than 700 m for St. Marys Cement), both scenarios for Boiling Liquid Expanding Vapour Explosion due to propane tanks explosions at Cigas Propane, and explosions due to diesel fuel tanks at St. Marys Cement were screened out, based on distance screening criterion [QL3]. (NOTE: The St. Marys Cement does not store large quantities of pressurized light hydrocarbons (unlike that in Cigas Propane).)

An assessment of missiles generated from an explosion associated with hydrogen used in the Tritium Removal Facility was performed in 2019 for DNGS in NK38-REP-03611-10043 (Reference 2.2-5). The Tritium Removal Facility is located directly west of the DNGS vacuum building. The assessment determined that missiles generated by an explosion in the Tritium Removal Facility are bounded by missiles generated by a design basis tornado, for which DNGS is protected.

The DNNP facility is approximately 1.0 km away from the Tritium Removal Facility, and the DNGS and its vacuum building provide an obstruction between the Tritium Removal Facility and the DNNP BWRX-300 facility. As such, this hazard is screened out based on [QL-3] for large missiles since the event cannot occur on or close enough to the DNNP site to affect the BWRX-300 facility. Small missiles generated by an explosion at the Tritium Removal Facility can also be screened out for the DNNP BWRX-300 design, using screening criterion [QL-4], since such small missiles are bounded by the design basis tornado automobile missiles (refer to Subsection 2.6.6).

2.2.5 Characterization of Stationary Nuclear Accidents Hazards

Stationary nuclear accident sources within the vicinity of DNNP that pose potential hazards from nuclear accidents are:

1. Cameco's Port Hope Uranium Conversion Facility – located about 40 km east of the Darlington Nuclear site where the DNNP is located
2. PNGS – located about 25 km west of the Darlington Nuclear site where the BWRX-300 is to be built
3. DNGS – located within one kilometer west of the BWRX-300 footprint

The Cameco facility and PNGS were assessed in the 2019 Hazard Screening Analysis NK38-REP-03611-10043 (Reference 2.2-5) performed for DNGS.

2.2.5.1 Evaluation of Cameco's Port Hope Uranium Conversion Facility Hazard

Cameco's Port Hope uranium conversion facility is a nuclear substance processing facility licensed to process uranium trioxide (UO_3) into both uranium dioxide (UO_2) and uranium hexafluoride (UF_6). Natural UO_2 is used to manufacture CANDU fuel for nuclear power reactors in Canada, while UF_6 is exported to companies in other countries for enrichment and fabrication into fuel for international nuclear power reactors. The facility is located on the north shore of Lake Ontario, approximately 40 km east of Darlington Nuclear site. The Cameco plant is a chemical processing facility with negligible radioactive releases, and therefore it is not included in the screening analysis for DNGS. Based on the DNNP proximity to DNGS, the screening results for DNGS are directly applicable to DNNP and hence screened out from further evaluation both deterministically and probabilistically.

2.2.5.2 Characterization of Pickering Nuclear Generating Station Hazards

PNGS is located on the shores of Lake Ontario, approximately 25 km west of Darlington Nuclear site. The PNGS is an eight-unit station with six operating CANDU reactors with a total output of 3100 MWe, and two units in safe storage. OPG is conducting a re-assessment, per the 2022 P-CORR-00531-23042 (Reference 2.2-13), involving a comprehensive technical examination of the potential for refurbishing Units 5, 6, 7 and 8 of PNGS. The results including recommendations of such an assessment are to be reported in 2023.

As described in the 2019 NK38-REP-03611-10043 (Reference 2.2-5), the accidental release of radioactive materials at PNGS can be screened out for DNGS given it is a slow developing event, and there are mitigating features as well as enough time for operators to take proper actions. As the DNNP is farther from PNGS and similar mitigation measures, if warranted, are implemented, the radiological hazards associated with such events are also screened out for DNNP. Any hazard from PNGS used CANDU fuel still within an irradiated fuel bay or a dry storage facility is bounded by the much closer event from DNGS discussed in Subsection 2.2.5.4.2.

2.2.5.3 Characterization of Darlington Nuclear Generating Station Hazard

The BWRX-300 Unit 1 footprint resides partly within the DNGS exclusion zone (nominally 914 m), that is within DNGS controlled area, per Subsection 5.10 of the 2020 NK054-CORR-00531-10533 (Reference 2.2-4). The closeness of DNNP to DNGS means that in the event of a nuclear accident within DNGS the ability to maintain safe operation of DNNP can potentially be affected.

2.2.5.4 Characterization of Other Radiological Hazards from DNGS

Potential radiological hazards in the area that could affect the safe operation of the new nuclear plant were evaluated in Section 4.8 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1). Nuclear events at the DNGS considered in this assessment were as follows:

- Tritium Removal Facility accidents leading to release of tritium
- In-plant fire near a storage area of active liquid waste
- Used irradiated fuel accident
- Design basis reactor accidents
- Beyond design basis reactor accidents which include severe accidents that have the potential for a significant off-site release of radioactive materials

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The assessment determined these events do not pose a concern to equipment but would likely impact the operating staff of the proposed plant (that is, the BWRX-300 facility). Four specific events as listed below are discussed in more details:

- Tritium Removal Facility – Subsection 2.2.5.4.1
- Irradiated Fuel Storage Facility – Subsection 2.2.5.4.2
- Used Fuel Dry Storage – Subsection 2.2.5.4.3
- Radioactive Waste Storage -Subsection 2.2.5.4.4

In October 2021, DNGS Power Reactor Operating Licence PROL 13.02/2015 was amended to authorize unit 2 to produce molybdenum-99, an isotope used in the medical industry for diagnostics. The CNSC decision concludes that the licensed activities will have a negligible effect on severe core damage frequency and large release frequency (Reference 2.2-17). In the future, DNGS may pursue production of other isotopes and/or molybdenum-99 in other units.

2.2.5.4.1 Characterization of Tritium Removal Facility Hazard

The Tritium Removal Facility is located within the boundary of the DNGS site, to the west side of the DNGS vacuum building. Release of tritium from an accident at the Tritium Removal Facility was evaluated in Section 4.8 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1). The assessment concluded that accidents leading to a tritium release do not pose concern to equipment but have the potential to impact operators. (Refer to Chapter 6, Subsections 6.4.1.1 and 6.4.1.2 for information on the BWRX-300 MCR and SCR habitability provisions, respectively.)

Helium-3 (He-3) is also extracted from tritium storage containers at the Tritium Removal Facility for medical and commercial uses. He-3 is a non-radioactive, inert, and non-toxic gas and therefore accidental release does not contribute any additional risk.

2.2.5.4.2 Characterization of Irradiated Fuel Storage Facilities Hazards

Following its useful life in the DNGS reactors, used CANDU fuel bundles are discharged from the fueling machine heads and initially stored underwater in modules in irradiated fuel bays at the West and East Fueling Facility Auxiliary Areas, located inside the DNGS protected area, adjacent to Unit 1 and Unit 4, respectively. Then the used fuel modules are transferred to and placed into seismic stacking frames inside the main irradiated fuel storage bays where the water in the bays removes heat produced by the decaying used fuel and provides shielding for workers. After a specified number of years, based on bays capacity and operational needs, the used fuel is transferred to an on-site irradiated fuel dry storage facility for short-term storage, and ultimately to an off-site long-term dry storage facility when it becomes available in the future. The hazards posed by both the irradiated fuel bays and the on-site irradiated fuel dry storage facility are analyzed in NK38-REP-03611-10043 (Reference 2.2-5).

Radiological releases from used fuel accidents were also evaluated in Section 4.8 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1). It was determined that used fuel accidents posed no concern for DBAs.

Analysis of human-induced hazards and natural hazards for the DNGS irradiated fuel bays was performed and documented in Section 5 and Section 6, respectively of the 2019 NK38-REP-03611-10043 (Reference 2.2-5). All human-induced hazards analyzed have been screened out (Table 5-1 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5)), which is applicable to DNNP as well. For natural hazards, Table 6-1 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5) summarizes hazards which are not screened out. Irradiated fuel bay accident analysis is documented in Subsection 3.6.4 of the 2017 NK38-SR-03500-10002 (Reference 2.2-9).

2.2.5.4.3 Characterization of Used Fuel Dry Storage Hazard

Analysis of human-induced hazards and natural hazards for irradiated CANDU fuel dry storage facility was performed and documented in Section 7 and Section 8, respectively, of the 2019 NK38-REP-03611-10043 (Reference 2.2-5). All human-induced and natural hazards analyzed have been screened out as not having a safety impact on DNGS. The results are directly applicable to DNNP BWRX-300 and have been screened out, as per Table 5-1 of the 2019 NK38-REP-03611-10043 (Reference 2.2-5).

2.2.5.4.4 Characterization of Radioactive Waste Storage Hazard

The scenario analyzed in Section 4.8 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) for radioactive waste storage accidents is an in-plant fire near a storage area of active liquid waste. This event poses no concern for DBAs.

2.2.6 Characterization Industrial Hazards

The primary industrial hazard of concern is uncontrolled underground blasting associated with the St. Marys Cement plant.

This hazard was assessed in Section 3.6 of the 2019 DNGS Hazard Screening Assessment NK38-REP-03611-10043 (Reference 2.2-5). The results of the assessment indicated blasting at St. Marys quarry leads to shock waves in the ground travelling to the Darlington Nuclear site.

Vibration monitors on the Darlington Nuclear site at the St. Marys' property boundary are designed to record the amplitude and frequencies of such shock waves, originating from the St. Marys Cement plant. St. Marys Cement commits to not carry out blasts that may exceed the maximum allowable horizontal, vertical, longitudinal, and radial velocities of 3 mm/s measured at the Darlington Nuclear site property boundary with St. Marys.

This agreement was originally put in place to avoid turbine trips at DNGS. Since DNNP is in geographic proximity to DNGS and is closer to St. Marys Cement plant than DNGS, this hazard is applicable to the BWRX-300 facility.

The agreement noting 3 mm/s is between OPG and St. Marys and is therefore applicable to DNNP.

The maximum allowable horizontal, vertical, longitudinal, and radial velocities of 3 mm/s measured at OPG's Darlington Nuclear site property boundary is screened in and shall be considered in the design of the BWRX-300 facility.

2.2.7 Characterization of Biological, Animal, and Frazil Ice Hazards

Lake Ontario is the reservoir of cooling water for the DNNP BWRX-300 facility. Fouling of the intake structures and components from growth of biological species (e.g., algae, mussels, or clams) and the presence of animals (e.g., birds, fishes, or other wildlife) impede the availability of water for heat sink purposes. Also, the formation of frazil ice at the intake can restrict or block supply to the Circulating Water System (CWS) (refer to Subsection 2.5.2). Both potential hazards are evaluated in the following two subsections.

2.2.7.1 Characterization of Biological and Animal Hazard

Biological Hazards

A variety of sources of organisms or organic material that could contribute to biofouling associated with cooling water systems originate from the pathway represented by Lake Ontario, thus restricting or blocking water supply to the BWRX-300 facility.

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The impact of biological and animal hazards on the safe operation of DNNP was considered and documented in the 2009 NK054-REP-01210-00019 (Reference 2.2-2), the 2020 NK054-CORR-00531-10533 (Reference 2.2-4), the 2009 NK054-REP-01210-00018 (Reference 2.2-7), and the 2021 NK054-REP-01210-00078 (Reference 2.2-8).

Section 2 of the 2009 NK054-REP-01210-00018 (Reference 2.2-7) assessed the hazards associated with blockage of cooling water intake. The primary species that can contribute to biofouling have been identified and assessed. Biofouling was identified as a potential hazard that can result in loss of cooling and fouling of cooling equipment, such as lines and heat exchangers.

Section 3.5 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) considered the adequacy of water supply affected by biofouling, where several species were assessed.

Further discussion on the prevention of biofouling for the cooling water intake is provided in Subsection 2.5.2.2.

Animal Hazards

Airborne animal hazards (e.g., birds or insects) have the ability to block the screens of the MCR air ventilation intakes. This event is equivalent to outside air damper isolation during off-normal conditions, as described in Chapter 9A, Subsection 9A.5.2.1.4. The airborne animal hazard is therefore screened out using screening criterion [QL6].

2.2.7.2 Characterization of Frazil Ice Hazard

Section 3.5 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) states frazil ice forms in turbulent, supercooled water (water temperatures of -0.01°C to -0.05°C). To generate these conditions, hydro-meteorological conditions must be such that there is sufficient heat loss from the water to cause water temperature to decrease to the freezing point. The physical parameters relevant to the formation of frazil ice include water temperature, air temperature, wind speed, and humidity.

In lakes, blockages of intakes are associated with open water, low temperatures, and clear nights. They also are often associated with strong winds, which increase the rate of heat loss at the water surface as well as potentially provide turbulence that can mix the supercooled water to the depth of the intake. The intake flow can also entrain the supercooled water if it is of sufficient velocity. The depth at which a lake intake will be free from the impacts of frazil ice is also dependent on other factors, such as lake bottom topography and intake structure dimensions.

Frazil ice is considered a potential hazard for causing water intake blockage to the BWRX-300 facility.

2.2.8 Characterization of Ice Storm Hazard

The impact of ice storms on the safe operation of the reactors at the Darlington Nuclear site was considered in the 2009 NK054-REP-01210-00008 (Reference 2.2-1) for DNNP and assessed in the 2019 NK38-REP-03611-10043 (Reference 2.2-5) for DNGS.

Section 3.2 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) considered ice storms as part of the freezing rain assessment under rare meteorological events. The major ice storm event on record for the Darlington Nuclear site occurred in January 1998, over a period of 5 days. During the storm event, 80 -100 mm of freezing rain affected areas from Kingston to Granby, Quebec. On average, Toronto Pearson Airport recorded 17.1 hours of freezing rain per year, 8.8 days per year; while Trenton airport reported 21.9 hours of freezing rain per year and 11.4 days per year, as per the 2021 NK054-REP-01210-00078 (Reference 2.2-8).

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Freezing rain totals ranging from 50 mm to 75 mm have been reported on few occasions in southern Ontario; whereas 10 mm of freezing rain is to be expected occasionally and up to 20 mm of freezing rain is highly likely to occur over the time the site will be operational. Historically, freezing rain events with more than 50 mm have been observed in the same broad climatological region but are not frequent, with maximal amounts near 100 mm (refer to the 2009 NK054-REP-01210-00008 (Reference 2.2-1)).

The ice storm hazard for DNGS was assessed in the 2019 Darlington Hazards Screening Analysis NK38-REP-03611-10043 (Reference 2.2-5) and documented there in Subsection 4.5.5. The analysis reviewed OPG and CANDU Owners Group operating experience databases, as well as databases for other power plants. The review showed ice storms have not had an impact on the plants, but severe storms were seen to lead to losses of off-site power and switchyard failures in several cases. In 1998, Hydro Quebec experienced a loss of grid for several days due to an ice storm. During this ice storm, 40 mm of freezing rain was observed in Kingston, Ontario, and as much as 120 mm of freezing rain was observed in certain parts of Quebec.

For the DNNP BWRX-300, the LOPP event, an Anticipated Operational Occurrence (AOO), which is in the Pressure Increase Group and is designated as a BL-AOO event (refer to Chapter 15, Subsection 15.5.3.2.4).

2.2.9 Characterization of Electromagnetic Interference Hazard

Electromagnetic interference can affect the functionality of instrumentation and control equipment and can be initiated by both on-site sources, such as high-voltage switchgear and off-site sources such as communication networks. It has the potential of disrupting electrical components and instrumentation leading to potential impairment of critical plant control signals. This hazard was assessed in the 2009 NK054-REP-01210-00008 (Reference 2.2-1), the 2009 NK054-REP-01210-00019 (Reference 2.2-2), and the 2020 NK054-CORR-00531-10533 (Reference 2.2-4) for DNNP and in the 2019 NK38-REP-03611-10043 (Reference 2.2-5) for DNGS.

Section 2.1 of the 2009 NK054-REP-01210-00019 (Reference 2.2-2) identified this hazard for consideration in the design to provide the required shielding of critical components and “fail safe” wherever required.

Section 4.9 of the 2009 NK054-REP-01210-00008 (Reference 2.2-1) assessed external sources of electromagnetic interference including high-voltage transmission lines at DNGS and telecommunications towers. The assessment concluded that since electromagnetic interference sources are continuously present (including lightning induced electromagnetic interference), the risk of electromagnetic interference at the site must be addressed in the design basis of the new plant (currently, that is the BWRX-300 facility).

2.2.10 Characterization of On-site Methane Hazard

During initial site investigation, naturally occurring gas (methane) was found at/or near the bedrock/overburden interface in Boreholes DN-34, DN-41, DN-44, DN-48, DN53, DN-57, and DN-60 as described in Subsection 5.3.1 and Section 9.3 of the 2009 NK054-REP-01210-00011 Site Evaluation (Reference 2.2-6). Methane gas is harmful to the health of humans and is combustible. Methane is naturally produced at low-level from the bedrock by decaying vegetation from long ago.

Excavation near the bedrock/overburden interface will monitor for the methane gas and precautionary measures during construction will be taken per work documentation as required by the Canadian Centre for Occupational Health and Safety. For the RB excavation, the methane is expected to dissipate quicker than what was observed in the boreholes due to the significantly larger air space.

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2.2.11 References

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- 2.2-17 CNSC DEC 21-H107, October 26, 2021, "Record of Decision – Application to amend Power Reactor Operating Licence PROL 13.02/2025 to Authorize Production of Molybdenum-99 at the Darlington Nuclear Generating Station."

2.3 Proximity of Industrial, Transportation and Other Facilities

Information in Section 2.3 describes potential hazards associated with transportation network, industrial facilities and the DNGS which are proximate to the DNNP site.

2.3.1 Transportation Network

There are multiple transportation networks within, adjacent to, and around the Darlington Nuclear site that present potential risks to the BWRX-300 facility operation.

a. Canadian National Railway

The Canadian National Railway line bisects the Darlington Nuclear site and is primarily used to transport commuters (VIA Rail) with services from Toronto to Kingston, Montreal, and Ottawa. Significant number of passengers travel this route annually and tremendous cargo is transported annually on the line, including coal, forest products (e.g., lumber), chemicals, petroleum products (e.g., asphalt), automotive parts/products, and agricultural goods (e.g., fertilizer).

Given the high frequency of both commuter and cargo traffic on this railway line, there is a potential risk of train derailment at the site. This risk is mitigated to some degree as the railway line is well buffered by berms on both sides of the railway corridor that would involve any possible derailment. In addition, VIA Rail announced in 2021 it was embarking the High Frequency Rail project that will divert a portion of the commuter rail to a separate line to relieve congestion on the current line and avoid congestion risks with cargo/freight shipments.

Additional information on hazards related to rail transportation accidents is provided in Subsection 2.2.3.2.

b. Highway 401

Highway 401, its official name Macdonald–Cartier Freeway, is a controlled-access 400-series highway stretching from Windsor in the west to the Ontario–Quebec border in the east. The highway runs along the north of the Darlington Nuclear site boundary as a six lane (three east-bound lanes and three west-bound lanes) highway.

Information on transportation risk associated with the 401 highway is described in Subsection 2.2.3.3.

c. Oshawa Executive Airport

The Oshawa Executive Airport, owned and managed by the City of Oshawa, is located northwest of the Darlington Nuclear site. It is located on an approximately 2.0 km² site with a modern terminal building and dual runways measuring approximately 1296 m and 809 m, respectively, to service different types of aircraft. The airport is required by the federal government to operate until 2047 but may close prior to 2047 (but not before 2033 at the earliest) if Pickering airport is opened. In 2018, total aircraft movement at the airport was over 78,000.

Information on risk associated with air transportation is presented in Subsection 2.2.3.1.

2.3.2 Industrial Facilities

There are few industrial facilities in proximity to the east of the DNNP site and to the west of the DNGS site that could cause potential risks to the BWRX-300 operation. Details are presented on such facilities in Subsection 2.1.1, and on pertinent potential hazards in Subsection 2.2.4 and Subsection 2.2.6.

2.3.3 Darlington Nuclear Generating Station Site

There are numerous activities at the DNGS that may impact the operation of the BWRX-300. The following activities apply:

- a. OPG uses arial photography drones, for inspection of the exterior of some of the DNGS buildings, as well as systems and components. The hazard of such drone crashing on the BWRX-300 buildings is bounded by the design basis automobile tornado missiles (refer to Subsection 2.6.6).
- b. Chemicals and gases used at the 2019 DNGS NK38-REP-03611-10043 (Reference 2.3-1) are screened out on the basis:
 - That their impact is bound by the impact of similar chemicals on the BWRX-300 (refer to Section 2.4, Table 2.4-1)
 - Of distance from the DNNP site
 - Of the probability of occurrence of relevant accidents.

Refer to Subsection 2.2.5.3 and Subsection 2.2.5.4 for additional and detailed coverage of other hazards related to the operation of the DNGS, or activities being undertaken at the DNGS site.

2.3.4 References

- 2.3-1 NK38-REP-03611-10043 R003, 2019, "Hazard Screening Analysis - Darlington," Ontario Power Generation.

2.4 Plant Site Activities Influencing Plant Safety

Section 2.4 includes two subsections:

- Subsection 2.4.1, which evaluates processes and activities at the DNNP site that, if incorrectly carried out, could affect or influence the safe operation of the BWRX-300 facility
- Subsection 2.4.2, which discusses measures for site and shoreline protection.

2.4.1 Site Hazards

Subsection 2.4.1 is limited to processes and activities at the DNNP site. Activities at DNGS or other off-site industrial locations are considered in Section 2.3. Subsection 2.4.1 information is focused on the following site-specific sources of hazards:

- Potentially explosive gases – Subsection 2.4.1.1
- Flammable vapour clouds – Subsection 2.4.1.2
- Toxic chemicals – Subsection 2.4.1.3
- Fire and smoke – Subsection 2.4.1.4

Table 2.4-1 provides a listing of gases and chemicals stored on the DNNP site.

Table 2.4-1: Summary of Gases and Chemicals Stored on DNNP Site

Chemical/Material (Formula/Trade/State)	Location (subject to change)	Quantity	Hazard Screening
Nitrogen	Gas Storage Area West of TB	Approximately 50 m ³ (Cryogenic Storage Tank)	Nitrogen is evaluated as potential asphyxiant concern for MCR and SCR habitability.
Hydrogen	Gas Storage Area West of TB	Each cylinder stores 356.1 standard cubic meters (SCM).	Hydrogen is a potential explosive and fire concern. Minimum separation distance between the cylinders and the BWRX-300 RB wall is determined based on explosive potential.
Diesel Fuel	Tank North of the Protected Area Access Building	Approximately 114 000 L Tank	Not a toxic or explosive hazard. Potential of fire hazard is addressed in Chapter 9A, Section 9A.6.
Turbine Oil	Tank North of the Protected Area Access Building	Approximately 20 000 L tank (volume of the tank does not impact MCR habitability)	Not a toxic or explosive hazard. Potential fire hazard is addressed in Chapter 9A, Section 9A.6.
Sodium Hypochlorite (7 to 15% Solution)	Adjacent to Pumphouse/Forebay, and Intake Shaft	Approximately 4000 L tank	Sodium hypochlorite is not considered a hazard due to being a liquid at 37.8 °C (100 °F) and normal atmospheric pressure. Sodium hypochlorite has a relatively low vapour pressure. Due to the relatively low vapour pressure, no significant unreported and prolonged release that could affect MCR habitability would be expected even in the event of a major spill.
Sodium Bisulphite (24 – 38% Solution)	Adjacent to Pumphouse/Forebay, and Intake Shaft	Approximately 11 400 L tank	Based on chemical safety data sheet sodium bisulphite is relatively stable. Sodium bisulphite is not considered a hazardous substance based on an absence of associated Immediately Dangerous to Life and Health exposure limits in National Institute of Occupational Safety and Health.
Captor Thiosulphate Dichlorination	Adjacent to Pumphouse/Forebay, and Intake Shaft	Approximately 11 400 L tank	Based on chemical safety data sheet captor thiosulphate is not a toxic hazardous substance.
Gasoline	Vehicle Maintenance Garage	Approximately 20 L containers	Gasoline is a potential explosion and fire concern. Small quantities do not pose a significant hazard.
Propylene Glycol	Within the P25, Chilled Water System, throughout the Power Block	39 000 L	Based on chemical safety data sheet propylene glycol is not a toxic hazardous substance for MCR habitability.

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Chemical/Material (Formula/Trade/State)	Location (subject to change)	Quantity	Hazard Screening
Tetrafluoroethane (R-134a Refrigerant)	P25 Chillers on RadWaste Building Roof	Each Chiller contains a refrigerant charge of 250 kg	R134a is not a toxic hazard for MCR habitability. Release of the entire contents of the R-134a into the Control Building does not result in an oxygen-deficient environment in the MCR.
Noble Metal Solution	Reactor Building	Approximately 38 L of 1% noble metal solution is utilized over a 2-week time frame per year.	The noble metal solution is not considered a hazard to MCR habitability based on an absence of associated Immediately Dangerous to Life and Health exposure limits in National Institute of Occupational Safety and Health. A potential release will be relatively confined to the RB and not impact MCR habitability.
Depleted Zinc Oxide	Turbine Building (TB)	90 kg dissolution vessel (quantity does not impact MCR habitability)	Zinc oxide is not considered a hazard to MCR habitability based on an absence of associated Immediately Dangerous to Life and Health exposure limits in National Institute of Occupational Safety and Health. A potential release of zinc oxide dust will be relatively confined to the Turbine Building and not impact MCR habitability.

2.4.1.1 Potentially Explosive Gases

The nearest source of potentially explosive gases is the hydrogen gas storage cylinders for the Reactor Water Chemistry System. Table 2.4-1 lists the maximum quantity of hydrogen stored at this location. The hydrogen is stored in several cylinders.

The safe separation distance between the hydrogen storage area and nearest safety-related structure is determined using a methodology such as the approach in EPRI NP-5283-SR, Guidelines for Permanent BWR Hydrogen Water Chemistry Installations, September 1987 (Reference 2.4-1). In the 1987 EPRI NP-5283-SR (Reference 2.4-1) the required separation distance is determined for two different considerations. The first consideration is the required separation distance such that the safety-related structure is not adversely affected by the postulated hydrogen explosion. The second consideration is the required separation distance to air pathways into safety-related structures versus the internal diameter of leaking high-pressure piping. The results of the determination of required separation distance are considered in establishing the layouts for the DNNP site and BWRX-300 facility.

2.4.1.2 Flammable Vapour Clouds

There are no liquids stored on the DNNP site that can generate a significant quantity of flammable vapour.

2.4.1.3 Toxic Chemicals

Table 2.4-1 identifies the chemicals on the DNNP site that are considered in the evaluation of potential toxic chemical hazards. Table 2.4-1 identifies the chemical, the quantity, and how the chemical is dispositioned. Chemicals are initially evaluated based on relative location, quantity stored, toxicity, and properties such as vapour pressure. As shown in Table 2.4-1, from a toxic chemical perspective, the potential hazards at the DNNP site except for nitrogen are dispositioned as not being hazardous for control rooms habitability. The liquid nitrogen, however, cannot be screened out and requires a detailed evaluation.

The threat from nitrogen is displacement of oxygen. No specific acceptance criterion is provided for limiting concentrations, and nitrogen is not considered a toxicity hazard. Nitrogen impacts control room habitability if it displaces sufficient quantities of air to the extent that oxygen levels in the room decrease below a specified threshold. Chemicals are asphyxiating if they result in an oxygen-deficient atmosphere of less than 19.5% oxygen by volume, as defined by the Canadian Centre for Occupational Health and Safety.

As described in Chapter 6, Section 6.4, control room habitability is served by a combination of individual systems that collectively ensure that continued occupancy in the MCR or SCR is possible under Postulated Initiating Events (PIEs) for a minimum of 72 hours as required by REGDOC-2.5.2 (Reference 2.4-4).

Two different scenarios are considered: a tank burst and a tank leak. In the tank burst scenario, all the contents of the tank are instantaneously released. For the tank leak scenario, the nitrogen is leaked at a constant mass flow rate until the tank is empty over an assumed time. Inputs to the analyses include meteorological stability classification, wind speed, air temperature, and the assumed leak rate for the tank leak scenario. Several sensitivity cases are run to determine the limiting input values. For each location, the control room ventilation system is modeled in the analyses to credit the effects of intake and dilution within the control room atmosphere during the passage of the plume.

The limiting results from the analyses of the postulated nitrogen tank burst and leak scenarios are used to confirm that the placement of the tank relative to the MCR and SCR ventilation intakes is acceptable.

2.4.1.4 Fire and Smoke

On-site flammable and combustible liquid or gas storage facilities are designed in accordance with applicable fire codes, and plant safety is not jeopardized by fires or smoke in these areas. A detailed description of the fire protection system, as well as the Fire Hazard Assessment (FHA) methodology is presented in Chapter 9A, Section 9A.6.

2.4.2 Measures of Site Protection

As described in Subsection 2.7.1, the plant grade elevation at 88 m CGD is established using grading and engineered fill. Excavation is performed to depths to reach materials of specific properties suitable for buildings foundations. Materials removed during the excavation are reconditioned for use as backfill material if the material meets the required specifications or are disposed as spoils. Engineered fill material requirements are specified in Subsection 2.7.5.2.1.

The hydrology for the site and vicinity is described in Section 2.5. The site does not credit dams or dikes for flood protection. As described in Section 2.5 the topography and grading at the plant site are considered in the site flooding analyses to demonstrate the plant is adequately protected from precipitation events.

As described in the 2022 NK054-REP-07730-00055 DNNP Environmental Impact Statement [EIS] Review Report for BWRX-300 (Reference 2.4-2), the BWRX-300 deployment will not require lake infilling and, consequently, the associated adverse effects on site drainage and water quality will not occur. The BWRX-300 deployment will still require some shoreline protection works, but such works are expected to be smaller in scale resulting in smaller residual adverse effects on shoreline processes than those assessed in the 2009 EIS for no specific reactor technology NK054-REP-07730-00029 (Reference 2.4-3).

The construction of the first BWRX-300 would provide an opportunity to retain the Bank Swallow nesting habitat as the bluff would be remaining in place and as the impact of excavation and construction activities will be kept to a minimum, per the 2022 NK054-REP-07730-00055 (Reference 2.4-2). When the DNNP site is built out to include a total of four BWRX-300 reactors, additional shoreline protection is to be implemented to stabilize the shoreline as described in the 2022 NK054-REP-07730-00055 (Reference 2.4-2).

The specific extent and location of the shoreline protection works is determined in later phases of the project.

2.4.3 References

- 2.4-1 EPRI NP-5283-SR, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," 1987, Electric Power Research Institute.
- 2.4-2 NK054-REP-07730-00055, 2022, "Darlington New Nuclear Project Environmental Impact Statement Review Report For Small Modular Reactor BWRX-300," Ontario Power Generation.
- 2.4-3 NK054-REP-07730-00029, 2009, "Environmental Impact Statement New Nuclear - Darlington Environmental Assessment," Ontario Power Generation.
- 2.4-4 CNSC Regulatory Document REGDOC-2.5.2, Version 1.0, "Design of Reactor Facilities: Nuclear Power Plants."

2.5 Hydrology

2.5.1 Introduction

Section 2.5 describes the hydrological conditions and their potential implications relevant to the DNNP site. Section 2.5 includes information on:

- The adequacy of the cooling water supply from Lake Ontario along with risks to the water supply (i.e., biofouling and frazil ice) - Subsection 2.5.2
- The potential flooding hazards, including the Probable Maximum Precipitation (PMP), Probable Maximum Flood (PMF), as well as flooding potential from runoffs, rivers, waves, storm surge and seiche, tsunami, and ice jamming - Subsection 2.5.3
- The potential impact of climate change - Subsection 2.5.4
- Assessment and monitoring of radionuclide dispersion in the groundwater – Subsection 2.5.5
- Assessment and monitoring of radionuclide dispersion in surface water – Subsection 2.5.6

Key hydrological characteristics and parameters described in Section 2.5 relevant to the DNNP site and the surrounding area are summarized and listed in Table 2.5-1. The list includes information on Lake Ontario adequacy as a water supply for use as a heat sink, maximum precipitation and flooding and associated probabilities, as well as surface and subsurface geotechnical properties relevant to transport of radionuclides.

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Table 2.5-1: Hydrological Characteristics Summary of DNNP Site and Surrounding Area

2.5.2 Description of Heat Removal Methods and Heat Sink			
Normal Heat Removal / Normal Heat Sink	The NHS is a once-through cooling water source from Lake Ontario to the CWS and the PCW		
Ultimate Heat Removal / Ultimate Heat Sink	The Isolation Condenser System consists of three high-pressure reactor isolation loops that passively remove heat from the reactor when the normal heat removal system is unavailable.		
2.5.2.1 Description of Lake Ontario Water Levels and Adequacy of Water Supply			
Water Level	Controlled by the International Joint Commission		
Variability of Water Level (at the intake)	Lowest water level	73.71 m (statistical data at Cobourg Water Level Station) (Reference 2.5-18)	→ 73.71 m
	Impact of seiche	0.75 m (reduction)	→ 72.96 m
	Wave trough (1 s passage)	4.08 m (reduction)	→ 68.88 m
	Spring tides	Less than 5 cm (hidden as part of normal fluctuation)	→ 68.88 m
	Wave downwash	Close to the shoreline with no effect	→ 68.88 m
	Tsunami	No risk expected	→ 68.88 m
Water Depth Available	Normal Conditions 73.71 – 62.50 m	11.21 m above the intake level of 62.50	Therefore, water supply is adequate under normal and extreme conditions
	Impact of Seiche 72.96 – 62.50 m	10.46 m above the intake level of 62.50	
	Impact of Wave Trough (1s duration) 68.88 – 62.50 m	6.38 m above the intake level of 62.50 m	
2.5.2.2 Potential Impacts of Biofouling on Water Supply			
Algae	Algae have the potential to be entrained at cooling water and water supply system intakes, resulting in blockage or restriction issues.		
Micro-biologicals	Biological coatings or biofilms and particulate deposition on tube surfaces can cause lost flow capacity, extensive repairs and material replacement costs in heat exchangers, fire protection systems, storage vessels, intakes, and water distributions systems.		
Macrophytes	Macrophytes can contribute to macrofouling through sticks, leaves and other plant constituents from either terrestrial or aquatic sources that become a component of lake drift and debris material.		
Mollusks	Zebra and quagga mussels can clog water intake structures, such as screens, tunnels and pipes.		
Fish	Various life stages of fish can be taken into a cooling water system with the cooling water (entrainment), and consequently fish reach screens that protect the cooling water and other water systems (impingement). An excessive load of fish can cause blockage to the screening system and sump. In extreme events where screens become overloaded water supply can be reduced with associated reduction in power supply.		
2.5.2.3 Potential Impacts of Frazil Ice Accumulation on Water Supply			
Frazil Ice Accumulation	Accumulation of frazil ice on the intake trash rack, which can partially or completely block the trash rack and rapidly and unexpectedly shut down the intake facility		

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2.5.3 Description of Potential Sources of Flooding		
2.5.3.1 Flooding Due to PMF	PMP	420 mm 12-hour precipitation, equivalent to 420 mm of total rainfall, with 51% in the 6 th hour with a return period of 1:1,000,000 year (Reference 2.5-19)
	Design Basis Flood	<u>Conservative Rainfall</u> : Standardized value of 12-hour PMP in Ontario of 420 mm, with zero infiltration (which greatly exceeds Hurricane Hazel in depth and intensity)
	PMF – Screened in	The event scenario involves a large volume of water runoff flooding the site (based on the application of PMP), while the drainage systems are blocked (due to debris or ice pellet), the soil nearby is saturated, and the lake water level is at 100-year high. Also, it is conservatively assumed that there is no time for implementing preventative measures or taking mitigating actions. The PMF sequence is expected to be worse than a lake level increase or heavy precipitation alone, and the event is not bounded by any other events. As such, flooding due to PMF could not be screened out based on screening criteria [QL1] through [QL5].
	Design Basis Flood Level	Using design basis flood (that is, PMP with zero infiltration), for modeling drainage for BWRX-300 Unit-1 or an assumed 4-unit layouts, Section 5.4.3 of Reference 2.5-18 resulted in flood water levels of up to 87.93 m CGD, considering realistic assumptions for stormwater infrastructure, including factors such as culverts sizing, conveyance, routing, and ponds.
2.5.3.2 Flooding Due to Runoffs	Natural or via Stormwater Management and infrastructure	<ul style="list-style-type: none"> Five of nine catchments drain directly to Lake Ontario or to Darlington Creek watershed. Remaining four catchments close to the BWRX-300 development area drain through a stormwater infrastructure directly to Lake Ontario and via engineered culverts stormwater infrastructure running to the southeast of DWMF to Lake Ontario (Reference 2.5-18). Measures are proposed to mitigate the impact of PMP flooding due to runoff.
	Screened out, per [QL2]	PMF bounds flooding caused by runoffs.
2.5.3.3 Flooding Due to Rivers	Screened out, per [QL3]	The distance, infrastructure, and topography between the Tooley Creek watercourse and the DNNP site precludes Tooley Creek as the source of a flood hazard. There is not any history of severe flooding along Darlington Creek within the recorded history of the area. This is confirmed by the (2022) Flood Hazard Assessment (Reference 2.5-18) that modeled drainage of Darlington Creek watershed under 100-year recurrence PMP.
2.5.3.4 Flooding Due to Waves	Screened in (related to Shoreline Protection issue)	Wave height of 6.1 m and peak period of 9.2 s is recommended (Reference 2.5-18) Data and models suggest wave uprush between 3.5 to 11.3 m, and overlapping from 0.015 to 0.591 m ³ /s/m. (Reference 2.5-2)
2.5.3.5 Flooding Due to Storm Surge and Seiche	Screened out	Models of most severe weather systems predicted a highest water level from storm surge or seiche of 0.75 m, per (Reference 2.5-2), and (Reference 2.5-18). The margins between the lake level and the top of the DNNP breakwater works are larger than 0.75 m.

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2.5.3.6 Flooding Due to Tsunami	Screened out, per [QL3]	A tsunami in Lake Ontario is an improbable event for DNNP.
2.5.3.7 Flooding Due to Ponds, Dams or Dikes	Screened out, per [QL3]	There are no large permanent human-made water storage ponds, dams or dikes near the Darlington Nuclear site that can threaten the site.
2.5.3.8 Flooding Due to Ice Jamming	Screened out per [QL2] or [QL3]	Bounded by the detailed PMF analysis (Reference 2.5-4); or based on the conclusion of negligible ice forming in Lake Ontario near the DNNP region (Reference 2.5-18).
2.5.4 Potential Effects of Climate Change		
Effect on Temperature, Precipitation, Lake Water Level	Screened in	<ul style="list-style-type: none">Some models showed increase in the intensity (about 14%) and frequency (about 22%) of extreme precipitation in southern Ontario (Reference 2.5-2)Maximum found historical lake water level is 75.6 m, which should be used as low estimate (Reference 2.5-13)For additional information, refer to the 2022 "Flood Hazard Assessment NK054-02730-00001 (Reference 2.5-18) and the 2023 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts NK054-PLAN-07007-00001 (Reference 2.5-20)
2.5.5 Groundwater		
2.5.5.1 Groundwater Conditions	Described in detail in Subsection 2.7.3.2.4.	
Groundwater Flow System	Categorized into three hydrostratigraphic units: Shallow/Water Table; Interglacial Deposits; and Shallow Bedrock. In general, groundwater flows from north to south, and discharges toward Lake Ontario.	
Groundwater Level	Groundwater is anticipated to be present between elevation 80 to 85 m corresponding to depths of between 3 and 8 m below the plant grade at elevation 88 m.	
Monitoring	Environment Monitoring Program is employed along with the use of groundwater wells that are located in key areas of the Darlington Nuclear site, including protected areas, controlled areas, and site perimeter.	
2.5.6 Surface Water		
2.5.6.1 Surface Water Properties		
Water movement near the site is predominantly along the shore, occurring for 73% of the time (35% to the west and 38% to the east).		
Depth Averaged Speed – all directions	12.4 cm/s	
Depth Averaged Speed – Easterly	14.1 cm/s	
Depth Averaged Speed – Westerly	11.3 cm/s	
Temperature	Lake-wide surface temperatures typically rage from freezing in winter to approximately 20 °C in summer.	
Ice Conditions	Typically, are limited to the nearshore areas at the eastern end of the lake within the Kingston Basin.	
2.5.6.2 Surface Water Monitoring		
Lake Current Monitoring	A real-time current profile measurement system to be used in the event of a radiological liquid emission.	
Monitoring	Environment Monitoring Program is employed along with the Lake Current Monitoring system which a real-time current profile measurement system to be used in the event of a radiological liquid emission.	

2.5.2 Description of Heat Removal Methods and Heat Sink

The NHS System that is described in Chapter 9A, Subsections 9A.2.5 provides cooling water source and heat rejection means to support the function of the Circulating Water System (CWS) (Chapter 10, Section 10.8) to supply cooling water to the MCA system (Chapter 10, Section 10.5), as well as to interface with the PCW (Chapter 9A, Subsection 9A.2.1). The NHS is a once-through cooling system using water from Lake Ontario. The water flows through the intake tunnel via the onshore intake vertical shaft to the Pumphouse/Forebay where the circulating water pumps deliver the cooling water to the MCA and PCW heat exchangers before returning the heated water back to the lake via the onshore discharge vehicle shaft through the discharge tunnel to the risers/diffusers.

The BWRX-300 Isolation Condenser System (ICS), described in Chapter 6, Section 6.2, consists of three independent trains, each containing a heat exchanger or Isolation Condenser (IC) that is submerged in a dedicated pool of water. The ICS provides the ultimate heat sink for protecting the reactor core for any off-normal event where the main condenser is not available, and the Reactor Pressure Vessel (RPV) is isolated.

The ICS Pool Cooling and Cleanup System (ICC) that is described in Chapter 9A, Subsection 9A.2.6 is designed to precondition and maintain the ICS pools in a state of readiness for postulated events that require reactor decay heat removal.

The FPC, as described in Chapter 9A, Subsection 9A1.3, has a primary function to provide continuous cooling of the water volume in the fuel pool to remove decay energy from irradiated fuel, and to provide replacement coolant inventory from a variety of sources, both to ensure irradiated fuel is kept cool and submerged under water throughout the life of the plant.

2.5.2.1 Description of Lake Ontario Water Levels and Adequacy of Water Supply

Lake Ontario is one of the main reservoirs of cooling water for the DNNP site. An assessment for the adequacy of water supply to DNNP was completed in the 2009 NK054-REP-01210-00018 (Reference 2.5-1) and validated in the 2022 Flood Hazard Assessment NK054-REP-02730-00001 (Reference 2.5-18), as described in the following paragraphs.

The water level in Lake Ontario is regulated by the International Joint Commission to reduce damages along the shores of the lake and the St. Lawrence River, per the 2022 Flood Hazard Assessment NK054-REP-02730-00001(Reference 2.5-18). The control of water levels by the International Joint Commission continues in the future and, though the plan for regulation may change, the fundamental function of eliminating extreme lake levels remains. However, the International Joint Commission acknowledges that it may become increasingly difficult to maintain levels within their currently defined operating band depending on the relevant impact of climate change in the future (refer to Subsection 2.5.4 which discusses the impact of climate change on Lake Ontario water levels). Careful consideration of the International Joint Commission study for management options, which included robust modeling of potential future levels under a range of stochastically generated hydrological and meteorological conditions, led to estimates greater than 100-year recurrence low water levels at 73 m as reported in Subsection 5.1.5 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2). However, analysis of historical data at the Water Survey of Canada Cobourg Water Level Station shows a minimum water level of 73.71 m, as reported in the 2022 Flood Hazard Assessment, NK054-REP-02730-00001 (Reference 2.5-18).

Additional factors which influence the minimum water level at the intake were considered in the 2009 NK054-REP-01210-00018 (Reference 2.5-1) as follows:

1. A numerical model of the hydrodynamics of Lake Ontario was developed to assess the potential for generation of surge and seiche in response to extreme severe weather

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systems tracking through the region. The maximum wave heights expected at the intake location will be depth limited. The lowest water level of 73.71 m, further lessened by 0.75 m due to seiche, yields an elevation of 72.96 m or a depth of 10.46 m at the intake of an elevation of 62.50 m.

2. Estimating wave breaking at about 0.78 times the water depth of 10.46 m yields maximum wave heights of about 8.08 m. An associated wave trough, taken as half the maximum wave height (that is 4.8 m), might reduce the depth to 6.38 m, though it is noted that the passage of large waves would be short-lived and on the order of 1s. (Note: The 8.16 m Maximum wave height is more conservative than the maximum wave height of 6.1 m recommended in Subsection 2.5.3.4.)
3. The largest spring tides in Lake Ontario are less than 5 cm in height and these minor variations are hidden by greater fluctuations in lake levels produced by wind and barometric pressure changes. Consequently, Lake Ontario is considered to be essentially non-tidal.
4. Wave downrush would occur within a relatively close distance to the shoreline and would have no effect on the water level near the intake.
5. The 2009 flood hazard assessment (Reference 2.5-2) concluded there is no risk of tsunamis so that there is no drawdown potential from that phenomenon that could affect nearshore lake levels. The 2022 Flood Hazard Assessment (Reference 2.5-18) also concluded the Darlington Nuclear site lies in a region with a low probability of tsunamis.

Consequently, even under the extreme scenario considered in the 2009 NK054-REP-01210-00018 (Reference 2.5-1) and the 2022 NK054-REP-02730-00001 Flood Hazard Assessment (Reference 2.5-18), a depth of more than 6 m remains over the intake at the lakebed elevation. Therefore, lake water supply is adequate for the DNNP cooling water intake.

Given the adequacy of the water supply from Lake Ontario, the potential for using groundwater sources in extraordinary situations is not considered.

Consideration for additional factors which might impact the availability of the cooling water supply were also assessed in the 2009 NK054-REP-01210-00018 (Reference 2.5-1), namely concerns related to biofouling and frazil ice conditions. These are discussed separately in the following two subsections.

Additional information on Lake Ontario's current, temperature, and ice conditions is provided in Subsection 2.5.4.2.

2.5.2.2 Potential Impacts of Biofouling on Water Supply

1. Algae: The Lake Ontario shoreline provides a favorable growth environment for Cladophora which are prominent nuisance filamentous algae that have the potential to affect the DNNP. Cladophora characteristically grows attached to hard surfaces within the littoral zone and where habitat conditions are optimal, thick mats of the algae can form across the lake substrates and become attached to infrastructure features. During mid-summer and fall, Cladophora senesces, the algae become detached from the substrate and drift in a suspended manner with waves and currents.

The loose filaments as well as more substantial clumps of algae have the potential to be entrained at cooling water and water supply system intakes, resulting in blockage or restriction issues at the inlet as well as further blockage and organic material loading at the trash racks or travelling screen system.

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2. Micro-biologicals: Biofilms consist of microorganisms immobilized at a substratum surface, typically embedded in an organic polymer matrix of bacterial origin. Such biofilms are ubiquitous in flowing aqueous environments, are not necessarily uniform in time and space, and may trap inorganic substances within the polymer matrix. Biofilms develop on virtually all surfaces immersed in natural aqueous environments, irrespective of whether the surface is biological (aquatic plants and animals) or abiological (stones, particles, metal, and concrete, etc.). Extensive bacterial growth, accompanied by excretion of copious amounts of extracellular polymers, thus leads to the formation of visible slimy layers (biofilms) on solid surfaces.

Thin biological coatings or biofilms associated with microorganisms can reduce the efficiency of heat exchangers (forcing shutdowns or de-rating), enhance silt and particulate deposition on tube surfaces (causing fouling and pipe wall pitting), lost flow capacity, extensive repairs and material replacement costs in heat exchangers, fire protection systems, storage vessels, intakes, and water distributions systems.

3. Macrophytes: Both terrestrial and aquatic plants can contribute to floating and suspended plant material that becomes susceptible to entrainment at water intakes. A variety of rooted aquatic macrophytes are common to Lake Ontario. The existing DNGS forebay was shown to contain a community of Eurasian watermilfoil (*Myriophyllum spicatum* L.), the only rooted plant observed. The biomass of this material was estimated at 1.5 tons providing an indication of the potential availability of organic mass that can contribute to the load on the screening system. A future regional increase in aquatic plants and algae was concluded as being a reasonable expectation as the lake water clarity increases with the filtering effects of the exotic invader zebra and quagga mussel.

Macrophytes can contribute to macrofouling through sticks, leaves and other plant constituents from either terrestrial or aquatic sources that become a component of lake drift and debris material. During the fall season when macrophytes typically senesce, the organic material of the plant stems and foliage have the potential to fragment and block travelling screens.

4. Mollusks: Lake Ontario contains confirmed populations of non-native invasive nuisance mussels including the zebra mussel, *Dreissena polymorpha*, and the quagga mussel, *Dreissena rostriformis bugensis*, inadvertently introduced to North America in the ballast water of oceangoing ships. More recent colonization has involved the quagga mussel, which has a preference for deeper, cooler water as compared to the zebra mussel and has now largely replaced the zebra mussel. Given the record of non-native introductions to Lake Ontario, additional nuisance mollusk species may appear in the future. The Asiatic clam *Corbicula fluminea* has been recorded in North America the longest of the three key invasive species arriving on the west coast in the 1920s and reaching the east coast by 1980s; however, it has not yet been reported as an issue in Lake Ontario.

Dreissena species ability to rapidly colonize hard surfaces causes serious economic problems and potential reduced efficiency of water supply systems. These major biofouling organisms can clog water intake structures, such as pipes and screens, therefore reducing pumping capabilities for power and water treatment plants. Power plant features that may become fouled include crib structures, trash bars, screenhouses, steam condensers, heat exchangers, penstocks, service water systems and water level gauges.

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5. Fish: Lake Ontario hosts a diverse population of both warm and cold-water fish species, many of which may utilize the project area either as local residents or seasonal migrants. During impingement investigations at DNGS operations from 1993 to 1995, fish encountered at the mitigative screen system and in sumps included at least 17 species. The predominant species were generally of a smaller body size which included alewife, shiner species and smelt, all representatives of the abundant forage fish-based community of the lake. Major community changes occurred with the introduction of non-native species through opening of waterways, intentional stocking, and unintentional introduction through ballast water of international shipping. This may have a bearing on future operational management systems at DNNP depending on the habits and productivity of a particular species.

Various life stages of fish can be taken into a cooling water system with the cooling water (entrainment), and consequently fish reach screens that protect the cooling water and other water systems (impingement). An excessive load of fish can cause blockage to the screening system and sumps contributing to maintenance requirements. In extreme events where screens become overloaded water supply can be reduced with associated reduction in power supply.

NOTE: The 2009 report NK054-REP-01210-00018 (Reference 2.5-1) concludes that mitigation measures have been successfully applied at power generating facilities along the north shore of Lake Ontario to address the various forms of biofouling.

2.5.2.3 Potential Impacts of Frazil Ice Accumulation on Water Supply

As described in the 2009 NK054-REP-01210-00018 (Reference 2.5-1), operating water intakes in lakes and rivers in northern regions is complicated by the presence of ice. Controlling the generation and accumulation of frazil ice affects both navigation and power generation. The cooling water intake tunnel can accumulate frazil ice on the intake trash rack, which can partially or completely block the trash rack and rapidly and unexpectedly shut down the intake facility.

2.5.3 Description of Potential Sources of Flooding

Subsection 2.5.3 describes the assessment of potential flood hazards at the DNNP site. (Refer to Subsection 2.1.1 for information on the topography of the Darlington Nuclear and DNNP sites.)

The review of the flood hazard assessment performed in support of the 2020 DNNP Power Reactor Site Preparation Licence (Reference 2.5-3) against the 2019 codes and standards concluded there is no impact on the conclusion of the 2009 Flood Hazard Assessment NK054-REP-01210-00012 (Reference 2.5-2) as documented in the 2019 DNNP Site Preparation Licence Renewal activity report NK054-REP-01210-00108 (Reference 2.5-7).

Also, as stated in Subsection 4.5.3 of the 2020 NK054-CORR-00531-10533 (Reference 2.5-3), the results of the 2019 Darlington Hazard Screening Analysis NK054-REP-03611-10043 (Reference 2.5-4) apply to the DNNP site since the DNNP site is encompassed in the Darlington Nuclear site.

As described and assessed in the 2019 NK054-REP-03611-10043 (Reference 2.5-4), and in the 2022 NK054-REP-02730-00001 Flood Hazard Assessment (Reference 2.5-18), the DNNP flooding hazards are:

- Flooding due to PMF - Subsection 2.5.3.1
- Flooding due to Runoffs - Subsection 2.5.3.2
- Flooding due to Rivers - Subsection 2.5.3.3

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- Flooding due to Waves - Subsection 2.5.3.4
- Flooding due to Seiche - Subsection 2.5.3.5
- Flooding due to Tsunami - Subsection 2.5.3.6
- Flooding Due to Ponds, Dams or Dikes - Subsection 2.5.3.7
- Flooding due to Ice Jamming - Subsection 2.5.3.8

These hazards are addressed in the following subsections.

2.5.3.1 Flooding Due to Probable Maximum Flood

As described in Section 5.4 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2), the design storm event used to determine the flood hazard is the PMF event in the 2011 International Atomic Energy Agency (IAEA) SSG-18 (Reference 2.5-10). This is a specific hydrologic term that is defined in conjunction with the PMP, as per the following paragraphs.

The PMF is the flood that may be expected from the most severe combination of critical meteorological and hydrologic conditions that are reasonably possible in a particular drainage area. The PMP is defined as the greatest depth of precipitation for a given duration meteorologically possible for a given size storm area at a particular location at a particular time of year, with no allowance made for long-term climatic trends. It is a common practice that the PMF is the flood which is a direct result of the PMP. The PMP is applied to sub-basin delineations that account for variations in soil type, land use, size and shape of the watershed, and average watershed slope to generate PMF flows.

There are two considerations when determining the PMP for a given application, the site location, and the duration of the storm event. Based on the 2017 Lakes and Rivers Improvement Act Technical Guidelines (Reference 2.5-11), for watershed areas less than 1295 km², the PMP maximum precipitation duration of 6 or 12 hours is normally used as it produces the highest peak flood flow

Subsection 4.4.1 of the 2019 NK054-REP-03611-10043 (Reference 2.5-4) states that the Review Level Condition assumes no runoff in the worst hour of the 12-hour PMP; therefore, the flood depth is 51% of the total 12-hour PMP of 420 mm, which is approximately 214 mm, per Table 5.4-1 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18). The PMF event scenario involves a large volume of water runoff flooding the site, while the sewer systems are blocked (due to debris or ice pellet), the soil nearby is saturated, and the lake level is at 100-year high. This PMF sequence is expected to be worse than a lake level increase or heavy precipitation alone, and the event is not bounded by any other events. Finally, it is conservatively assumed that there is no time for implementing preventative measures or taking mitigating actions. As such, flooding due to PMF could not be screened out based on screening criteria [QL1] through [QL5] (refer to Subsection 2.2.2 for descriptions of the screening criteria).

The PMF values which are commonly estimated using a combination of flood-inducing drivers such as snowmelt and rainfall can alternatively be estimated using an extreme rainfall outside the snow season that is higher than spring values. In Subsection 5.4.1 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18), it is assumed that the summer PMP produces extreme floods (i.e., PMFs) at least comparable to the spring PMFs that consider snowmelt. This assumption was verified by comparing the precipitation values of spring (March-April) with summer-fall (May-November); so that a summer PMP can be deemed as the key driver of the PMF, per the 2022 NK054-REP-02730-00001 (Reference 2.5-18).

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As described in Subsection 2.1.1 of the 2022 NK054-REP-02730-00002, PMP Validation (Reference 2.5-19), the PMP for watershed areas in the vicinity of and the DNNP site is a 12-hour precipitation equivalent to 420 mm of total rainfall, with 51% of the storm falling in the sixth hour, with a return period of 1:1,000,000. This value is on the conservative side considering the historical observed 24-hour point rainfall in the region is 212 mm (hurricane Hazel).

The design basis flood is the flood resulting from the PMP assuming zero infiltration in the drainage areas on site. In Subsection 5.4.1 of the 2022 Flood Hazard Assessment NK054-REP-01730-001 (Reference 2.5-18) design flood values in the DNNP site region are based on the 1:100-year return period storm or Hurricane Hazel, whichever is the greater. The 1:100-year return period storm was used to calibrate a Darlington Creek model, and as a comparison to the PMP results (refer to Table 2.5-2 under Subsection 2.5.3.3). For small watersheds such as Darlington Nuclear site, where no stream gauge is available, 1:100-year return period rainfall is assumed to produce a 1:100-year return period flood. Since the 420 mm, 12-hour duration PMP greatly exceeds Hurricane Hazel in depth and intensity, Hurricane Hazel was not used in this assessment. the 420 mm, 12-hour duration PMP was selected with zero infiltration as the current design basis storm for the DNNP, as shown in Table 2.5-2.

2.5.3.2 Flooding Due to Runoffs

Existing Pre-development Catchments and Flood Hazard

Section 3.2 of the 2022 NK054-REP-02730-00001 Flood Hazard Assessment (Reference 2.5-18) identified in Section 3.2 nine delineated catchments (A through I) for the pre-developed DNNP site, as shown in Figure 2.5.3.2-1. Information related to catchments A to I are provided in Table 3.2-1 of (Reference 2.5-18), such as area size, land use, soil/surface conditions and runoff. The runoff from Catchment A drains directly into Lake Ontario close to the DNGS forebay. The runoffs from Catchments B, C, D, and E in the north and east flow via the Canadian National Railway right-of-way ditch or through a wetland discharging into the Darlington Creek watershed. The runoffs from Catchments F, G, and H, which are former lay down areas in the DNNP site, flow through culverts southeast of the DWMF building and drain directly into Lake Ontario. The last runoff from Catchment I, a former lay down area, drains through various outlets into Lake Ontario. Potential existing on-site flood hazards include:

- Runoff from Catchments C and D overflowing the Canadian National Railway right-of-way ditch
- Capacity of designed stormwater infrastructure to convey potentially increased peak flows due to proposed DNNP site development.

Subsection 5.4.3 of the 2022 Flood Hazard Assessment (Reference 2.5-18) describes the flood hazard associated with a design basis flood involving PMP falling directly on the DNNP site, assuming 100% impervious land cover. The flood hazard due to direct precipitation is related to the ability of the site development to convey stormwater runoff through the site.

A nodal model (PCSWMM), per Subsection 5.4.3. of the 2022 Flood Hazard Assessment (Reference 2.5-18), of the nine catchments conveyance and retention as well as drainage structures was used to evaluate on-site flood hazards and to size conveyance and retention elements of stormwater for pre-development conditions.

The pre-development results indicate:

- None of Catchments A, B, E, G, and I pose a PMP flood risk on the DNNP site (refer to Table 5.4-11 of Reference 2.5-18)

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- Catchments C and D showed significant overflow into the Canadian National Railway right-of-way ditches with no flooding (refer to Table 5.4-12 of Reference 2.5-18)
- While the stormwater infrastructure in Catchment F performs adequately under, for instance, the 1:100-year storm, significant PMP/PMF overflow occurred between its sub-catchments or into neighboring Catchment H, suggesting development is necessary to alter Catchment F and its drainage system (refer to Table 5.4-13 of Reference 2.5-18)
- The stormwater conveyance and retention capacity of Catchment H represents a significant potential overflow under the PMP, between its sub-catchments within the existing infrastructure (refer to Table 5.4-14 of Reference 2.5-18)

These results were carried forward to explore and compare with the post-development results.

Post-development of BWRX-300 Unit 1 Catchments and Flood Hazard

A large portion of the pre-development areas of Catchments F and H would be replaced by Catchment N, within which the BWRX-300 Unit 1 footprint would almost entirely be contained, as shown in Figure 2.5.3.2.2. The runoff from Catchment N flows through a series of culverts, roadside ditches, and a pond to a southern outlet into Lake Ontario. The proposed site layout of the BWRX-300 Unit-1 facility will therefore have significant impact on-site catchments and runoff flow directions. Though the upstream reaches of these catchments will still mostly be intact, most of the pre-development of Catchment F and roughly half of Catchment H will be covered by the footprint of the BWRX-300 facility Unit-1 (refer to Figure 3.2-1 in Reference 2.5-18). Conveyance and retention structure of such catchments would consequently require re-configuration.

The same nodal model (PCSWMM) was used for post-development conditions including Catchment N. Culvert locations, diameters, conveyance (in m^3/s) and ditch depths were considered in the assessment. The post-development results for BWRX-300 Unit 1 indicate:

- Catchments A, B, C, D, E, G, and I do not represent a flood hazard for the DNNP site (refer to Subsection 5.4.3.4.1 in Reference 2.5-18)
- Under the PMP, there is significant flooding through the sub-catchments of Catchment F, and to Catchment G (refer to Table 5.4-16 in Reference 2.5-18)
- Current configuration of conveyance and retention structures in Catchment H will experience under the PMP significant flooding into its sub-catchments that may overtop into Catchment G (refer Table 5.4-17 in Reference 2.5-18)
- Catchment N system, comprising ditches, culverts, flood routes and storages, is sized to convey and retain adequately the PMP and is split into 12 sub-catchments described as follows:
 - Sub-catchment N_1 contains an administrative building and a parking lot and drains south through a culvert into N_10
 - Sub-catchment N_2 is a large laydown area, drains through ditch and outlets into N_12. In the model, all the flow within N_2 passes through a culvert adjacent to the Power Block, which is a conservative assumption to ensure N_2 runoff does not flood the Power Block area
 - Sub-catchments N_3 through N_7 contain the Power Block area, and each drain through a dedicated culvert into various downstream sub-catchments, with the culvert sizes chosen to ensure zero flooding of the Power Block area
 - Sub-catchment N_8 is a parking or laydown area draining through a culvert into N_12

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- Sub-catchment N_9 is a parking or laydown area draining through a culvert in N_10
- Sub-catchment N_10 is a low area adjacent to the Power Block containing a storm water management pond that drains to the south through a culvert into N_12
- Sub-catchment N_11 is a low area immediately south of the Power Block accepting flow from N_5 and N_7 and conveying through a culvert to N_12
- Sub-catchment N_12 is a perimeter ditch, accepting flows from the remainder of Catchment N and conveying toward the Catchment H outlets to Lake Ontario

Post-development BWRX-300 Unit 1 Modeled Available Freeboard

The post-development peak flow and flooding results for Catchment N, shown in Table 5.4-18 of (Reference 2.5-18), indicate with “realistic” assumptions (i.e., the largest culvert in the system is 1 m in diameter) for sizing of conveyance and retention structures, the maximum flood level within the Catchment N system is 87.93 m CGD. This provides 0.07 m of freeboard below the 88 m CGD construction grade which is a flood hazard, but by increasing the conveyance and retention capacity of the system, this freeboard can be brought to a comfortable level.

Comparison of Pre- and Post-development of BWRX-300 Unit 1 Results

Comparison of pre-developed and post-developed modelling results of BWRX-300 Unit 1 indicate (refer to Subsection 5.4.3.5 and Table 5.4-22 of Reference 2.5-18):

- There are no changes in Catchments A, B, C, D, E, G, and I.
- Maximum flood depth elevation changes between -0.02 m to +0.06 m in Catchment F since it is reconfigured from pre-development conditions.
- Maximum flood depth elevation changes between -0.23 m to +0.17 m occurred in Catchment H since it is also changed in post-development conditions, and it must convey and retain runoff from Catchment F and some runoff that may overtop into Catchment G.

Impact of Modeling of Four BWRX-300 Units

Additional modeling analysis showed with proper sizing and arrangement of additional conveyance and retention infrastructure in future site plans, the construction of additional three BWRX-300 units will not impact the functionality of the stormwater infrastructure protecting the first BWRX-300.

Proposed Flood Mitigation, Proofing, and Practice for DNNP

In Section 6 of the 2022 Flood Hazard Assessment (Reference 2.5-18), flood mitigation or flood proofing practices applicable to the DNNP as well as mitigation measures are proposed. Options for flood mitigations applicable to the DNNP site include:

- Constructing barriers to stop floodwater from entering the structure/site areas
- Constructing retention and detention ponds to slow and/or stop floodwaters entering the site area
- Wet Flood Proofing whereby floodwaters are allowed to enter the structure/site area, but ensuring that there is no or minimal damage to the building's structure/site and to its contents
- Emergency management/flood forecasting.

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Summary of Flood Hazard for the DNNP Site

Table 6.2-2 in the 2022 Flood Hazard Assessment (Reference 2.5-18) summarizes the primary source of flood hazards for the DNNP site due to runoff. In essence, the flood hazards would be to backwatering and flooding of various sub-catchments causing overtopping of the receiving catchments or overloading the existing stormwater management infrastructure. Proposed mitigation includes measures such as:

- Increase the size of specific culverts draining into specific sub-catchments
- Increase the storage capacity of one or more stormwater management ponds
- Route runoff from specific catchments into other specific catchments
- Ensure progressing designs have sufficient conveyance and detention capacity and the stormwater infrastructure is adequate.

Per Subsection 5.4.1 of the 2022 Flood Hazard Assessment NK054-REP-01730-001 (Reference 2.5-18), the PMF, mentioned in Subsection 2.5.3.1, includes a design basis flood (involving a PMP and zero infiltration) concurrent with disabled sewer and drainage systems due to, for example, debris. Therefore, the flooding due to runoff can be screened out based on screening criterion [QL2]. The PMF assessment is the bounding assessment that includes the impacts of potential runoffs.

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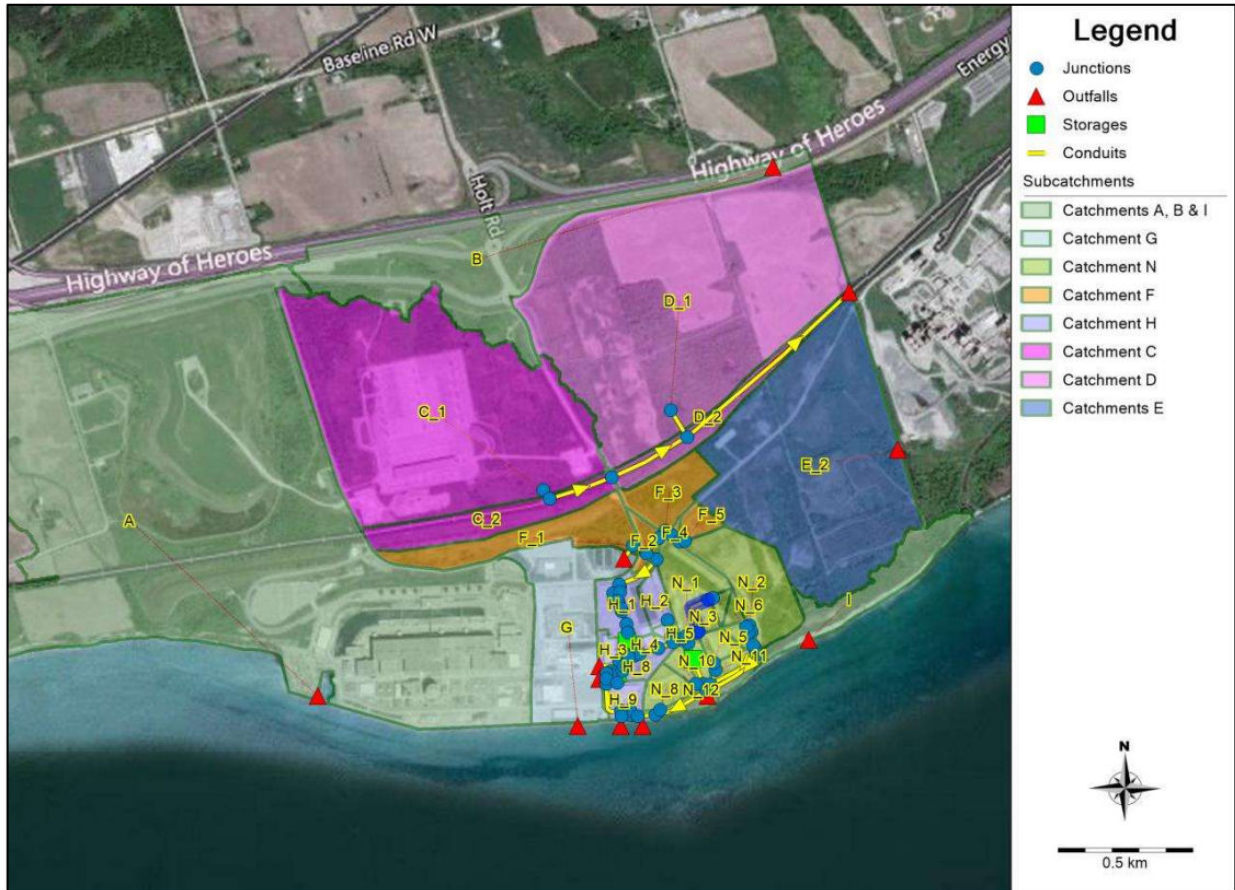


Figure 2.5.3.2-1 Pre-development Darlington Nuclear Site Drainage (Reference 2.5-18)

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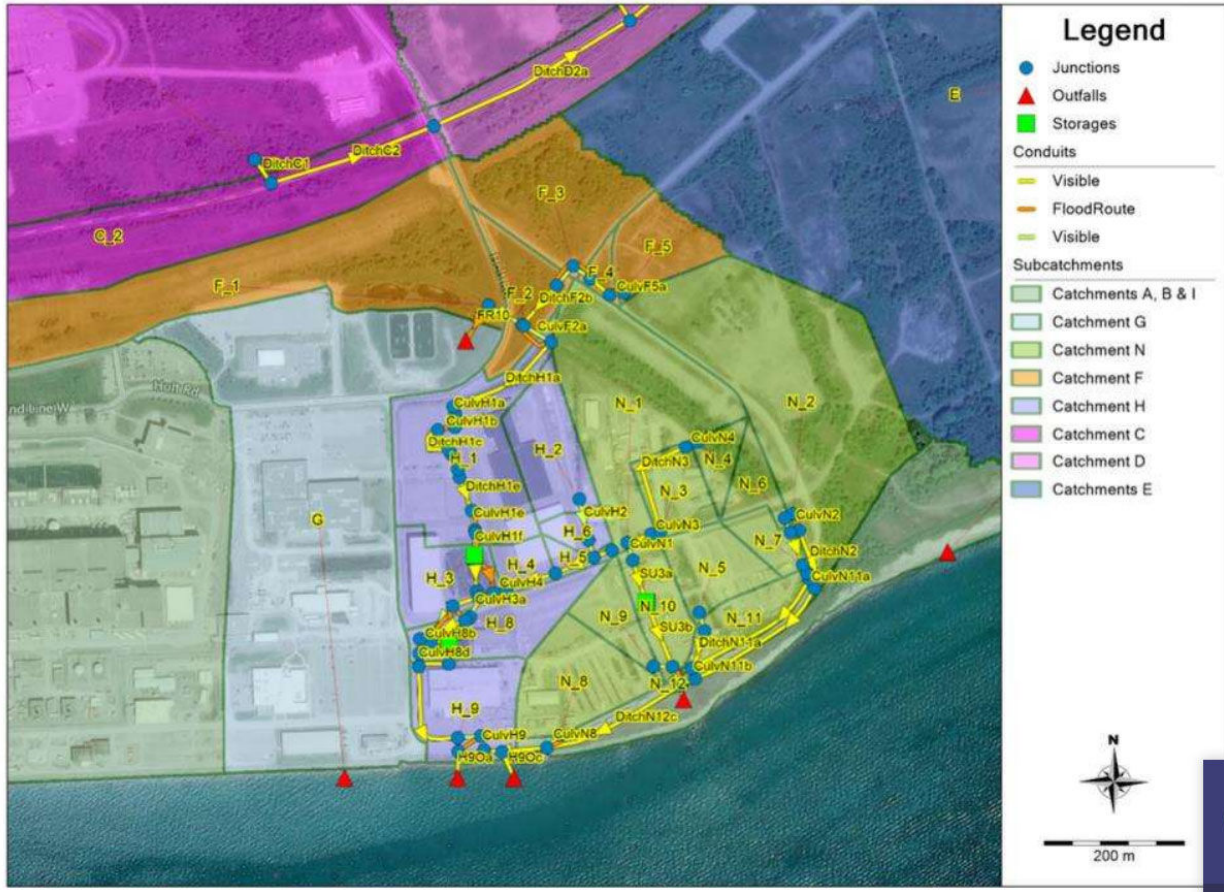


Figure 2.5.3.2-2 Post-development Darlington Nuclear Site Drainage (Reference 2.5-18)

2.5.3.3 Flooding Due to Rivers

Section 3.1 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) names two riverine systems within the local regional drainage basin: Tooley Creek and Darlington Creek.

The distance, infrastructure, and topography between the Tooley Creek watercourse and the proposed DNNP site precludes Tooley Creek as the source of a flood hazard.

Regarding Darlington Creek, the Central Lake Ontario Conservation Authority indicated there is not any history of severe flooding along Darlington Creek within the recorded history of the area. Figure 3.1-6 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) illustrates the current regulatory and 100-year recurrence floods inundation limits and shows that the inundation limits associated with these events do not represent a flood hazard to the DNNP site.

Subsection 5.4.2 in the 2022 Flood Hazard Assessment, NK054-REP-02730-00001 (Reference 2.5-18), describes comprehensive hydrologic and hydraulic models that are used to estimate drainage for the Darlington Creek watershed and its associated 14 sub-watersheds under 100-year recurrence PMP conditions, as replicated in Table 2.5-2. The models considered parameters such as length and slopes of the feeding reaches, time of concentration, storage coefficient, and future 100-year timeframe land use and development. The modelled Darlington Creek flood water elevations under PMP conditions in the future is estimated at 88.5 CGD at a stream gauge cross-section located just south of Highway 401. This is above the DNNP site construction grade of 88 CGD. However, to overtop into the DNNP site, flood waters would have to surpass the lowest elevation along the boundary separating the DNNP site from Darlington Creek, which is 95 CGD. Therefore, no external flood hazard to the DNNP site has been identified from Darlington Creek.

Thus, flooding due to the Tooley Creek and Darlington Creek is screened out for the DNNP site.

Table 2.5-2 Key Modelling and Assessment Parameters for Darlington Creek and On-site External Flood Hazards (Reference 2.5-18)

Parameter	Darlington Creek	On-site
Design Storm(s)	2.5-Hour 1:100-Year Storm (4 mm)	12-Hour PMP (420 mm)
	6-Hour PMP (405 mm)	
	12-Hour PMP (420 mm)	
Land Cover	Existing and Future Conditions	Zero infiltration
Threshold Water Level Constituting Flood Hazard	Above 95 m CGD	Above 88 m CGD

2.5.3.4 Flooding Due to Waves

The potential for flooding due to waves is discussed in Section 5.3 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2):

1. Subsection 5.3.1 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) describes the data and models used to assess the flooding hazard by waves, including the Lake Ontario wind and wave hindcast developed by the Wave Information Studies of the Office, Chief of Engineers, U.S. Army Corps of Engineers. The Simulating Waves Nearshore model was used to propagate extreme wave conditions from a selected 'offshore' wave information studies node to the shoreline, using the SPLASH numerical model for calculations of wave uprush and wave overtopping on shoreline beaches and structures.

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2. Subsection 5.3.2 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) describes the wave hindcast extreme analysis and determines that it is appropriate to use the wave information studies #192 100-year H_s of 4.7 m with peak wave period T_p of 9.7 s as input from the SW (225° N) to wave propagation/overtopping models.
3. Subsection 5.3.3 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) describes the wave propagation modeling for two water level scenarios and two site layout scenarios.
4. Subsection 5.3.4 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) describes the wave uprush and overtopping estimates.

Based on these scenarios, the wave uprush estimates range from 3.5 m to 11.3 m, and wave overtopping estimates range from 0.015 to 0.591 $\text{m}^3/\text{s}/\text{m}$.

In the 2022 Flood Hazard Assessment in NK054-REP-02730-00001 (Reference 2.5-18), the calculated wave heights extreme values were based on the latest hindcast data from two stations closest to the DNNP for the period from January 1979 to January 2020. Using a specific fitted method, wave heights were calculated for selected return periods of 10, 50 and 100 years. Based on the results, it was recommended to use an updated design wave of 6.1 m from the SW (225° N) with peak wave period T_p of 9.2 s to account for a more conservative estimate of the wave flooding potential at the DNNP site.

2.5.3.5 Flooding Due to Storm Surge and Seiche

Storm surges may cause seiches, because as a storm moves past the lake, the wind and pressure are no longer pushing the water, therefore the piled-up water moves toward the other end of the lake. The water sloshes from one end of the lake to the other few times until the water level is returned to normal. This sloshing back and forth is called a seiche. Seiches can be created due to other meteorological effects, seismic activities, or also tsunamis.

Section 5.2 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2) describes the numerical hydrodynamic model of Lake Ontario which was developed to assess the potential for generation of storm surge and seiche response to extreme severe weather systems tracking through the region. The model was implemented on a bathymetric grid of Lake Ontario with a 2.7 km resolution.

The most severe types of weather systems in the region of Lake Ontario are:

1. Post Tropical Storms: A good example of a post tropical storm with very severe wind conditions for Lake Ontario was Hurricane Hazel (1954). A storm like Hazel would typically approach Lake Ontario from between the southeast and south. A Hazel-like post tropical storm with extremely severe characteristics could have sustained winds up to 100 km/h and a pressure drop as low as 95 kPa.
2. Alberta Clippers: They are compact fast moving winter storms with sustained winds up to about 80 km/h and a pressure drop of about 97 kPa. They would typically track from northwest to west-southwest.
3. Colorado Lows: They are less compact than the Alberta Clippers but have otherwise similar characteristics and would track from the southwest or south-southwest.
4. Gulf Lows: A good example of a very severe Gulf low is the Great Blizzard of 1978. The pressure dropped down to the extremely low value of 95.8 kPa. Characteristic severe sustained winds were up to about 100 km/h.

The parameters used to represent the four idealized storms listed above are shown in Table 5.2-1 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2). The highest predicted water level at Darlington Nuclear site resulting from surge or seiche is about 0.75 m. This level can be produced either directly as a surge by a storm of Hazel-type tracking from the south over the western end of the lake, or indirectly after an Alberta Clipper from the west builds up a large surge at the eastern end of the lake resulting in a seiche of large amplitude. The 2022 Flood Hazard Assessment in NK054-REP-02730-00001 (Reference 2.5-18) also recommended 0.75 m as the highest water level produced by storm surge or seiche, in concurrence with the value predicted in the 2009 NK054-REP-01210-00012 (Reference 2.5-2).

Table 4.2 of the 2019 Darlington Hazard Screening Analysis NK054-REP-03611-1004 (Reference 2.5-4) shows the margin between the lake level and the top of the breakwater works at Darlington Nuclear site. As the margins are larger than the 0.75 m highest water level resulting from surge or seiche, the potential flood impacts are screened out.

2.5.3.6 Flooding Due to Tsunami

As described in Section 5.7 of the 2009 NK054-REP-01210-00012 (Reference 2.5-2), tsunamis are long period gravity waves generated by seismic disturbances of the sea bottom or shore, or landslides resulting in a sudden displacement of the water surface with the resulting wave energy spreading outwards across the ocean or lake at high speed. An additional consideration is the potential for a tsunami to occur as a series of waves (rather than a single wave) with associated increased impact from cumulative damage or flooding effects.

Due to the geological stability of the Great Lakes region where the largest measured seismic activity results in only small earthquakes typically of magnitude 3 or 4, the 2009 flood hazard assessment NK054-REP-01210-00012 (Reference 2.5-2) concludes a tsunami in Lake Ontario is an improbable event for DNNP. This conclusion is confirmed in the 2022 Flood Hazard Assessment NK054-REP-02730-00001 (Reference 2.5-18).

2.5.3.7 Flooding Due to Ponds, Dams or Dikes

As noted in Subsection 4.4.7 of the 2019 NK054-REP-03611-10043 (Reference 2.5-4), there is no large permanent human-made water storage pond or dam near the Darlington Nuclear site that can threaten the site. Therefore, this potential flood mechanism is screened out. Per the 2020 NK054-CORR-00531-10533 (Reference 2.5-3), this conclusion is applicable to the DNNP site since it is encompassed by the Darlington Nuclear site. Subsection 5.5.1 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18) also concluded no hazard assessment for the failure of human-made structures such as dams or dikes is required for the DNNP site.

Any temporary ponds and body of water that could potentially be created during a severe storm (for example on the rail track, by the embankments, overflowing culverts) are addressed in the 2009 hydrological assessment NK054-REP-01210-00012 (Reference 2.5-2) and the 2022 Flood Hazard Assessment NK054-REP-02730-00001 (Reference 2.5-18) (refer to Subsection 2.5.3.2).

2.5.3.8 Flooding Due to Ice Jamming

As described in Subsection 4.4.8 of the 2019 NK054-REP-03611-10043 (Reference 2.5-4), this event scenario is concerned with late winter conditions when large ice blocks, accumulated over winter, melt rapidly as the weather temperature rises above the freezing point.

The 2014 DNGS hydrological assessment NK38-REP-03611-10094 (Reference 2.5-12) examined the worst-case scenarios and concluded that a summer PMP, with storm drains blocked, would bound winter PMP with snow covering the ground and ice blocking the drains. The event consequences of ice jamming at the lakeshore, and rapid melting of the accumulated

ice blocks may result in localized high water levels and flooding, but the consequences are not worse than the PMF assessed in the DNGS hydrological assessment.

Therefore, the hazard is screened out based on screening criterion [QL2], as both types of consequences (accumulation on the roof tops, and accumulation at the lakeshore) have consequences less severe than the events assessed in the 2014 DNGS hydrological assessment (Reference 2.5-12). This conclusion can be applicable to the DNNP site due to proximity to the DNGS site, per the 2020 NK054-CORR-00531-10533 (Reference 2.5-3).

The 2022 Flood Hazard Assessment in the 2022 NK054-REP-02730-00001 (Reference 2.5-18) states that in the DNNP site area, Lake Ontario freezing starts from the Bay of Quinte, east of the DNNP site. The ice then propagates eastward to the St. Lawrence River. As shown in Figure 5.6-2 of the 2022 Flood Hazard Assessment in the 2022 NK054-REP-02730-00001 (Reference 2.5-18), the ice coverage over Lake Ontario is about 17% by mid-February with an average of 10% coverage for the winter period. Ice breaking accelerates in early March. Thus, the DNNP site region of Lake Ontario is ice-free year-round, in an average year. This is mirrored in the fact that, on a weekly basis, between December 4 and May 14, the median ice concentrations near the DNNP site are 0%. Furthermore, Lake Ontario is the smallest Great Lake in terms of surface, but it is the second deepest and as such, has a large volume compared to its surface area, resulting in an exceptionally high heat storage capacity. Temperature changes occur at a much lower rate in Lake Ontario compared to the other Great Lakes.

Therefore, the 2022 Flood Hazard Assessment in the 2022 NK054-REP-02730-00001 (Reference 2.5-18) confirms that the flood hazard due to ice jamming is screened out based on the basis of screening criterion [QL3].

2.5.4 Potential Effects of Climate Change

The potential impacts of climate change are discussed and summarized in Subsections 7.2 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18), where Subsections 7.2.1 and 2.7.2 address the effect of climate change on temperature and precipitation.

The total annual precipitations are forecast to slightly increase (+3% to 10%) in 2071-2100 compared to present-day conditions. However, precipitations are expected to remain stagnant during summer, hence resulting in higher percentage increases for other seasons (from +2% to 21%) depending on the emission scenario chosen. Considering that temperature is also forecast to significantly increase during winter, more liquid precipitations are to be expected as well.

Maximum daily precipitations are expected to vary from -4% to +25% depending on season and emission scenario. The seasonal trend follows a similar pattern as total precipitations with stagnant conditions during summer (-4% to 0% compared to present-day conditions) in contrast to spring and winter for example (from +10% to 25%).

Although maximum daily precipitations should not increase by much during fall and especially summer, these seasons remain the period when this extreme weather event will occur. While the projected increase in daily 1:100-year return period precipitation is up to 10.7% by 2100 in the high greenhouse gas emissions scenarios, the PMP evaluated is not anticipated to be exceeded due to climate change, and no additional flood hazard is identified on account of climate change.

Subsection 5.1.2.5 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18) describes a 2014 plan that was adopted in 2017 to allow for control of extreme low or high water level conditions. Under the modelled conditions in the 2014 plan, the weekly mean water levels would never have exceeded 75.8 m. However, since the adoption of the new plan in 2017, water level exceeded the previous maximum on two occasions, in 2017 and 2019. Climate change was identified as the probable cause of these maximum water levels.

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Subsection 7.2.5 of the 2022 NK054-REP-02730-00001 (Reference 2.5-18) discusses the impact of climate change on Lake Ontario water levels. Lake Ontario water levels are primarily controlled by variations in precipitation, runoff, and evaporation over the watershed. Climate change influences these parameters that control lake water level fluctuations. Climate change would contribute to increasing low and high extremes in Lake Ontario water levels. Anticipated increases in precipitation would contribute to high Lake Ontario water levels. The report recommends higher lake levels experienced recently in 2017 and 2019 should be considered as appropriate design lake levels for shoreline assessment and design bases.

According to the 2019 IAEA Site Evaluation for Nuclear Installations Safety Requirements for Flood Hazard (Reference 2.5-14), the reference water level upon which the computed surge or seiche is superimposed should be selected to have a sufficiently low probability of being exceeded. Usually the 100-year recurrence monthly average high water is adopted or, if the water level is controlled, the maximum controlled water level is used. However, the International Joint Commission Lake Ontario 2021 plan (Reference 2.5-13) allows deviations, so that no maximum level is set, and a stochastic approach is still necessary. In this case the controlled water level with a probability of exceedance of 1% is 75.6 m; however, the highest level during a century is about 76.6 m. In addition, measured water levels at Cobourg have exceeded 75.6 m for duration of about three months in 1973.

Therefore, 75.6 m may be a low estimate, and 76.6 m should be used, which is close to the maximum found in the historic data and greater than the 100-year recurrence level. This level assumes the International Joint Commission Lake Ontario continues with the current water level control plan.

The 2023 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts, NK054-PLAN-07007-00001 (Reference 2.5-20) is developed to address the potential impact of climate change on external hydrological and meteorological hazards. The strategy summarizes life cycle considerations including long-term monitoring (Subsection 2.11.9) and describes the plan to ensure the BWRX-300 facility is resilient to climate change as a potential external hazard.

The 2023 NK054-REP-07007-1049426 DNNP Hazard Bounding Analysis (Reference 2.5-22) presents a bounding analysis of climate change impacts and establishes probable extreme values for climate hazards where feasible. The 2022 NK054-REP-07007-1028871 DNNP Gradual Climate Change and Natural Hazard Identification (Reference 2.5-23) describes the process used in identifying a comprehensive list of natural external events for DNNP, which are screened for climate change impact for evaluation against the DNNP BWRX-300 design basis.

2.5.5 Groundwater

Relevant to the assessment of radioactive material transported through the groundwater system and potentially dispersed in the environment, the following subsections discuss the characterization of the hydrogeological subsurface properties as well as relevant monitoring programs.

The in-situ soil properties are derived based on existing subsurface investigations completed at the DNNP site and in the vicinity of the BWRX-300 SMR location, as described in Subsection 2.7.3.2.4.

2.5.5.1 Groundwater Conditions

The groundwater conditions are described in detail in Subsection 2.7.3.2.4. Groundwater flow maps are available in Section 2.7, Figures 2.7.3.2-3 to 2.7.3.2-9. In general, groundwater on the site flows from north to south, and discharges toward Lake Ontario, as confirmed in the 2022 DNNP Phase 1 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.5-21).

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The predominant groundwater flow patterns reported in the 2022 geotechnical investigation NK054-REP-01210-00175 (Reference 2.5-21) remain unchanged from the historical interpretations of groundwater flow conditions documented in the 2009 NK054-REP-01210-00011 (Reference 2.5-15) and the 2009 NK054-REP-07730-00005 (Reference 2.5-16).

Relevant information is provided in Subsection 2.8.2.2 on the impact of hydrogeological conditions on the dispersion of radioactive material.

2.5.5.2 Groundwater Level

Based on the groundwater conditions at the DNNP site presented in Subsection 2.7.3.2.4 and Table 2.7-11, groundwater is anticipated to be present approximately between elevation 80 m to 86 m corresponding to depths between about 2 m and 8 m below the plant grade at elevation 88 m. (refer to Subsection 2.7.5.2.6)

2.5.5.3 Groundwater Monitoring

The OPG Environmental Monitoring Program (EMP) N-REP-03443-10027 (Reference 2.5-17) examines the chemical, radiological, and physical characteristics of the groundwater beneath the Darlington Nuclear site. The groundwater monitoring wells are located in key areas of the Darlington Nuclear site including the protected areas (near the RBs), controlled areas (farther away from the RBs but within the fence), and the Darlington Nuclear site perimeter. Wells on DNNP site are considered site perimeter wells (refer to the NK38-REP-10140-10032 (Reference 2.5-8)).

2.5.6 Surface Water

As related to the assessment of radioactive material transported through the surface water system and potentially dispersed in the environment, this subsection discusses the characterization of the surface water properties in Subsection 2.5.6.1, as well as the relevant monitoring programs in Subsection 2.5.6.2.

2.5.6.1 Surface Water Properties

The pertinent properties of the surface water (i.e., Lake Ontario) are described below:

1. Lake-Wide Circulation

The Darlington Nuclear site is situated on the northern shore of Lake Ontario where the lake-wide circulation is generally eastward from the Niagara River to the discharge to the St. Lawrence River, per the 2021 D-REP-07701-00001 (Reference 2.5-9). Water movement near the site is predominantly along the shore, occurring for 73% of the time (35% to the west and 38% to the east), as described in the 2012 NK054-REP-01210-00016 (Reference 2.5-5). Onshore and offshore movement occurs about 15% of the time, as reported in the 2012 NK054-REP-01210-00016 (Reference 2.5-5). Table 2.7 in the 2021 D-REP-07701-00001 (Reference 2.5-9) shows the frequency of lake current flowing toward each direction and the maximum speed that occurred in each direction, per the 2021D-REP-07701-00001 (Reference 2.5-9). Table 2.5-3 shows the averaged lake current direction and speeds.

**Table 2.5-3 Summary of Lake Ontario Depth Averaged current speed and direction
(Reference 2.5-9)**

Month	Direction	Depth Averaged Speed All Directions	Depth Averaged Speed Easterly	Depth Averaged Speed Westerly
	Degree from North	cm/s	cm/s	cm/s
January	142	17.5	20.6	12.4
February	145	16.2	18.9	13.1
March	159	13.5	15.5	12.7
April	165	11.8	12.7	12.3
May	181	9.4	12.0	7.8
June	177	9.5	10.5	9.7
July	183	13.3	16.0	11.4
August	193	10.9	12.2	11.1
September	196	9.9	10.3	10.9
October	170	11.8	13.0	11.9
November	159	11.5	13.2	9.8
December	169	12.9	14.4	12.5
Annual Average		12.4	14.1	11.3

2. Lake Water Temperature

Lake Ontario is classified as a dimictic lake because it undergoes a complete cycle of isothermal and vertically stratified conditions every year. The thermal structure depends on the season because of large annual variation in surface heat fluxes. Lake-wide surface temperatures typically range from freezing in winter to about 20 °C in summer, per the 2021 D-REP-07701-00001 (Reference 2.5-9). Statistical summary of ambient water temperatures near Darlington Nuclear site (from 1984 to 1996 and 2011 and 2012) is provided in Table 2-9 of the 2021 D-REP-07701-00001 (Reference 2.5-9).

3. Ice Conditions

Ice formation in winter is typically limited to the nearshore areas at the eastern end of the lake within the Kingston Basin, per the 2021 D-REP-07701-00001 (Reference 2.5-9) and the 2022 NK054-REP-02730-00001 (Reference 2.5-18).

2.5.6.2 Surface Water Monitoring

As described in Subsection 3.2.2 of the 2019 NK38-OM-61100 (Reference 2.5-6), the Lake Current Monitoring system is a real-time current profile measurement system to be used in the event of a radiological liquid emission. Further details of the radiological baseline conditions of lake water at the Darlington Nuclear site are provided in Subsection 2.9.1.1.

The OPG EMP N-REP-03443-10027 (Reference 2.5-17) identifies the contaminants and physical stressors to be monitored in the environment surrounding the site. Locations considered to be outside the influence of site operations are also monitored to allow for a baseline comparison with

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background values. This includes monitoring and sampling of lake water, municipal drinking water, and other means of aquatic sampling. Further details on the EMP are provided in Chapter 20, Subsection 20.11.2.

2.5.7 References

- 2.5-1 NK054-REP-01210-00018 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington - Additional Considerations," Ontario Power Generation.
- 2.5-2 NK054-REP-01210-00012 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington - Part 5: Flood Hazard Assessment," Ontario Power Generation.
- 2.5-3 NK054-CORR-00531-10533, 2020, "Application for Renewal of OPG's Darlington New Nuclear Project (DNNP) Nuclear Power Reactor Site Preparation Licence (PRSL)," Ontario Power Generation.
- 2.5-4 NK054-REP-03611-10043 R003, 2019, "Hazard Screening Analysis – Darlington," Ontario Power Generation.
- 2.5-5 NK054-REP-01210-00016 R002, 2012, "Site Evaluation of the OPG New Nuclear at Darlington - Part 2: Dispersion of Radioactive Materials in Air and Water," Ontario Power Generation.
- 2.5-6 NK38-OM-61100 R013, 2019, "Environmental Monitoring – Air and Water," Ontario Power Generation.
- 2.5-7 NK054-REP-01210-00108 R000, 2019, "DNNP – Site Preparation Nuclear Safety Licence Renewal Activity Report," Ontario Power Generation.
- 2.5-8 NK38-REP-10140-10032 R000, "Darlington Nuclear Groundwater Monitoring Program Results," Ontario Power Generation.
- 2.5-9 D-REP-07701-00001 R001, 2021, "Environmental Risk Assessment for the Darlington Nuclear Site," Ontario Power Generation.
- 2.5-10 IAEA Safety Standards No. SSG-18, 2011, "Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations," International Atomic Energy Agency.
- 2.5-11 "Lakes, and Rivers Improvement Active Technical Guidelines Administrative Guide," 2017, Ministry of Natural Resources and Forestry.
- 2.5-12 NK38-REP-03611-10094 R000, 2014, "Darlington Nuclear Generating Station Hydrological Assessment," Ontario Power Generation.
- 2.5-13 International Joint Commission Lake Ontario, "St. Lawrence River Water Levels, June 2021," <https://ijc.org/en/loslrb/watershed/water-levels>.
- 2.5-14 IAEA Safety Standards Series No. SSR-1, 2019, "Site Evaluation for Nuclear Installations Safety Requirements," International Atomic Energy Agency.
- 2.5-15 NK054-REP-01210-00011 R001, 2009, "Site Evaluation of The OPG New Nuclear at Darlington - Part 6: Evaluation of Geotechnical Aspects," Ontario Power Generation.
- 2.5-16 NK054-REP-07730-00005 Rev. R000, 2009, "Geological and Hydrogeological Environment, Existing Environmental Conditions, Technical Support Document, New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.
- 2.5-17 N-REP-03443-10027 R000, 2021, "Results of Environmental Monitoring Programs," Ontario Power Generation.

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- 2.5-18 NK054-REP-02730-00001, 2022, "Flood Hazard Assessment," Ontario Power Generation.
- 2.5-19 NK054-REP-02730-00002, 2022, "PMP Validation," Ontario Power Generation.
- 2.5-20 NK054-PLAN-07007-00001, 2023, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts," Ontario Power Generation.
- 2.2-22 NK054-REP-01210-00175 R000, 2022, "Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project", Volume 2 of 2 "Geotechnical Interpretation of Design Parameters," Ontario Power Generation
- 2.5-22 NK054-REP-07007-1049426 R001, 2023 "Darlington New Nuclear Project – Hazard Bounding Analysis," Ontario Power Generation
- 2.5-23 NK054-REP-07007-1028871 R000, 2022 "Darlington New Nuclear Project – Gradual Climate Change and Natural Hazard Identification," Ontario Power Generation

2.6 Meteorology

2.6.1 Introduction

Section 2.6 describes the meteorological aspects relevant to the DNNP site based on the consideration of the local climatic effects. Details are included in Section 2.6 on the characterization of extreme values of meteorological events in relation to potential hazards to the BWRX-300 facility, as well as in relation to the transportation of radioactive materials and the dispersion of radionuclides with the potential to impact the DNNP site. The meteorological characteristics and conditions included in the following list are assessed in relation to the design and the evolution of extreme parameters over the lifetime of DNNP BWRX-300:

- Temperature (Subsection 2.6.2)
- Humidity (Subsection 2.6.3)
- Rainfall (Subsection 2.6.4)
- Wind Speed (Subsection 2.6.5)
- Tornadoes and Hurricanes (Subsection 2.6.6)
- Waterspouts (Subsection 2.6.7)
- Dust Storms and Sandstorms (Subsection 2.6.8)
- Snow Load and Ice Load, Freezing Rain, and Ice Storm (Subsection 2.6.9)
- Lightning (Subsection 2.6.10)
- Windborne Debris (Subsection 2.6.11)
- Climate Change (Subsection 2.6.12)

Key metrological characteristics and parameters relevant to the DNNP site and the surrounding area are listed in Table 2.6-1. The list includes characteristics such as temperature, humidity, precipitation, high wind, tornadoes, snowfalls, lightning, and climate change impact.

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Table 2.6-1: Meteorological Characteristics Summary of DNNP Site and Surrounding Area

Characteristic	Value/Description				
2.6.2 Temperature	Highest ever recorded	Toronto Bowmanville		36 °C 40.6 °C	
	Extreme minimum	-40 °C, with annual degree-days below 18 °C of 4130 degree-days			
	Maximum	Dry bulb 37 °C		Wet bulb 23 °C	
	Design Basis Duration at low Temperature		Temperature	Duration	
			-40 °C	1 h	
			-35 °C	5 h	
			-30 °C	10 h	
			-25 °C	20 h	
			-20 °C	70 h	
	-15 °C	150 h			
Safety Class 1 SSC Design Conditions	Highest 40 °C		Lowest -40 °C		
Impact of extreme temperatures	Mist and white frost during winter		Heatwaves during summer		
Impact of Climate Change by 2100	Increase between 2 °C and 5 °C (References 2.6-3 and 2.6-4) Recent analysis: increase by up to 7.2 °C (Reference 2.6-17)				
2.6.3 Humidity	Lowest	During winter and air is quite dry due to Arctic air from the north			
	Highest	During summer and fall due to the air from the Gulf of Mexico.			
	Mean value	65 to 80% throughout the year			
	Design Conditions	No indication of extreme conditions that require design mitigation			
2.6.4 Rainfall / Precipitation	Mean annual	Oshawa 877.9 mm		Toronto 800 mm	
	Maximum daily	Oshawa 88.6 mm		Toronto 79.3 mm	
	Average (DNGS PO-SAR)	145 days/yr, with of 800 mm average, with 20% due to snowfall			
	Greatest per day	In Oshawa, 144.8 mm			
	PMP (vicinity of DNNP)	420 mm in 12-hours, with 51% in the 6 th hour, for a watershed area of < 1295 km ²			
	Severe flooding	PMP conditions, combined with a 1 in 100-year recurrence lake level high, and storm surge			
	Extreme Daily	Unlikely to exceed the PMP value in a 100-year recurrence for DNNP			
	For roof design	16 mm in 5 min – 50-year, 5-minute storm 25 mm in 15 min - 50-year return, 15-Minute storm 47 mm in 60 min – 50-year return 1-hour storm 210 mm in 24 h – Regional storm (Hurricane Hazel)			
	Climate Change Impact by 2100	Increase in heaviest precipitation intensity and frequency of 12% and 22%, respectively. Plausible increase in extreme precipitation amount over southern Ontario by 14% (7 mm) (Reference 2.6-3).			

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Table 2.6-1: Meteorological Characteristics Summary of DNNP Site and Surrounding Area

Characteristic	Value/Description	
		Recent analysis indicates total precipitation and maximum 24-hour re anticipated to increase by up to 25% (Reference 2.6-17). The 12-hour PMP of 420 mm remains bounding of this increase as the summer and fall projections (when PMP would occur) are lower, up to +10%, and the PMP value is conservative (Reference 2.6-18). Such predicted changes is to be considered in the design and monitored for long term as discussed in Subsection 2.11.9.
2.6.5 Wind and Wind Speed	Typical	The prevailing winds were from the north-westerly quarter (10.38% of the time) and from the west quarter (9.98% of the time) (Refer to Subsection 2.8.1.3)
	Average and Clam	Approximately 2.4 m/s (~8.6 km/h) and less than 2 m/s (~7.2 km/h), respectively at 10 m level (Refer to Subsection 2.8.1.3)
	Maximum	64 km/h at 10 m level and 80 km/h at 50 m level (for a 100-year return period)
	Wind 3-sec Gust	Extreme gusts – Occur mostly in the West, Southwest, and Northwest directions Speeds exceeding 120 km/h are rare Higher speeds of up to 174.4 km/h occurred in some instances
	Climate Change Impact by 2100	Wind speeds are expected to change due to climate change. Decline in average wind speed over the years in a warmer world
2.6.6 Tornadoes and Hurricanes	Maximum Pressure Drop	6.3 kPa
	Maximum Rotational Speed	257.4 km/h
	Maximum Transitional Speed	64.4 km/h
	Maximum Wind Speed	321.8 km/h (Upper limit - Enhanced Fujita scale 4 (EF-4) tornado)
	Radius of Maximum Rotational Speed	45.7 m
	Rate of Pressure Drop	2.5 kPa/s
	Design Basis – Tornado Missile Spectrum types	Schedule 40 pipe, Automobile 5 m x 2 m x 1.3 m, and Solid Steel Sphere (Refer to Table 2.6-6)
	Hurricanes, Cyclones, Tropical Storms, Tropical Depression	Very low probability of an actual hurricane directly impacting the DNNP site, and it describes the probable maximum tropical cyclone as unlikely to yield gusts of more than 100 km/h - lower than that of the design basis tornado. As such, wind hazard from a hurricane is not considered further.
2.6.7 Waterspouts	A tornado that forms over water that are rarely reported. Covered by the design basis tornado	
2.6.8 Dust and Sandstorms	Not identified as phenomena for southern Ontario, and as such are not identified as potential hazards for DNNP.	
	Average daily snowfall	3 cm to 5 cm from December to March

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Table 2.6-1: Meteorological Characteristics Summary of DNNP Site and Surrounding Area

Characteristic	Value/Description	
2.6.9 Snow and Ice Load, Freezing Rain, and Ice Storm	Highest Daily snowpack	Mean value of 8.6 cm in January
	Darlington Nuclear site characteristic Value	2.2 kPa
	Combined snow load and winter PMP event	1.80 kPa for 50-year recurrence 1.71 kPa for 100-year recurrence, without Winter PMP
	Freezing Rain	Screened out due to low frequency
	Ice Storm	This issue is resolved as part of Pressure Increase Group (refer to Subsection 2.2.8).
2.6.10 Lightning	Frequency	2 to 3 cloud-to-ground flashes per year per square km, causing induced fires and electromagnetic compatibility. Screened out due to low hazard to the site.
2.6.11 Windborne Debris	Wind-propelled missiles are similar to tornado missiles which is assessed as part of the high wind PSA.	
2.6.12 Climate Change Impact	Impact of climate change is considered in the 2023 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts NK054-REP-07007-00001 (Reference 2.6-19) which summarizes life cycle considerations including long-term monitoring, described in Subsection 2.11.9	

2.6.2 Temperature

Since DNNP is in proximity to DNGS within the Darlington Nuclear Site, similar meteorological conditions are expected. The highest temperatures ever recorded at Bowmanville, and Toronto are 36 °C and 40.6 °C, respectively per Subsection 2.1.1 of the 2019 NK054-REP-01210-00108 (Reference 2.6-2). As shown in Table 2-1 of Part 2 of the 2018 NK38-SR-03500-10001 DNGS Safety Report (Reference 2.6-7), the extreme minimum temperature chosen for DNGS was -40°C, with annual degree-days below 18 °C of 4130 degree- days. Per Subsection B.8.4. Table 3 of the 2010 N-REP-01200-10000 (Reference 2.6-9), the Darlington Nuclear site characteristic value for maximum dry bulb temperature is 37°C, and the maximum wet bulb temperature is 23°C. The design basis durations at low temperature for DNGS site in the 2018 NK38-SR-03500-10001 (Reference 2.6-7), which are applicable to the DNNP site, are listed in Table 2.6-2.

Table 2.6-2: DNGS Design Basis Durations at Low Temperature Applicable to DNNP

Temperature	Duration
-40°C	1 h
-35°C	5 h
-30°C	10 h
-25°C	20 h
-20°C	70 h
-15°C	150 h

According to Subsection 4.5.1 of the 2012 NK054-REP-01210-00016 (Reference 2.6-6), Safety Class 1 (SC1) SSCs that are exposed to ambient environment conditions in DNGS are designed for extreme temperatures of -40 °C during the winter and +40 °C during the summer. The design temperature for the DNNP SSCs is -40 °C, while the design temperature of +40 °C is approximately the same as the highest recorded temperature of 40.6 °C as baseline data on extreme conditions. Although the HVAC system efficiency is generally reduced due to extreme high temperature conditions, the system is expected to provide sufficient cooling to maintain design limits for equipment rooms and to support control rooms habitability. This information is also relevant to DNNP SSCs which require the implementation of appropriate mitigating measures, as necessary.

Refer to Chapter 9A, Section 9A.5 for information on the functions, design bases, description, maintenance, performance, and safety evaluation of the BWRX-300 HVAC systems.

Furthermore, global climate models projected in 2009 an increase of the temperatures in southern Ontario of between 2 °C and 5 °C over the next century, due to rising greenhouse gas emissions, as indicated in Subsection 7.2.8 of the 2009 NK054-REP-01210-00012 (Reference 2.6-3). This information is in line with the contents in Subsection 4.1.2.2 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4), which stated temperatures in the vicinity of DNNP site were expected to rise by 2 °C in 2040 and by as much as 5 °C in 2100 during winter and summer months. In the 2022 NK054-REP-02730-00001 Flood Hazard Assessment (Reference 2.6-17), Subsection 7.2.3 indicates temperatures at the DNNP site are anticipated to increase by up to 7.2 °C by 2100. Mitigation of these environmental changes is to be provided at DNNP. Subsection 2.11.9 describes the long-term monitoring of parameters susceptible to be impacted by climate change,

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as mentioned in the 2023 DNNP Strategy for Addressing Climate Change Impacts, NK054-PLAN-07007-00001 (Reference 2.5-20).

The extreme temperatures expected in the vicinity of DNNP site have the potential to result in mist and white frosts during winter, and heatwaves during summer, per Subsection 4.5.1 of the 2019 NK38-REP-03611-10043 (Reference 2.6-1). In the event of extremely high temperature conditions, an extended heatwave could lead to a high demand on the transmission lines, which could potentially cause a loss of grid condition.

Temperatures Normals at and near the Darlington Nuclear site are described in Subsection 2.8.1.1, as related to the meteorological impact on the dispersion on radioactive material.

2.6.3 Humidity

The 2009 Site Evaluation of Meteorological Events NK054-REP-01210-00013 (Reference 2.6-4) states the average relative humidity in the vicinity of DNNP is the lowest during winter, as the air is quite dry due to the Arctic air moving down from the north; the highest humidity values occur during summer and fall as the humid air from the Gulf of Mexico moves across southern Ontario.

Currently, humidity values are not recorded on-site by the meteorological tower as indicated in Subsection 2.2.1 of the 2012 NK054-REP-01210-00016 (Reference 2.6-6). However, this information is available from several Environment Canada stations such as Oshawa WPCP and Toronto Island. Based on the available data, the mean relative humidity ranges from 65 to 80% throughout the year, per Section 2.2 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5). Section 3.11 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5) also states the meteorological values evaluated with respect to humidity show no indications of extreme conditions requiring design mitigation. Based on Subsection 4.5.2 of the 2020 NK054-CORR-00531-10533 (Reference 2.6-8), no further evaluation is required on the impact of humidity, as the design of the BWX-300 facility is expected to fit within the Plant Parameter Envelope (PPE) values per commitment D-C-3 in the 2021 NK054-REP-01210-00078 DNNP Commitments Report (Reference 2.6-10).

2.6.4 Rainfall / Precipitation

The Bowmanville Mosert climate station is the closest to the Darlington Nuclear site. The Precipitation Normals (from 1981 to 2010) are described in Subsection 2.8.1.2, where the monthly averages and daily extremes (for precipitation (mm), rain (mm), and snow (cm)) are listed in Table 2.8-3.

The concept of PMP is defined in the 2009 NK054-REP-01210-00012 (Reference 2.6-3) as the greatest depth of precipitation possible for a given storm area at a particular location and time of the year (refer also to Subsection 2.5.3.1 for details on PMP and PMF definitions and values). According to Section 4.1 of the 2019 NK38-REP-03611-10043 (Reference 2.6-1), the PMP for watershed areas less than 1295 km² in the vicinity of DNNP site has been estimated as a 12-hour precipitation equivalent to 420 mm of total rainfall (with 51% in the 6th hour). Hence, based on the maximum daily precipitation predicted in Subsection 3.4.3 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4) using data from the monitoring stations in Toronto Island and Oshawa (79.3 mm and 88.6 mm, respectively), it is unlikely for extreme daily precipitations to exceed the 420 mm PMP value in a 100-year period for DNGS. This conclusion, which is also applicable to DNNP given its proximity to DNGS, is confirmed in the 2022 DNNP Flood Hazard Assessment in NK054-REP-02730-00001 (Reference 2.6-17).

Precipitation, along with other meteorological factors such as wind direction and speed, influence dispersion and, in case of precipitation, especially deposition. Radioactive materials tend to flow toward low-pressure systems and rainfall often occurs around those systems. Having the PMP

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value for the DNGS site available for the DNNP PPE ensures that this (maximum probable) value is considered in the DNNP's dispersion (and deposition) models. Models/codes (such as ADDAM and PAVAN) would use the precipitation rate as input to wet deposition. Precipitation Normals at and near the Darlington Nuclear site are described in Subsection 2.8.1.2, as related to the meteorological impact on the dispersion on radioactive material.

According to Table 3-1 in the 2022 PMP Validation reported in NK054-REP-02730-00002 (Reference 2.6-18), the DNNP storm values to be considered as part of roof design are as follows:

- 210 mm in 24 h – Regional storm (Hurricane Hazel)
- 47 mm in 60 min – 50-year return 1-hour rainfall
- 25 mm in 15 min - 50-year return 15-Minute storm
- 16 mm in 5 min – 50-year 5-minute storm

In relation to the changes in precipitation over time, few studies have examined changes in precipitation over Canada. The 2009 site evaluation report on flood hazard assessment, NK054-REP-01210-00012 (Reference 2.6-3) provides references to a number of studies in Subsection 7.2.1. Based on the conclusions in this report, the heaviest precipitation events are becoming more frequent during the spring and summer, and less frequent during the winter. The information provided indicates a reported increase in extreme precipitation intensity and frequency of 12% and 22%, respectively. In addition, Subsection 7.2.8 of the 2009 NK054-REP-01210-00012 (Reference 2.6-3) states some models show a plausible increase in the amount of precipitation for the most extreme precipitation events over southern Ontario by 14% (7 mm). In the 2022 NK054-REP-02730-00001 DNNP Flood Hazard Assessment (Reference 2.6-17), Subsection 7.2.3 indicates the total precipitation and the maximum 24-hour for certain seasons to increase by up to 25% by 2100. The PMP event is not coincident with this increase and remains conservative considering anticipated coincident increases. Consequently, no additional flood hazard is considered for rainfall increase due to climate change. However, as discussed in Subsection 2.11.9, long-term monitoring of precipitation is included as part of the 2023 DNNP Strategy for Addressing Climate Change Impacts NK054-PLAN-07007-00001 (Reference 2.6-19).

2.6.5 Wind and Wind Speed

Wind data sets at a standard height of 10 m are collected from Darlington Nuclear site meteorological tower as well as from nearby monitoring stations. The Darlington Nuclear site average and calm wind speeds and wind direction data are presented in Subsection 2.8.1.3. The maximum wind speed at 10 m level and 50 m level at Darlington Nuclear site was estimated to be 64 km/h and 80 km/h, respectively, for a 100-year return period, per the 2009 NK054-REP-01210-00013 (Reference 2.6-4).

Wind gust analysis is performed in the 2022 NK0054-REP-02730-00003 (Reference 2.6-14) for the DNNP site. Although wind speed was collected at the DNGS for 12 years at 15-minutes intervals, 3-second wind gust data were not available. In the 2022 NK0054-REP-02730-00003 (Reference 2.6-14) high-quality Government of Canada publicly available 3-second wind gust data were used from four different stations located within 100 km from DNNP: the three airports in Toronto, Peterborough, and Trenton, as well as the Toronto City Centre. Wind roses were used to analyze the gust magnitude and frequency for each station in eight gust directions. Annual Maximum Series data were then extracted and statistically tested and analyzed. Based on the summary of the maximum and mean of gust Annual Maximum Series, extreme gusts were found to occur mostly in the West, Southwest, and Northwest directions.

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To estimate the extreme design gust speeds for various return periods, the Extreme Value Type I model (known as Gumbel distribution model) was fitted to the extracted Annual Maximum Series values, as described in the 2022 NK0054-REP-02730-00003 Wind Gust Analysis (Reference 2.6-14). The extreme design gust speeds were then calculated for various return periods, particularly, for the design of reactor buildings based on ASCE7 IV risk category which corresponds to 3000-year return period. Other commonly used values corresponding 300-, 700-, and 1700-year return periods were also estimated. Finally, Inverse Distance Weighted interpolation technique was applied to transfer the estimated 3-second gust values from the four selected stations to the DNNP site; the results are listed in Table 2.6-3. Also, bounding envelop 3-second gust extreme values were computed for the DNNP site, as listed in Table 2.6-4. The envelop values are found to be on average 6% higher than the values estimated through interpolation for the DNNP site. Hence, for the design to be conservative, the 2022 NK0054-REP-02730-00003 Wind Gust Analysis (Reference 2.6-14) recommends using the envelop values.

Table 3-5 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5) presents the historical data available for wind gusts in the nearby area to the Darlington Nuclear site. Similar to the methodology used in Subsection 3.4.2 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4) and the 1990 N-REP-NGD-IR-61100-0002 (Reference 2.6-11), site-specific 3-second gust wind speeds of more than 120 km/h or more are rare. However, 3-second gust wind speeds have occurred in some instances with a maximum historical wind gust in the area of 154 km/h. This is consistent with Table 4-4 of the 2022 NK0054-REP-02730-00003 (Reference 2.6-14), as presented in Table 2.6-3, noting maximum speeds of up to 174.4 km/h occurred in some instances.

[The hazards associated with high winds were not addressed in the 2019 DNGS hazard screening analysis report NK38-REP-03611-10043 (Reference 2.6-1). However, there is a commitment in place by OPG to perform a high wind PSA as part of the Licence to Construct application, as indicated in Subsection 4.5.2 of the 2020 NK054-CORR-00531-10533 (Reference 2.6-8). The high wind PSA will consider the impact from wind pressure-loading effects and wind-propelled missile analysis from various categories of high wind, and their impact on severe core damage and large release analysis.

The review of literature and simulations from Environment Canada indicated in Subsection 7.2.8 of the 2009 NK054-REP-01210-00012 (Reference 2.6-3) points to expected changes in wind speed due to increased greenhouse gas emissions. The same subsection states the global average winds are expected to decrease in a warmer world due to the decrease in temperature differential between the equator and poles. In the 2022 NK054-REP-02730-00003, Wind Gust Analysis (Reference 2.6-14), it was reported that Lake Erie shores will experience a decrease in wind speeds of 5% by 2071-2100, while other areas in Ontario like James Bay and Georgian Bay will experience an increase in wind speeds ranging from 1.4% to 10%.

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Table 2.6-3: Extreme 3-second Gust Speeds for the DNNP (Reference 2.6-14)

	Return Period (year)	Gust speed (km/h) for each direction							
		NE	E	SE	S	SW	W	NW	N
DNNP Site	3000	116.5	153.4	106.2	131.7	172.3	165.6	145.5	115.8
	1700	115.5	147.7	101.5	125.7	165.3	159.5	140.5	111.3
	1000	106.7	142.0	97.4	120.2	158.8	153.9	136.0	107.3
	700	103.8	138.9	94.6	116.7	154.6	150.3	132.9	104.5
	300	96.5	130.5	87.9	108.0	144.3	141.4	125.6	97.9
	200	93.0	126.0	84.7	103.6	139.6	137.2	122.0	94.6
	100	87.0	120.0	79.1	96.8	131.0	129.7	116.0	89.4
	50	81.0	113.0	73.6	89.7	122.5	122.5	110.0	83.9
	20	72.9	103.0	66.2	80.2	111.5	112.9	102.0	76.8
	10	66.7	96.3	60.6	72.8	102.5	105.2	95.8	71.3

Table 2.6-4: Extreme 3-second Gust Speeds Envelop Based on Four-station Data Around the DNNP (Reference 2.6-14)

	Return Period (year)	Gust speed (km/h) for each direction							
		NE	E	SE	S	SW	W	NW	N
DNNP Site	3000	123.3	153.4	121.4	135.2	174.4	170.7	145.5	143.1
	1700	118.1	147.7	116.0	129.0	167.6	164.6	140.5	137.7
	1000	113.0	142.0	111.0	123.0	161.4	159.0	136.0	133.0
	700	110.2	138.9	107.8	119.6	157.2	155.4	132.9	129.3
	300	102.6	130.5	100.0	110.6	147.3	146.6	125.6	121.3
	200	99.0	126.0	96.3	106.0	142.5	142.0	122.0	117.0
	100	92.8	120.0	89.8	99.0	134.4	135.0	116.0	111.0
	50	86.5	113.0	83.4	91.6	126.2	128.0	110.0	104.0
	20	78.2	103.0	74.8	81.7	115.2	118.0	102.0	95.6
	10	71.8	96.3	68.1	74.1	106.8	111.0	95.8	88.8

2.6.6 Tornadoes and Hurricanes

Tornadoes

As discussed in Section 3.2 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5), tornadoes are characterized as a rare and non-negligible threat, and a study of a design basis tornado was conducted to estimate the probability of occurrence at the DNNP site. The results of this study are presented in Table 3-7 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5).

The DNNP site characteristics associated with the design basis tornado are outlined in Table 3 of the 2010 N-REP-01200-10000 (Reference 2.6-9), summarized in the following, and listed in Table 2.6-5 and Table 2.6-6):

1. Maximum Pressure Drop – The design assumption for the decrease in ambient pressure from normal atmospheric pressure due to the passage of the tornado
2. Maximum Rotational Speed – The design assumption for the component of tornado wind speed due to the rotation within the tornado
3. Maximum Translational Speed – The design assumption for the component of tornado wind speed due to the movement of the tornado over the ground
4. Maximum Wind Speed – The design assumption for the sum of maximum rotational and maximum translational wind speed components
5. Radius of Maximum Rotational Speed – The design assumption for distance from the centre of the tornado at which the maximum rotational wind speed occurs
6. Rate of Pressure Drop – The assumed design rate at which the pressure drops due to the passage of the tornado
7. Tornado Missile Spectra – The design assumptions regarding missiles that could be ejected either horizontally or vertically from a tornado. The spectra identify mass, dimensions, and velocity of credible missiles

The DNNP site characteristics values in the 2010 N-REP-01200-10000 (Reference 2.6-9) are based on the U.S. NRC Regulatory Guide 1.76 Rev 1 (Reference 2.6-13), Region 2 design basis tornado values. The characteristics, and appropriate reasoning are summarized from the 2022 NK054-CORR-01210-1015770 Engineering Direction for DNNP Design Basis Tornado Values (Reference 2.6-12). The DNNP site is conservatively assumed to have the Site Characteristic Maximum Wind Speed Site Characteristic value of 321.8 km/h for maximum wind speed. This is supported by the following reasons:

- The Maximum Wind Speed of 321.8 km/h is the upper limit for an Enhanced Fujita scale 4 (EF-4) tornado.
- The Maximum Wind Speed of EF-4 is a conservative value for the Darlington Nuclear site, as the Maximum Wind Speed value is not a measured value for the site.
- The assessment performed of the occurrence of tornadoes within an area of 100 000 km² of the Darlington Nuclear site during the past 50 to 60 years indicated two category Enhanced Fujita scale 4 (EF-4) tornadoes were observed within 180 km of the site during that period.
- A probability of approximately 0.01% per year was predicted corresponding to an EF-4 category of damage for the Darlington Nuclear site.
- The U.S. NRC RG-1.76 Rev1 (Reference 2.6-13) values for the two subregions adjacent to the Eastern Great Lakes and the northeastern boundary of Region 1 are 327 Km/h and

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296 km/h, respectively. This further supports the use of 321.8 km/h as a bounding value for Darlington Nuclear site.

The missile spectrum in Table 2.6-6 is extracted from Table 2 of U.S. NRC RG-1.76 Rev1 (Reference 2.6-13), Region 2 values, which correspond to a maximum wind speed of 321.8 km/h.

Table 2.6-5: DNNP Site Characteristics for Design Basis Tornado (Reference 2.6-9)

Parameter	Value
Maximum Pressure Drop	6.3 kPa
Maximum Rotational Speed	257.4 km/h
Maximum Translational Speed	64.4 km/h
Maximum Wind Speed	321.8 km/h
Radius of Maximum Rotational Speed	45.7 km/h
Rate of pressure Drop	2.5 kPa/s

Table 2.6-6: DNNP Site Tornado Missiles Spectrum for Maximum Horizontal Speed (Reference 2.6-9)

Missile Type	Dimensions	Mass	Horizontal Velocity (V_{mh}^{max})	Vertical Velocity (0.67 of V_{mh}^{max})
Schedule 40 Pipe	0.168 m dia x 4.58 m long	130 kg	34 m/s	22.8 m/s
Automobile 5 m x 2 m x 1.3 m	5 m x 2 m x 1.3 m	1810 kg	34 m/s	22.8 m/s
Solid Steel Sphere	2.54 cm dia	0.0669 kg	7 m/s	4.7 m/s

Hurricanes

A tropical cyclone is a rapidly rotating storm system characterized by a low-pressure centre. Depending on the wind speed, it can be designated as hurricanes, tropical storms, or tropical depressions. Based on the information presented in Subsection 3.5.2 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4), there is a very low probability of a hurricane directly impacting the DNNP site, and it describes the probable maximum tropical cyclone as unlikely to yield gusts of more than 100 km/h which is lower than that of the design basis tornado. As such, wind hazard from a hurricane is not considered further.

Additionally, the 2009 NK054-REP-01210-00013 (Reference 2.6-4) states that a tropical storm such as Hazel, which occurred in 1954, would be the worst-case scenario from systems of tropical origin. During this storm, Toronto Pearson reported over 150 mm of rain in 2 days with sustained winds of 92 km/h for 2 hours and multiple hours with winds of over 70 km/h, per Subsection 3.5.2 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4). Precipitation caused from a tropical cyclone is covered in Subsection 2.5.3.5.

2.6.7 Waterspouts

A tornado forming over water is a waterspout. The Site Evaluation on Nuclear Safety Considerations in the 2009 NK054-REP-01210-00008 (Reference 2.6-5), Section 3.2, states tornadoes over water or waterspouts generally leave no trace and are rarely reported. Additionally, the report states it is less likely for tornadoes to form over water than over land.

However, the report assumes an equal distribution of tornadoes and waterspouts for a given area and calculates the probability of a tornado at DNNP site. It then concludes that with such a frequency, tornadoes can be characterized as a rare, but non-negligible threat and a study of a design basis tornado was required in order to estimate the probability of occurrences on-site. The DNNP site characteristics for design basis tornado is described in Subsection 2.6.6.

2.6.8 Dust Storms and Sandstorms

The assessment for the potential of dust storms or sandstorms was captured in the 2009 NK054-REP-01210-00013 (Reference 2.6-4) where Subsection 3.5.5 states a lack of evidence of these phenomena was identified from an extensive search through the available meteorological information relevant to southern Ontario. Hence, neither dust storms nor sandstorms were identified as potential hazards since the possibility of occurrence for these phenomena at the DNNP site is deemed to be highly unlikely.

2.6.9 Snow and Ice Load, Freezing Rain, and Ice Storm

Snow and Ice Load

The average daily snowfall recorded at the nearest monitoring station to the Darlington Nuclear site is between 3 cm and 5 cm from December to March, per Section 2.2 of the 2009 NK054-REP-01210-00008 (Reference 2.6-5). Similarly, the daily snowpack is typically recorded at the same location, and its highest point tends to occur in January, with a mean value of 8.6 cm.

Table 2.6-7 shows under Loading 1 the characteristic value of 2.2 kPa for snow and ice load for reactor designs considered for the DNNP site, per Subsection B.1.3, Table 3 of the 2012 N-REP-01200-10000 (Reference 2.6-9).

The 2019 DNGS hazard screening analysis report NK38-REP-03611-10043 (Reference 2.6-1) used the 1975 NBCC design criteria for the snowpack of 2.1 kPa (Loading 2 in Table 2.6-7).

For the DNNP, Subsection 4.5.2 of the 2020 NK054-CORR-00531-10533 (Reference 2.6-8) assumed that similar snowfall conditions to the ones experienced in DNGS are expected to occur at DNNP due to their proximity. In 2022, a study was performed in NK054-REP-02730-00004 Winter PMP Validation (Reference 2.6-15) where a 50-year recurrence snow fall depth and maximum one-day late winter rain load nearby Oshawa are used to calculate the roof loading. The resulting loading is 1.8 kPa, as shown in Table 2.6-7, Loading 3. Furthermore, Loading 4 of 1.71 kPa in Table 2.6-7 represents the calculated DNNP snow load based of an NBCC 100-year recurrence, following the recommendation of CSA N291:19 (Reference 2.6-16) and employing a 50- to 100-year conversion multiplying factor of 1/0.82, as described in the 2022 NK054-REP-02730-00004 Winter PMP Validation (Reference 2.6-15), noting CSA N291:19 (Reference 2.6-16) does not require adding WPMP. The DNNP estimated snow loads and winter PMP values listed in Table 2.6-7 for 50-year recurrence or 100-year recurrence with or without WPMP are equal or lower than the Darlington Nuclear site characteristic value (Loading 1) of 2.2 kPa listed in Subsection B.1.3, Table 3 of the 2012 N-REP-01200-10000 (Reference 2.6-9).

Table 2.6-7: Snow Loads and Winter PMP Values for DNGS and DNNP (Reference 2.6-15)

Loading ID	Nuclear Site	Values	Compliance Notes
1	Darlington Nuclear	2.2 kPa	Characteristic value for reactor designs considered for the site (2010 PPE - Reference 2.6-9)
2	DNGS	Snow: 2.1 kPa	Meets the 1975 NBCC requirements (2019 SNGS - Reference 2.6-1)
3	DNNP (50-year recurrence)	Snow: 1.4 kPa + WPMP: 0.4 kPa = Total: 1.8 kPa	Meets 2015 NBCC requirements for 50-year recurrence snowpack, plus 50-year recurrence winter PMP near Oshawa (2022 DNNP - Reference 2.6-15)
4	DNNP (100-year equivalent recurrence)	Snow: (1.4/0.82) = Total 1.71 kPa	Meets 2015 NBCC requirements and CSA N291:19 requirements using a multiplying ASCE/SEI 7-10 factor of 1/0.82 to calculate the 100-year recurrence snowpack (2022 DNNP - Reference 2.6-15), noting N291:19 does not require adding WPMP.

Freezing Rain

With respect to freezing rain, Subsection 4.5.2 of the 2020 NK054-CORR-00531-10533 (Reference 2.6-8) indicates this item was considered for assessment as part of the safety analysis for DNNP. The hazards associated with freezing rain were also screened out for DNNP due to low consequence, as indicated in the 2019 hazard screening analysis report NK38-REP-03611-10043 (Reference 2.6-1) and in the 2019 Site Preparation Nuclear Safety Licence Renewal Activity Report NK054-REP-01210-00108 (Reference 2.6-2).

Ice Storm

Ice storms present a potential hazard for the systems located outside the DNNP BWRX-300, as indicated in Subsection 4.5.2 of the 2020 NK054-CORR-00531-10533 (Reference 2.6-8). According to Subsection 4.5.5 of the 2019 NK38-REP-03611-10043 (Reference 2.6-1), a review of operating experiences indicates minor ice storms have not had an impact on other plants, but significant storms have caused losses of off-site power and switchyard failures. This event is described as an LOPP and is covered in Chapter 15, Subsection 15.5.3.2.4.

2.6.10 Lightning

The assessment of lightning strikes is provided in Subsection 3.5.3 of the 2009 NK054-REP-01210-00013 (Reference 2.6-4) in the context of frequency of occurrence, where Table 3.5.10 provides estimates of the cloud-to-ground flashes for Toronto and Trenton, while Figure 3.5.8 displays graphically the Average Annual Flash Density in southern Ontario. Based on the data evaluated, the vicinity of the DNNP site will likely experience a frequency of 2 to 3 cloud-to-ground flashes per year per square kilometer. The 2020 DNNP lightning data collected and evaluated per NK054-CORR-00531-10533 (Reference 2.6-8) confirmed lightning occurrences are frequent in southern Ontario.

Subsection 4.5.7 of the 2019 Hazard Screening Assessment NK38-REP-03611-10043 (Reference 2.6-1) for DNGS summarizes the potential consequences of lightning occurrences as induced fires and electromagnetic compatibility issues affecting the functionality of electrical

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systems. As shown in Table 4-3 of the 2019 NK38-REP-03611-10043 (Reference 2.6-1), the criterion assigned for lightning events is screening criterion QL-1 for DNGS, which is described as “an event of equal or lesser damage potential than similar events for which the plant has been designed.” This screening criterion is applicable to the DNNP site on the basis that adequate measures, such as fire barriers and qualification for electromagnetic compatibility, are incorporated in the BWRX-300 design, as described in Chapter 7, Section 7.1 and Section 7.3.

2.6.11 Windborne Debris

An analysis of windborne debris from various categories of high wind, also known as wind-propelled missiles, is assessed as part of the 2020 high wind PSA per NK054-CORR-00531-10533 (Reference 2.6-8). This assessment evaluated the impact of windborne debris on severe core damage and large release analysis. Tornado windborne missile hazard design basis is described in Table 2.6-6 in Subsection 2.6.6, Tornadoes and Hurricanes.

2.6.12 Climate Change Impact

As described in Subsection 2.5.4, the 2023 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts NK054-PLAN-07007-00001 (Reference 2.6-19) is developed with the objective of summarizing life cycle climate change considerations including relevant long-term monitoring that is described in Subsection 2.11.9.

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2.6.13 References

- 2.6-1 NK38-REP-03611-10043 R003, 2019, "Hazards Screening Analysis – Darlington," Ontario Power Generation.
- 2.6-2 NK054-REP-01210-00108 R000, 2019, "Site Preparation Nuclear Safety Licence Renewal Activity Report," Ontario Power Generation.
- 2.6-3 NK054-REP-01210-00012 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington - Part 5: Flood Hazard Assessment," Ontario Power Generation.
- 2.6-4 NK054-REP-01210-00013 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington - Part 4: Evaluation of Meteorological Events," Ontario Power Generation.
- 2.6-5 NK054-REP-01210-00008 R001, 2009, "Site Evaluation for OPG New Nuclear at Darlington - Nuclear Safety Considerations," Ontario Power Generation.
- 2.6-6 NK054-REP-01210-00016 R002, 2012, "Site Evaluation of the OPG New Nuclear at Darlington - Part 2: Dispersion of Radioactive Materials in Air and Water," Ontario Power Generation.
- 2.6-7 NK38-SR-03500-10001 R005, 2018, "Darlington Safety Report, Part 1 and 2," Ontario Power Generation.
- 2.6-8 NK054-CORR-00531-10533, 2020, "Application for Renewal of OPG's Darlington New Nuclear Project (DNNP) Nuclear Power Reactor Site Preparation Licence (PRSL)," Ontario Power Generation.
- 2.6-9 N-REP-01200-10000 R003, 2010, "Use of Plant Parameters Envelope to Encompass the Reactor Designs Being Considered for the Darlington Site," Ontario Power Generation.
- 2.6-10 NK054-REP-01210-00078 R007, 2021, "Darlington New Nuclear Project Commitments Report," Ontario Power Generation.
- 2.6-11 N-REP-NGD-IR-61100-0002, 1990, "NGD Meteorological Towers System Description and Operating Recommendations," Ontario Power Generation.
- 2.6-12 NK054-CORR-01210-1015770 R00, 2022, "Engineering Direction for Darlington Nuclear Project Design Basis Tornado Values," Ontario Power Generation.
- 2.6-13 U.S. NRC Regulatory Guide 1.76 Rev 1, Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants, July 2007.
- 2.6-14 NK054-REP-02730-00003, 2022, "Wind Gust Analysis," Ontario Power Generation
- 2.6-15 NK054-REP-02730-00004, 2022, "Winter PMP Validation," Ontario Power Generation
- 2.6-16 CSA N291:19, "Requirements for Nuclear Safety-related Structures," CSA Group.
- 2.6-17 NK054-REP-02730-00001, 2022, "Flood Hazard Assessment," Ontario Power Generation.
- 2.6-18 NK054-REP-02730-00002, 2022, "PMP Validation," Ontario Power Generation.
- 2.6-19 NK054-PLAN-07007-00001, 2023, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts," Ontario Power Generation.

2.7 Geology, Seismology and Geotechnical Engineering

Section 2.7 covers the following DNNP site-specific information:

- Site Location and Description - Subsection 2.7.1

Subsection 2.7.1 presents a general description of the site and identifies the study areas considered for the characterization of the site geological and geotechnical conditions

- Geological Characteristics - Subsection 2.7.2

Subsection 2.7.2 contains the geological characteristics of the site including descriptions of the site physiography, surficial and bedrock geology, and offshore bathymetric contours and lakebed geology

- Geotechnical Characteristics - Subsection 2.7.3

Subsection 2.7.3 describes the geotechnical and geological data collected at the site, presents subsurface soil and rock profiles and groundwater conditions, and provides an assessment of potential geotechnical hazards on structures

- Seismology Characteristics - Subsection 2.7.4

Subsection 2.7.4 summarizes the seismological characteristics of the site including descriptions of the regional geology and tectonic history, hazard models, regional seismicity and seismic sources, ground motion characterization, methodologies used for the PSHA, and geological hazards that could potentially affect the site and the plant design.

- Geotechnical and Seismological Requirements and DNNP Site Parameters - Subsection 2.7.5

Subsection 2.7.5 presents geotechnical and seismological parameters for the DNNP site including evaluation of bearing capacity and settlement, static and dynamic properties of rock, soil and engineered fill materials, geotechnical variability and uncertainty, Site Response Analysis (SRA), and groundwater level

The presented summary of geological, seismological, and geotechnical characteristics of the DNNP site and the surrounding region are based on:

- Site-specific characteristics from DNNP documents including the PSHA and the geological mapping of subsurface soil layers and bedrock, as well as relevant Darlington Nuclear site data.
- Available information developed during the DNNP site selection and preparation stages

In 2022 and 2023, several DNNP site-specific investigations and studies are completed as follows:

1. NK054-REP-01210-00175 R001, 2022, "Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project," Volumes 1 of 2 and 2 of 2 (Reference 2.7-39)
2. NK054-REP-10180-00001 R000, 2023 "Offshore Geotechnical Investigation," (Reference 2.7-40)
3. NK054-REP-03500.8-00001 R000, 2022, "Darlington New Nuclear Project - Site-Specific Probabilistic Seismic Hazard Assessment", (Reference 2.7-41)
4. NK054-REP-03500.8-00002 R000, 2022, "Darlington New Nuclear Project - Seismically-Induced Soil Liquefaction Assessment," (Reference 2.7-42)

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5. NK054-REP-03500.8-00003, 2023, "Darlington New Nuclear Project Foundation Interface Analysis (FIA) Report," (Reference 2.7-38)

These investigations, assessments and analyses are used to validate and update DNNP-specific geological and geotechnical characteristics and parameters of subgrade materials, results of PSHA, potential of liquefaction underneath the BWRX-300 facility buildings, as well as Lake Ontario bathymetry and lakebed geology.

2.7.1 Site Location and Description

The Darlington Nuclear site, where the DNNP BWRX-300 facility is to be built, is located about 65 km east of the City of Toronto on the north shore of Lake Ontario in the Municipality of Clarington, Region of Durham in Ontario, Canada. The DNNP site is located to the east of the existing DNGS as shown in Chapter 1, Appendix A, Figure A1.1-2. The site is at latitude 43° 53' north and longitude 78° 43' west, per the 2009 site geotechnical aspects evaluation NK054-REP-01210-00011 (Reference 2.7-1). (Refer to Section 2.1 for further information on the Darlington Nuclear site and the DNNP site description, layout, geography, and demography.)

The topography of the Darlington Nuclear site, shown in Figure 2.7.1-1, based on the Darlington Topographic Drawing NK054-DRAW-01210-00003 (Reference 2.7-26), indicates a gentle slope rising upward towards the east from an approximate elevation of 80 m to 88 m CGD, in a horizontal distance of about 400 m. Further east, the existing ground rises substantially to an elevation of about 100 m CGD near the east site boundary. The existing shoreline along the Darlington Nuclear site consists of a narrow beach with steep bluffs. Additional information about the Darlington Nuclear site topography is provided in Subsection 2.1.1.

The site is situated in an undulating to moderately rolling glacial till plain. However, the upper soils at the site are glaciolacustrine, indicating the site is in the Iroquois Plane. The previously irregular terrain was graded for the existing DNGS to an elevation of about 78 m CGD. For the DNNP, the terrain is planned to be graded to a grade elevation of 88 m CGD. The surface elevation for the DNNP site rises towards the north with a mean elevation of 100 m CGD just south of the Canadian National Railway tracks. To the north of the railway tracks, the ground is irregular ranging from 98 m to 106 m CGD. A higher ridge, starting from the shore just east of Raby Head, extends diagonally across the site in a northwesterly direction with levels of up to 15 m above the surrounding ground. Offshore from the site, the Lake Ontario bottom slopes away gradually reaching a depth of 6 m at about 425 m from shore and 14 m at approximately 1.2 km from shore. Offshore bathymetry is discussed in Subsection 2.7.2.4.

2.7.2 Geological Characteristics

Summaries based on the information in the 2009 DNNP Site Geotechnical Aspects Evaluation NK054-REP-01210-00011 (Reference 2.7-1), the 2013 DNNP Geotechnical Data Report NK054-REP-01210-00098 (Reference 2.7-29), the 2023 NK054-REP-10180-00001 Offshore Geophysical Investigation Report (Reference 2.7-40), and the 1989 DNGS Preoperational Summary Report No. 89575 (Reference 2.7-2) are presented in:

- Subsection 2.7.2.1 - Surficial Geology
- Subsection 2.7.2.2 - Site Physiography
- Subsection 2.7.2.3 - Bedrock Geology
- Subsection 2.7.2.4 - Offshore Bathymetric Contours / Lakebed Geology

These summaries furnish a framework within which the geological characteristics of the DNNP site and the surrounding region are described.

2.7.2.1 Surficial Geology

The regional surficial geology, for an area within an approximately 50 km radius from the DNNP site, is shown in Figure 2.7.2-1, as replicated from the 2009 NK054-REP-01210-00011 (Reference 2.7-1).

For the surficial geology, there are three general physiographic regions:

- The Oak Ridges Moraine on the north side of the regional study area
- The South Slope in the middle
- The Iroquois Plain, a wide belt along Lake Ontario in the south

The Oak Ridges Moraine Physiographic Region

The Oak Ridges Moraine is a significant geologic/hydrogeologic feature specific to southern Ontario. The moraine is a major source of groundwater recharge, and many creeks and rivers are derived from groundwater discharge from the moraine. It was formed by regional glaciation, the advance and recession of several ice sheets and the subsequent melting of the glaciers. The moraine marks the boundary between the Lake Simcoe ice lobe advancing from the north and the Lake Ontario ice lobe advancing from the south. It is a ridge of high land separating drainage northward to Lake Simcoe and southward to Lake Ontario.

The moraine consists of interbedded layers of glacial till, sand and gravel. The moraine has a distinctive hummocky terrain with knobs and kettles. The southern flank of the moraine is covered by the Halton Till, a silty to silt-clay till.

The South Slope Physiographic Region

The South Slope fills the area between the moraine and the Iroquois Plain. It consists of gentle to steep slopes but is more uniform compared to the irregular terrain of the moraine. It contains a number of drumlins which point to the southwest, indicating the general direction of glacier movement.

The Iroquois Plain Physiographic Region

The Iroquois Plain, an 8 to 12 km wide plain, lies between the former shoreline of Lake Iroquois and present-day Lake Ontario. Shoreline deposits and glaciolacustrine sediments are found in this area overlying the glacial tills. The shoreline deposits include sand and gravel bars and beach terraces as well as some deltas from former rivers and creeks flowing into Lake Iroquois. The lacustrine deposits, consisting of silts and clays overlying till are found further from the former shoreline. In the area of the site, the Iroquois Plain contains drumlins with a southeast orientation indicating the northwest glacial advance.

2.7.2.2 Site Physiography

The DNNP site is generally covered by upper and lower till deposits, per the 2009 NK054-REP-01210-00011 (Reference 2.7-1), as described in the following paragraphs.

The surface till in the DNNP area is similar to the Newmarket Till, a sandy silt to silt till. An earlier dense, to very dense, sandy silt to hard silty clay till overlies the bedrock. Bounded between the upper and lower tills are deposits of water-bearing sand or sand and gravel.

Earlier deposits of lacustrine varved silt and clay and stratified fine to medium sand overlie the upper till at lower elevations near the DNNP BWRX-300 location, as described in the 2013 NK054-REP-01210-00098 (Reference 2.7-29). These surficial lacustrine deposits consist of varved silt and clay and fine to medium sand of variable thickness, per the 2013 NK054-REP-01210-00098 (Reference 2.7-29).

Fill material of variable composition is present at the ground surface over portions of the DNNP site, as described in the 2013 NK054-REP-01210-00098 (Reference 2.7-29). The fill consists of a mixture of clay, silt, sand, and gravel.

Overburden thickness varies significantly from the north to the south. Overburden thickness in the Oak Ridges Moraine is approximately 200 m reducing in thickness towards the south with about 10 m of overburden at Lake Ontario.

2.7.2.3 Bedrock Geology

The bedrock is completely covered by Quaternary deposits and bedrock outcrops are found only in local quarries, as described in the 2009 NK054-REP-01210-00011 (Reference 2.7-1). The bedrock surface, from east to west, consists of the Simcoe Group overlain by the younger Blue Mountain (formerly the Whitby Formation) and Georgian Bay Formations. The Simcoe Group consists of the Gull River, Bobcaygeon, Verulam and Lindsay Formations (from deep to shallow). The dip of the bedrock formations is approximately 0.5 percent to the southwest.

The Blue Mountain Formation is a shale formation. The lower 2 m to 3 m includes what was formerly known as the Whitby Formation, a black, petroliferous calcareous shale which tends to weather grey on exposure. The shale is fissile and fossiliferous. The Lindsay Formation is a grey argillaceous limestone with a full formation thickness of approximately 67 m.

The Verulam, Bobcaygeon and Gull River Formations lie below the Lindsay Formation. They are shale and limestone formations. The Shadow Lake Formation, a sandstone and shale formation, lies unconformably on the Precambrian Basement, as explained in the 2009 NK054-REP-01210-00011 (Reference 2.7-1).

Based on the described bedrock geology, the bedrock at the site of the DNNP is mainly the Lindsay Formation overlying the Verulam and Bobcaygeon and Gull River Formations. The upper few meters of bedrock are shaley limestone and shale of the Blue Mountain Formation that overlies the Simcoe Group, as detailed in the 2013 NK054-REP-01210-00098 (Reference 2.7-29).

2.7.2.4 Offshore Bathymetric Contours / Lakebed Geology

The bathymetric contours of the lakebed along Lake Ontario shoreline of the Darlington Nuclear site are provided in the 2023 NK054-REP-10180-00001 Offshore Geophysical Investigation Report (Reference 2.7-40). This investigation was conducted to characterize the lakebed and sub-bottom materials and profile the depth to bedrock. The offshore geophysics methods used were:

- Seismic reflection
- Sub-bottom profiling
- Electrical resistivity tomography
- Multi-beam echosounder
- Side scan sonar
- Magnetometer

The most prominent feature of the lakebed topography reported in the 2023 NK054-REP-10180-00001 Offshore Geophysical Investigation Report (Reference 2.7-40) is a crescent shaped ridge and peninsula of shallower depths which wraps from the northeast to the west of the surveyed area. The shape of this ridge creates a deeper “bay” in the central west part of the surveyed area; to the southeast the lakebed drops off into deeper water, as shown in Figure 2.7.2-3 and Figure

2.7.2-4. The results are aligned with previous studies of the offshore bathymetry and lakebed surface geology, per the 1989 Report No. 89575 (Reference 2.7-2), as depicted in Figure 2.7.2-5 and Figure 2.7.2-6.

2.7.3 Geotechnical Characteristics

Subsection 2.7.3 includes the following information related to the geotechnical characteristics of the DNNP site:

- Subsection 2.7.3.1 describes available geotechnical and geological data collected for the DNNP site
- Subsection 2.7.3.2 presents subsurface stratigraphic soil and rock profiles and groundwater conditions at the DNNP site
- Subsection 2.7.3.3 provides an assessment of potential geotechnical hazards on the DNNP structures

2.7.3.1 Geotechnical Information Collected at the DNNP Site

Multiple geotechnical investigations have been completed for the DNNP site. The data compiled in the investigations described in this subsection are used in determining the static and dynamic subgrade properties of the DNNP site presented in Subsection 2.7.5.

2.7.3.1.1 CH2MHILL (2007, 2008) Study

The investigation was performed by CH2MHILL in late 2007 and early 2008 and included installing monitoring wells in 11 borings. The results of this study are presented in two reports, the 2009 DNNP Geotechnical Aspects Site Evaluation NK054-REP-01210-00011 (Reference 2.7-1) and the 2009 DNNP Geological and Hydrogeological Environment NK054-REP-07730-00005 (Reference 2.7-30). These boreholes covered an area larger than the boundary of the DNNP site. The locations of the monitoring wells and the corresponding borehole numbers (DN) within the area planned for the construction of the DNNP in the CH2MHILL study, are marked with red circles in Figure 2.7.3.1-1.

2.7.3.1.2 AMEC (2012) Study

Three vertical boreholes completed within the DNNP area by AMEC in the 2012 DNNP Geologic and Geophysical Evaluation NK054-REF-01210-0418696 (Reference 2.7-28) are used to obtain subsurface information to the depth of the Precambrian Basement rock. The results of this study are presented in the 2012 NK054-REF-01210-0418696 (Reference 2.7-28). The locations of these deep borings are shown in Figure 2.7.3.1-2. The boreholes included: AMC-01 to a depth of 231.6 m, AMC-02 to a depth of 239.6 m, and AMC-03alt to a depth of 239.6 m. This study provides detailed boring logs, downhole geophysical measurements including televiewer data, surface geophysical measurements, and laboratory testing results. The data compiled in this study was mainly used to characterize the bedrock units. The geotechnical data provided in this AMEC study for the soil units are limited.

2.7.3.1.3 EXP Service INC. (2013) Study

In the 2013 DNNP Geotechnical Data Report NK054-REP-01210-00098 (Reference 2.7-29), eight sampled boreholes were drilled at locations within the DNNP area as shown in Figure 2.7.3.1-3. The drilled boreholes were advanced to various depths between 34 m to 85 m below the surface. The geotechnical data include detailed stratigraphic information, results of in-situ Standard Penetration Tests (SPTs) with calibrated hammers, and data from laboratory testing of soil and rock samples. Subsurface cross-section diagrams developed as part of the EXP study are presented in Figure 2.7.3.1-4 and Figure 2.7.3.1-5.

2.7.3.1.4 WSP GOLDER (2022) Phase 1 Geotechnical Investigation Report

In the 2022 Geotechnical Investigations NK054-REP-01210-00175 (Reference 2.7-39), extensive drilling was conducted at locations within the DNNP area to determine engineering properties of soil and rock, with specific focus on the first BWRX-300 location as shown in Figure 2.7.3.1-6. The stratigraphic units identified for the DNNP site and corresponding description are listed in Table 2.7-1. The site investigation followed the guidelines of NEDO-33914-A (Reference 2.7-27), Section 3.1, to ensure an adequate characterization of the subsurface conditions that meet additional requirements specific to the BWRX-300 design as a deeply embedded Small Modular Reactor (SMR).

Sampling was conducted in conjunction with in-situ SPTs performed with calibrated automatic hammers and data from laboratory testing of soil and rock samples are outlined in Section 4 and Section 5 of the 2022 Phase-1 investigations report (Reference 2.7-39).

The methodology for the in-situ and laboratory test are outlined in Volume 1 – Factual Geotechnical Data Report of NK054-REP-01210-00175 (Reference 2.7-39). The types of tests conducted include:

- Soil chemical analysis for the following constituents:
 - Soil pH of soil for corrosion
 - Water-soluble sulfate
 - Chloride in water
 - Sulfate in water for concrete
- Soil resistivity analysis
- Vane shear tests (cohesive soils)
- Pressuremeter testing (soil), dilatometer testing (rock), piezocone soundings (soil), soil resistivity, packer testing (rock), over-coring stress testing (rock)
- Uniaxial Compression Stress (UCS) testing (rock)
- Triaxial compression stress testing (soil)
- Constant stress direct shear creep testing on rock joints
- Swell testing (rock)

2.7.3.2 Subsurface Stratigraphic Profile

2.7.3.2.1 Profiles for the DNNP Site (2022)

The stratigraphy for the DNNP site soil and bedrock units listed in Table 2.7-1 is developed based on the work performed in the 2022 Geotechnical Investigations NK054-REP-01210-00175 (Reference 2.7-39).

Details of the in-situ stratigraphic layers average and range of thicknesses are provided in Table 2.7-2 for the soil units and in Table 2.7-3 for the rock units. The interpreted soil and rock stratigraphy are presented in east-west oriented and north-south oriented cross-sections in Figure 2.7.3.2-1 and Figure 2.7.3.2-2, respectively. Further details for subsurface soil and bedrock profiles are described in the following paragraphs.

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Unit 1 – Topsoil/Fill

The uppermost layer is topsoil and/or fill consisting of either poorly graded sand with gravel or sandy lean clay. Unit 1 was encountered at ground surface at all boreholes drilled for the 2022 Phase 1 Geotechnical Report NK054-REP-01210-00175 (Reference 2.7-39). This layer has an average thickness of 1.59 m ranging from 0.53 m in borehole 27 to 3.53 m in borehole 67.

Units 2a and 2b – Surficial Glaciolacustrine Deposits

Two glaciolacustrine deposits are found below the upper topsoil and fill. The upper deposits (Unit 2a) are encountered below the topsoil/fill layer. Unit 2a consists of silt, clay, fine to coarse sand and trace to some subrounded to subangular gravel. The lower deposits (Unit 2b) consist of silt with some clay, fine to coarse sand and subrounded to angular, fine to coarse gravel.

In some boreholes, Units 2a and 2b were observed to be interlayered. The combined thickness of Units 2a and 2b averages 1.74 m, and is ranging from zero in borehole 6, to 6.1 m in borehole 5.

Unit 3 - Upper Till

Deposits of silty sand with gravel to sandy lean clay with gravel are encountered below Units 2a and 2b. Unit 3 is described as a till layer generally consisting of a heterogeneous mixture of dense to very dense gravel, boulders, and cobbles in a matrix of silty sand. This deposit consists of silt, clay, fine to coarse sand and subrounded to subangular to angular, fine to coarse gravel. Unit 3 ranges in thickness from zero in borehole 4 to 13.49 m in borehole 17, with an average thickness of 7.35 m.

Units 4a and 4b – Intermediate Glaciolacustrine Deposits

Two distinct glaciolacustrine deposits are founded below Unit 3. The upper deposit, Unit 4a consists of silt, clay, fine to coarse sand and subrounded to angular, fine to coarse gravel. Boulders and cobbles are also present within Unit 4a. Below Unit 4a is Unit 4b which consists of silt, clay, fine to coarse sand and trace to some subrounded to angular gravel.

In some boreholes, Units 4a and 4b were observed to be interlayered. The combined thickness of units 4a and 4b averages 11.3 m, and ranges between zero in borehole 11SB to 17.7 m in borehole 27.

Unit 5 – Lower Till

Below the intermediate glaciolacustrine deposits (Units 4a and 4b), a deposit of very dense silt and sand to hard lean clay (Unit 5) is encountered. Unit 5 is described as a lower till layer generally consisting of a heterogeneous mixture of gravel, boulder, and cobbles in a matrix of silt sand and silty clay. This deposit consists of silt, clay, fine to coarse sand, and subrounded to angular, fine to coarse gravel. It has an average thickness of 3.57 m, ranging from zero in borehole 16 to 6.63 m in borehole 15.

Unit 6a – Blue Mountain Formation Bedrock

The top of the bedrock is at an average elevation of 64.20 CGD, ranging from 62.72 m CGD in borehole 6 to 65.80 m CGD in borehole 70.

Below Unit 5, is a moderately weathered to fresh, very thinly to medium bedded, fine grained, faintly porous, slightly to moderately reactive to hydrogen chloride, weak to strong shale with thin, limestone interbeds. Unit 6a has an average thickness of 2.98 m, ranging from 1.38 m in borehole 73 to 5.87 m in borehole 30.

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Unit 6b – Lindsay Formation Bedrock

Below Unit 6a is a slightly weathered to fresh, very thinly to medium bedded, fine to medium grained, faintly porous, slightly to moderately reactive to hydrogen chloride, weak to medium strong to very strong limestone with shale interbeds, Unit 6b has an average thickness of 61.36 m, ranging from 60.61 m in borehole 16 to 61.93 m in borehole 65.

Unit 6c – Verulam Formation Bedrock

Below Unit 6b is a fresh very thinly to medium bedded, grey, fine to medium grained, faintly porous, moderately reactive to hydrogen chloride, medium strong to very strong limestone with shale interbeds. Full thickness of Unit 6c was not tested.

Table 2.7-1: Stratigraphic Units for the DNNP Site

Unit No.	Description
1	Topsoil / Fill
2a	Surficial Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay
2b	Surficial Glaciolacustrine Deposits – Silty Clayey Sand to Silty Sand/Sandy Silt
3	Upper Till
4a	Intermediate Glaciolacustrine Deposits – Silty Sand to Sandy Silt
4b	Intermediate Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay
5	Lower Till
6a	Blue Mountain Formation Bedrock
6b	Lindsay Formation Bedrock
6c	Verulam Formation Bedrock

Table 2.7-2: In-situ Soil Units Stratigraphy under the Power Block (Reference 2.7-39)

Layer	Layer Thickness (m)							
	Reactor Building ¹		Power Block ²		BWRX-300 Protected Area ³		BWRX-300 Study Area ⁴	
	Average	Range	Average	Range	Average	Range	Average	Range
Unit 1	1.25	0.61 – 2.13	1.81	0.61 – 3.28	1.77	0.61 – 3.53	1.59	0.53 – 3.53
Unit 2a Unit 2b	1.73	0.61 – 3.81	2.32	0.00 – 6.09	2.35	0.00 – 6.09	1.74	0.00 – 6.10
Unit 3	6.24	1.07 – 8.87	6.01	0.00 – 9.06	6.26	0.00 – 13.47	7.35	0.00 – 13.49
Unit 4a, Unit 4b	9.32	0.00 – 14.48	9.78	0.00 – 14.32	9.07	0.00 – 14.54	11.30	0.00 – 17.70
Unit 5	2.29	1.36 – 2.98	3.78	1.36 – 6.63	3.25	0.86 – 6.63	3.57	0.00 – 6.63

Notes:

1. Includes borings BH 9, BH 10, BH 11, BH 11S, BH 11 SB, BH 12, BH 14 (Reference 2.7-39)
2. Includes borings BH 2, BH 4, BH 5, BH 9, BH 10, BH 11, BH 11S, BH 11SB, BH 12, BH 13, BH 14, BH 18, BH 19, BH 67, BH 68, BH 71, BH 73, BH 78 (Reference 2.7-39)
3. Includes borings BH 2, BH 4, BH 5, BH 6, BH 8, BH 7, BH 9, BH 10, BH 11, BH 11S, BH 11SB, BH 12, BH 13, BH 14, BH 15, BH 16, BH 17, BH 18, BH 19, BH 20, BH 66, BH 66S, BH 66SB, BH 67, BH 68, BH 71, BH 73, BH 77, BH 78 (Reference 2.7-39)
4. All boreholes considered in the study area in (Reference 2.7-39)

Table 2.7-3: Rock Units Stratigraphy (Reference 2.7-39)

	Layer Thickness or Depth (m)	
	Average	Range
Elevation Top of Bedrock	64.20 (CGD)	62.72 (BH 6) - 65.80 (BH 70)
Thickness Unit 6a - Blue Mountain Formation	2.98	1.38 (BH 73) - 5.87 (BH 30)
Thickness Unit 6b - Lindsay Formation	61.36	60.61 (BH 16) - 61.93 (BH 65)

Notes:

1. Full thickness of the Verulam Formation (Unit 6c) was not tested (Reference 2.7-39)
2. Lindsay formation thickness determined from small sample ~ (15%) of boreholes which extended fully through the formation (Reference 2.7-39)

2.7.3.2.2 Planned As-Built Soil Profile

Stratigraphic Units 1 and 2 are generally loose, have liquefaction potential (Subsection 2.7.4.7.6), and are not suitable for supporting the heavy foundations of the power block buildings. As a result, during site development, these soil layers will be excavated and replaced with compacted engineered fill.

Consequently, the as-built conditions at the site after construction of the BWRX-300 facility are anticipated to include compacted engineered fill from about elevation ranging between 80 m to 82 m CGD to the final grade at elevation 88 m CGD. The excavated soil from this site may be used as compacted engineered fill material if it meets the engineered fill gradation requirements outlined in the 2023 DNPP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

2.7.3.2.3 Bedrock Profile

The bedrock profile was developed based on readily available top-of-rock information from boreholes drilled for the geotechnical study in the 2022 NK054-REP-01210-00175 (Reference 2.7-39). Data between boreholes have been interpolated.

The top of the bedrock surface undulates relatively locally and slopes gently to the south from an elevation of 67 m CGD near the northern extent of the site to an elevation of 64 m CGD. This bedrock surface is consistent with the mapped sub-horizontal dip of the Paleozoic sequence observed within the vicinity of the project area.

Subsurface rock conditions may vary between and beyond the borehole/drillhole locations. The interpreted stratigraphy is therefore a simplification of the subsurface bedrock contacts. Variations in the stratigraphic boundaries between boreholes/drillholes will exist and are to be expected. Table 2.7-3 presents the top of bedrock elevation and bedrock thicknesses.

The BWRX-300 deeply embedded RB is anticipated to extend through the Blue Mountain Formation (Unit 6a) and be founded in the Lindsay Formation (Unit 6b) at 52.93 m CGD. The top of the Blue Mountain Formation near the BWRX-300 RB is anticipated to be at about 64 m CGD based on the depth to bedrock at BH 10, BH11 and BH 12 (refer to Figure 2.7.3.1-6), as explained

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in the 2022 Power Block geotechnical investigations NK054-REP-01210-00175 (Reference 2.7-39).

2.7.3.2.4 Groundwater Conditions

Based on the information provided in the 2022 DNNP Phase 1 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.7-39), the following three groundwater flow patterns are identified:

- The Unit 3 water table (shallow groundwater), shown in Figure 2.7.3.2-3
- Unit 4a groundwater flow in the integrated deposits, shown in Figure 2.7.3.2-4
- Unit 5 groundwater in the interglacial deposits located above the bedrock, shown in Figure 2.7.3.2-5
- Units 6a-6b groundwater in bedrock, shown in Figure 2.7.3.2-6

The groundwater flow interpretations in these figures (Figures 2.7.3.2-3 to 2.7.3.2-6) are based on a monitoring well-network with only a few months of monitoring data. The actual long-term interpretation may change. The contours are based on data from the new monitoring wells installed within the investigation area, which are limited in aerial extent, and have not been considered with contemporary groundwater elevation data from the pre-existing monitoring well-network at the site.

As shown on the figures, the groundwater flow direction in the upper and lower till (Units 3 and 5, respectively) is inferred to be toward the southwest and, in the intermediate glaciolacustrine deposits (Unit 4a) and shallow bedrock (Units 6a and 6b), to be toward the south-southeast.

Regional groundwater flow and flow at the DNNP site generally follows topography from higher elevations in the north towards the south, per the 2009 DNNP Geological and Hydrogeological Environment NK054-REP-07730-00005 (Reference 2.7-30). In general, this flow is driven by recharge from rainfall and snowmelt infiltration across the area and at higher elevations along the Oak Ridges Moraine north of the DNNP site with discharge, ultimately, to Lake Ontario to the south. The shallow groundwater system at the DNNP site deviates from this flow pattern near surface water conveyances and local recharge areas. Interpreted regional groundwater flow patterns documented in the 2009 report NK054-REP-01210-00011 (Reference 2.7-1) are shown in Figures 2.7.3.2-7, 2.7.3.2-8 and 2.7.3.2-9 for shallow water table, interglacial deposits, and shallow bedrock groundwater, respectively.

The hydro-stratigraphic units at the DNNP site follow the soil and geologic units. The upper till (Unit 3) forms an aquitard or confining layer at the site which restricts downward groundwater flow from the upper fill and glaciolacustrine materials. The interglacial deposits (Units 4a-4b) are the most significant hydrogeologic unit at the site since they extend across the site and to the recharge areas north of the site, as described in the 2009 NK054-REP-07730-00005 (Reference 2.7-30). There may be significant groundwater flow in the interglacial deposits due to the higher gradient and higher permeability of the materials. The lower till (Unit 5) beneath the interglacial deposits is also considered an aquitard with low permeability. Although flow in the upper till is downward due to under-draining by the interglacial deposits, there may be an upward component of flow through the lower till in some areas from the underlying upper bedrock aquifer, per the 2009 NK054-REP-07730-00005 (Reference 2.7-30).

The upper bedrock is likely fractured and weathered with higher secondary permeability and transmissivity compared to the intact bedrock. Flow in the upper bedrock is expected to be enhanced in areas where the lower till is absent, and the upper bedrock is in direct contact with the more permeable interglacial deposits. The lower bedrock at the DNNP site generally has low

permeability and does transmit much groundwater. The groundwater conditions in the deeper bedrock formations below the Lindsay Formation have not been considered for study.

2.7.3.3 Evaluation of Geological Hazards on DNNP Structures

Subsection 2.7.3.3 provides an assessment of geological hazards that could impact the DNNP structures.

2.7.3.3.1 Karst Cavities

No evidence of significant karst cavities was encountered in the 2022 geotechnical boreholes (Reference 2.7-39). Some zones of lost core were encountered in the boreholes generally within 40 m of the ground surface and ranging from 5 cm to 66 cm in height, but no noticeable drop in the drilling rods was noted and therefore these are thought to be associated with zones of weathered and fragmented rock that had been washed out by the core drilling.

In addition, the previous geophysical reports associated with this site indicate the absence of anomalies in the rock that could indicate the presence of depressions or voids that may be indicative of large karst or faults. There is good seismic coverage with nine seismic refraction lines being executed at the site.

Review of the previous surface geophysical data as well as the numerous boreholes drilled in 2021 to 2022 for the power block, and the associated data (Reference 2.7-39) confirm the absence of karst features at this site.

2.7.3.3.2 Ground Frost

The conventional approach for protection of building foundations against frost action is to locate base of foundations and/or the base of grade beams (supported on deep foundation) at a depth at least equal to the depth of frost penetration. A minimum frost penetration depth of 1.3 m is therefore recommended, in accordance with OPSD 3090.101 (Foundation Frost Penetration Depths for Southern Ontario), as per the 2022 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.7-39). Partial or complete frost protection may also be achieved by using rigid polystyrene insulation.

Frost heaving may occur in fine grained soils where ice lenses occur when moisture is drawn to freezing horizons. Based on the existing site subsurface conditions, shallow silty fine sand and silt deposits below surficial granular fill are generally expected to be moderately to highly frost susceptible to heaving under freezing conditions. Therefore, adequate frost cover of 1.3 m depth is required for all foundations exposed to frost conditions.

2.7.3.3.3 Bearing Failure (Collapse)

The 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38) evaluated the bearing capacities for the RB foundation and resulting bearing capacities for the Turbine Building (TB), Control Building (CB), Radwaste Building (RWB), and Reactor Auxiliary Bay foundations surrounding the deeply embedded RB using data reported in the 2022 geotechnical site investigations (Reference 2.7-39). The anticipated bearing pressure and bearing capacity for each building in the power block is summarized and discussed in Subsection 2.7.5.1.

2.7.3.3.4 Stability of Foundation

The 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38) provides the anticipated maximum uniform and differential settlements of the RB, TB, CB, RWB, and Reactor Auxiliary Bay foundations. The anticipated bearing pressure and associated settlements are summarized and discussed in Subsection 2.7.5.1.

2.7.3.3.5 Stability Of Subgrade Surrounding the Reactor Building

A stability analysis was performed following the guidelines of NEDO-33914-A (Reference 2.7-27), Section 4.0, using the finite element software PLAXIS (Bentley) to perform advanced non-linear Soil-Structure Interaction (SSI) numerical modeling. In addition to the stability analysis, the potential for instability of the potentially unstable blocks or wedges surrounding the RB deep excavation were performed using UnWedge (RocScience), a 3D stability analysis and visualization program. The stability analysis is discussed further in Subsection 2.7.5.1 and all the analyses are detailed in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

2.7.3.3.6 Transitional Ground Heave and Settlement

As part of site grading and development, there will be unloading and transitional ground heave resulting from excavation of the upper soft-to-loose soil layers of Units 1 and 2 at the site. Additionally, some of the heave will be offset by settlement, which will occur on completion of backfilling. Depending on the net change in the overall effective stress profile, net ground heave is expected to occur due to reduction in the finished ground level compared to existing levels.

During the process of unloading and re-loading, stratigraphic Units 3, 4 and 5 are expected to react quickly to the changes in the ground stresses with minimal lag. Hence, long-term consolidation or heave is not expected to occur. Rather only transitional elastic rebound, and compression are expected to occur, as documented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

It is anticipated that there will be about 10 mm of heave from offloading due to excavation and some nominal heave/settlement after the completion of fill placement. There may be some ongoing creep settlement from the fill placement; however, ground movements will be small and the impact on structures founded on or in the overburden soils will be insignificant, as described in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

2.7.3.3.7 Stability of Natural Slopes

The structures located within the power block are level at finished grade and over 100 m away from the shoreline. The structures are expected to be founded on or in either engineered fill, very dense native Unit 3 soil or deeply embedded in strong to very strong bedrock. Therefore, slope instability will not be threat to these power block structures. However, the natural shoreline is prone to erosion, especially the steep bluffs to the east of the power block area. Erosion of the shoreline has the potential to pose a hazard eventually, through gradual reduction of the ground pressure, if allowed to progress over long periods. This is discussed in the 2022 NK054-REP-03500.8-00002 Darlington New Nuclear Project - Seismically-Induced Soil Liquefaction Assessment (Reference: 2.7-42). Prevention of erosion is to be achieved through the establishment of engineered shoreline protection. The steep bluffs as a slope do not pose a hazard to the first BWRX-300 unit planned, and design of subsequent units will mitigate the hazard as required.

2.7.3.3.8 Stability of Cut and Fill Slopes

The existing ground to the east of the existing DNGS will be excavated to form a large level area for the DNNP and its associated structures. For preliminary design purposes, cut slopes into the competent interglacial/till deposits will be at a general inclination of 1V:3H (18.4°). The excavated soils will be partially stored at the north-east part of the site. The fill slopes will be designed to ensure stability.

2.7.3.3.9 *Stability of Dikes and Dams*

No dams are currently present or planned for the DNNP. No dikes are currently present or planned on DNNP, and lake infilling is no longer planned for the project.

2.7.4 Seismology Characteristics

Subsection 2.7.4 summarizes findings of past seismic hazard investigations as well as of the 2022 site-specific PSHA (Reference 2.7-41) that were performed for the DNNP and DNGS site.

Subsection 2.7.4 includes:

- Subsection 2.7.4.1 - provides background seismological information and data collected since 1997
- Subsection 2.7.4.2 - describes the regional geological structure and tectonic history of the Darlington Nuclear site
- Subsection 2.7.4.3 - presents information on the seismicity of the region surrounding the site and the development of earthquake catalogue
- Subsection 2.7.4.4 - describes the seismic hazard model containing regional and local sources
- Subsection 2.7.4.5 - describes aspects related to ground motion characterization
- Subsection 2.7.4.6 - discusses the PSHA methodology and the results for the DNNP site
- Subsection 2.7.4.7 - describes protentional geological and seismological aspects at the DNNP site

2.7.4.1 Background and Data Collection

In 2009, the Darlington Nuclear site was evaluated for suitability for the DNNP. A PSHA was performed, per the 2009 NK054-REP-01210-00014 (Reference 2.7-4) in accordance with:

- CNSC Regulatory Document RD-346 Site Evaluation for New Nuclear Power Plants (Reference 2.7-5), which is superseded by CNSC's REGDOC 1.1.1 Site Evaluation and Site Preparation for New Reactor Facilities (Reference 2.7-6)
- IAEA NS-R-3 (Reference 2.7-7), which is superseded by SSR-1 (Reference 2.7-8)

The 2009 PSHA (Reference 2.7-4) details assembly of the geological, geophysical, and seismological data collection for the region, near region and vicinity of the DNNP site. The approach adopted utilized the 1997 study (Reference 2.7-3) as a starting point. The database assembled for that study was updated, and the effects of the updates of regulatory requirements in CNSC RD-346 (Reference 2.7-5) and IAEA NS-R-3 (Reference 2.7-7) were evaluated, and changes were incorporated. The 2009 PSHA was thereafter revised three times: in 2011 in NK38-REP-03611-10041 R000 (Reference 2.7-9), in compliance with CSA Standard N289.2 (Reference 2.7-31); in 2019 in NK38-REP-03611-10041 R002 (Reference 2.7-10), and in 2021 in NK38-REP-03611-10041 R003 (Reference 2.7-11), with minor changes to address CNSC comments not previously incorporated. The PSHA updates in both the 2019 NK38-REP-03611-10041 R002 (Reference 2.7-10) and the 2021 NK38-REP-03611-10041 R003 (Reference 2.7-11) include:

- Updates to the Earthquake Catalogue
- Updates to the Maximum Magnitude Assessment
- Updates to Earthquake Occurrence Rates

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- Application of the Next Generation Attenuation -East Ground Motion Model

In 2022, a DNNP site-specific PSHA (Reference 2.7-41) was conducted in accordance with the requirements of CNSC REGDOC 2.5.2 and CSA N289 series, as well as with the BWRX-300 SMR specific design requirements listed in NEDO 33914-A (Reference 2.7-27). In addition, the 2022 PSHA study (Reference 2.7-41) used the 2001 NUREG/CR-6728 (Reference 2.7-20) to develop site-specific ground motions considering local site conditions.

2.7.4.2 Regional Geological Structure and Tectonic History

2.7.4.2.1 Regional Geological Structure Stratigraphy

The Darlington Nuclear site lies within the western Lake Ontario region in the tectonically stable interior of the North American continent, which is characterized by low rates of historical seismicity, as described in the 1994 EPRI TR-102261-V1 (Reference 2.7-12). The region is underlain by middle Proterozoic (about 900 to 1600 million years ago) Grenville basement rock and overlying Paleozoic (about 250 to 570 million years ago) shallow-water sedimentary strata.

The Grenville Province formed in response to several phases of compression and metamorphism. The “Grenville Front” and “Grenville Front Tectonic zone”, shown in Figure 2.7.4.2-1, is the contact between the Grenville Province to the east and the continental Eastern Granite-Rhyolite provinces to the west. Rocks of the Central Gneiss Belt are between the “Grenville Front Tectonic Zone” and the Central Metasedimentary Belt Boundary Zone. The Central Metasedimentary Belt Boundary Zone underlies the western end of Lake Ontario, and the Central Metasedimentary Belt underlies the rest of Lake Ontario and the site study region. The Central Metasedimentary Belt is an intensely faulted and folded zone formed less than 1,300 million years ago. The southeastern portion of the Central Metasedimentary Belt consists of slightly younger rock. The Grenville orogeny (mountain-building episode) is widely attributed to a continental collision; however, deformation occurred in several episodes of extension and compression.

The Grenville Province’s crustal structure is characterized by north-northeast-striking, relatively shallow east-southeast-dipping ductile thrust faults that developed at mid- to lower-crustal depths during the middle Proterozoic Grenville orogeny. Prominent north-northeast-trending geophysical anomalies associated with exposed Grenville structures extend southward beyond the Canadian Shield and beneath the unconformable lower Paleozoic cover rocks. Regional geologic maps (e.g., Ontario Geological Survey, 1991) indicate that the overlying Paleozoic rocks are, with few exceptions, relatively flat-lying and laterally continuous, indicating that no large-scale, major faulting has occurred in the region since they were deposited.

The notable exception to the lack of regional-scale faulting in southern Ontario and Quebec occurs within the St. Lawrence rift system, as described in the 1966 Canadian Journal of Earth Sciences, Volume 3, No. 5 (Reference 2.7-13), which is a remnant of the late Proterozoic/early Paleozoic Iapetus passive margin, as described in the 1996 published article of R.L. Wheeler (Reference 2.7-14). The St. Lawrence rift system comprises abundant large-scale normal faults displacing lower Paleozoic strata and underlying Grenville basement on the order of many hundreds of meters along the Ottawa, Champlain, St. Lawrence, and Saguenay River valleys (Reference 2.7-13). These extensional faults generally cut discordantly across Grenville-aged structures instead of reactivating them. Mesoscopic-scale faulting of the lower Paleozoic strata, with fault displacements ranging from less than a meter to several tens of meters, has been recognized locally throughout much of the Lake Ontario region outside of the St. Lawrence rift system. The St. Lawrence rift system is associated with zones of elevated and persistent seismicity, per Slemmons, D.B., et al. in 1991 (Reference 2.7-15).

Worldwide, the seismic potential of a stable continental region varies according to the degree of crustal extension that it experienced in the geologic past, and to a lesser extent, the age of the

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crust, per the 1994 EPRI TR-102261-V1 (Reference 2.7-12). Three types of crust are identified in eastern North America, per the 1994 EPRI TR-102261-V1 (Reference 2.7-12):

- Unrifted - the craton and the Appalachian fold belt
- Failed intracontinental rifts—the Ottawa and Saguenay aulacogens and the Reelfoot rift complex
- Rifted passive continental margin—the Atlantic passive margin produced by the present opening of the Atlantic Ocean in the late Mesozoic, and a relic passive margin produced by lapetan rifting in the late Proterozoic/early Paleozoic

The north-northeast-trending faults along the Champlain and St. Lawrence River valleys, once attributed to a two-sided, failed intracontinental rift, are now recognized as part of the southeast-facing lapetan margin, per R. L. Wheeler, in 1996 (Reference 2.7-14). The present-day Atlantic passive margin comprises transitional crust (continental-oceanic) and the extended and faulted inboard continental shelf.

Evidence of lapetan rifting of the craton adjacent to the northern Appalachians is recorded within the St. Lawrence rift system (Reference 2.7-13) in the form of rift-related extensional structures, sediments, and magmatic/volcanic products that developed along the ancient continental margin. The rift structures include zones of echelon faults parallel to the ancient margin, possible fracture zones transverse to the ancient margin, and two well-defined aulacogens (failed rifts)—the Ottawa and Saguenay grabens.

The Appalachian orogen lies approximately 400 km east of the Darlington Nuclear site. Northern Appalachian orogenic events occurred from Ordovician to Permian time and consisted of several distinct tectonic episodes. As discussed in the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference 2.7-41), the key structural elements that mark the boundaries of the various crustal provinces (e.g., the western limit of Mesozoic extensional structures) are used to define regional seismic source zones that are characterized by similar crustal properties (for an example of one boundary interpretation, refer to Figure 2.7.4.2-2).

2.7.4.2.2 *Neotectonics Setting*

The geologically most recent evidence for major tectonic activity in the region is Alleghanian (late Permian) thrust faults formed in the Appalachian foreland basin and late Triassic to late Jurassic normal faults along the Atlantic margin related to continental rifting and the subsequent opening of the Atlantic Ocean, per the 2009 NK054-REP-01210-00014 DNNP PSHA (Reference 2.7-4). However, historical seismicity along the St. Lawrence rift system, in the Charleston, South Carolina, area, and in other concentrated zones; local geologic evidence of Cenozoic reactivation of faults; evidence of seismically-induced liquefaction in susceptible sands and silts; and geologic and geodetic data indicative of regional and local crustal deformation suggest continuing neotectonic activity, albeit at much lower rates than during the last episode of major tectonic deformation.

Slemmons, D.B., et al. in 1991 (Reference 2.7-15) have reported that most large historical and instrumental earthquakes in eastern Canada have occurred near Paleozoic or younger rift zones. This is similar to stable continental region earthquakes worldwide, as described in the 1994 EPRI TR-102261-V1 (Reference 2.7-12). The early Paleozoic St. Lawrence rift system, which is delineated by a persistent pattern of seismicity, is the postulated source of numerous large, historical earthquakes in southeastern Canada, per Slemmons, D.B., et al. in 1991 (Reference 2.7-15). Seismicity along this rift system appears to be concentrated in a number of well-defined clusters, including the Ottawa River, Charlevoix, and lower St. Lawrence River seismic zones, which are all separated by relatively aseismic regions.

Equivocal evidence for neotectonism, per Thomas, R.L., et al. in 1993 (Reference 2.7-16), has been found in the Lake Ontario region, and there are difficulties in distinguishing between deformation related to glacial processes and that related to deep-seated tectonic processes.

East-northeast/west-southwest-trending lakebed features in the Rochester basin of Lake Ontario and the Hamilton-Presqu'île fault zone, along with some of the features observed in western Lake Ontario, have been proposed by Thomas, R.L., et al. in 1993 (Reference 2.7-16), as neotectonic evidence for the southwest continuation of the St. Lawrence rift system through Lakes Ontario and Erie.

The postulated northwestern boundary of the late Proterozoic/early Paleozoic lapetan rifted margin tectonic province lies approximately 80 km east of the site, per Wheeler, R.L. in 1995 (Reference 2.7-17). There also is deep seismic evidence suggesting that the western boundary of the lapetan margin may lie farther to the west, along the Central Metasedimentary Belt Boundary Zone of the Grenville province as described by Milkereit, B., et al. in 1992 (Reference 2.7-18). These alternative boundaries are considered in defining regional seismic source zones (for an example of one boundary interpretation, refer to Figure 2.7.4.2-2).

The rate of historical seismic activity in the Grenville Province west of the lapetan rifted margin is low and appears typical of stable cratonic crust, per the 1994 EPRI TR-102261-V1 (Reference 2.7-12). In general, seismic activity and the geologic conditions most associated with earthquake activity in the stable continental region of Central and Eastern North America increase towards the east, away from the Precambrian central craton and towards the rifted passive continental margin.

2.7.4.3 Seismicity

Characterization of the seismicity of the region surrounding the DNNP site forms an essential part of the assessment of the seismic hazard. The primary means of characterization of seismicity is the use of the earthquake catalogue to assess earthquake occurrence rates and maximum magnitudes for earthquake sources.

In the Darlington Nuclear site PSHA studies, presented in the 2019 NK38-REP-03611-10041 R002 (Reference 2.7-10) and the 2021 NK38-REP-03611-10041 R003 (Reference 2.7-11), the 2012 NUREG-2115 (Reference 2.7-21) earthquake catalogue was updated to include independent earthquakes from the end of 2008 through 20 May 2019. The earthquake catalogue was again updated in the 2022 NK054-REP-03500.8-00001 DNNP PSHA (Reference 2.7-41) to extend the duration of the catalogue to the end of December 2021 using the:

1. National Earthquake Database of Canada
2. U.S. Geological Survey earthquake catalogue
3. Weston Observatory earthquake catalogue

The 2012 NUREG-2115 (Reference 2.7-21) contains data collected through mid-2009. Expected moment magnitudes were determined for the added earthquakes as described in 2022 PSHA NK054-REP-03500.8-00001 (Reference 2.7-41).

Figure 2.7.4.3-1 depicts the spatial distribution of earthquakes in the updated de-clustered catalogue exclusively in the time window between 2008 and December 31, 2021, as described in the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference-2.7-41).

The maximum magnitude (M_{\max}) distributions for the distributed seismicity sources (seismicity source zones) were obtained using the project earthquake catalogue and the methodology developed in NUREG-2115 (Reference 2.7-21). The project earthquake catalogue was also used to obtain updated earthquake recurrence assessments for the seismic sources

2.7.4.4 Seismic Source Characterization

The seismic source zonation model used in the 2022 DNNP PSHA NK54-REP-03500.8-00001 (Reference 2.7-41) is that presented in the 2021 Darlington PSHA NK38-REP-03611-10041 R003 (Reference 2.7-11) with the exception of updates to the 2020 Geological Survey of Canada historical seismicity zonation (H model) based on Adams, et al. (Reference 2.7-19). The seismic source characterization model comprises regions of distributed seismicity and local sources representing identified geological/geophysical features. An overview of the information in the 2022 DNNP PSHA (Reference 2.7-41) with respect to the regional and local seismic sources is summarized in the following paragraphs.

2.7.4.4.1 Regional Source Zones

Three alternative approaches to regional seismic zonation are used to represent the sources of distributed seismicity throughout the study region. Figure 2.7.4.4-1 presents the logic tree structure used in the 2022 DNNP PSHA (Reference 2.7-41) as well as previous PSHA studies in the Darlington Nuclear site area, representing the epistemic uncertainty in regional seismic source zonation. The three alternative approaches are used to define the source zonation for distributed seismicity sources as follows:

1. The favored approach (weight 0.8) was to define source zones on the basis of seismotectonic evaluations. Epistemic uncertainty in defining the boundaries between these seismotectonic sources led to the set of alternative zonations.
2. An alternative approach (weight 0.1) was to use the historical seismicity zonation developed by the Geological Survey of Canada as part of the Canadian National Earthquakes Hazards Program (Adams, J., et al., 2019) (Reference 2.7-19) These regional Seismicity Zones are shown in Figure 2.7.4.4-2.
3. The third alternative was to use a zoneless model (weight of 0.1) in which seismicity parameters were defined for individual cells comprising 1 degree longitude by 1 degree latitude within the study region shown in Figure 2.7.4.4-3.

2.7.4.4.2 Local Source Zones

There are six potential local seismic source zones that are defined based on their identified geological/geophysical features, per the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference 2.7-41). These six source zones are: Clarendon-Linden Fault System, Georgian Bay Linear Zone, Hamilton-Presqu'île Fault, Mississauga Magnetic Domain, Niagara-Pickering Linear Zone, and Wilson-Port Hope Magnetic Lineament. These sources act as potential concentrators of seismic activity and are critically assessed for their seismogenic potential. The locations of these sources have been extracted from the 2022 DNNP PSHA (Reference 2.7-41) and are depicted in Figure 2.7.4.4-4.

2.7.4.5 Ground Motion Characterization

Ground motion models are needed to calculate the effects at the site of earthquakes occurring in the characterized seismic sources. Two aspects are considered as follows:

1. Estimation of the amplitude of ground motions as a function of earthquake size and the source-to-site distance that is provided by ground motion models
2. Assessment of the effect of the local site conditions on the generic hard rock ground motions by results of site response analyses performed in a manner that achieves hazard-consistent ground motions at the site surface

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In active tectonic environments, ground motion models are often developed from the analysis of recorded strong motion data. The seismic hazard was computed using the Pacific Earthquake Engineering Research Center model documented in the 2018 PEER Report No. 2018/08 by Goulet, C., et al. (Reference 2.7-22). The model is the most comprehensive ground motion model available for Central and Eastern United States (CEUS) Seismic Source Characterization.

2.7.4.6 PSHA Results for the DNNP Site

The 2022 PSHA study in NK054-REP-03500.8-00001 (Reference 2.7-41) presents the seismic hazard characterization for the deeply embedded BWRX-300 RB at the DNNP site. The study meets the requirements and follows the guidance of CNSC REGDOC-2.5.2 (Reference 2.7-32), CSA N289 Series (Reference 2.7-31, 2.7-32, and 2.7-33), and the Licensing Topical Report NEDO-33914-A (Reference 2.7-27).

The PSHA presented in the 1994 EPRI TR-102261-V1 (Reference 2.7-12) and the 2021 Darlington Risk Assessment (Reference 2.7-11), developed Uniform Hazard Response Spectra (UHRS) for rock outcropping motions at the anticipated level of the foundation of the DNGS plant at the top of the Paleozoic bedrock strata. The DNGS foundation level is not at the same elevation as the foundation of the BWRX-300 deeply embedded RB.

The seismic hazard model used in the 2022 DNNP PSHA (Reference 2.7-41) is based on the seismic hazard model employed in the 2021 Darlington Nuclear site PSHA (Reference 2.7-11) and is updated using new data and information. Differences between the two seismic hazard models, overall, are minor and include:

- Recalculated earthquake recurrence parameters, such as rates, maximum magnitude (M_{\max}), and spatial distributions, using the updated earthquake catalogue
- Slight increase in the probability that the Wilson-Port Hope local source is associated with small magnitude earthquake, resulting from additional earthquakes being recorded in the region (this produces a minor increase in the overall probability of activity for this source)
- The source zone geometry for zonation based on historical seismicity is updated to be consistent with the 6th Generation of seismic hazard maps of Canada, H2 model for source zonation, per Adams, J., et al. (Reference 2.7-19).

The approach to site-specific hazard differs between the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference 2.7-41) and the 2021 Darlington site PSHA NK38-REP-03611-10041 (Reference 2.7-11). In the 2021 Darlington site PSHA (Reference 2.7-11) site-specific hazard results were obtained solely for the reactor basemat elevation for the existing DNGS using the two options for application of the EPRI 2006 Cumulative Absolute Velocity (CAV) model specified in USNRC (2012a) (Reference 2.7-35):

- Option 1 specified computing the hazard integrating from a minimum magnitude of M 5 (M 4 was used) but only applying the CAV filter to the contributions from magnitudes less than M 5.5
- Option 2 specified computing the hazard integrating from a minimum magnitude of M 5 without applying the CAV filter. Deterministic site amplification functions from reference rock were computed using a site profile truncated at the reactor foundation elevation.

Epistemic uncertainty in site amplification scaling reference rock motions to foundation level motions was incorporated into the CAV calculations but aleatory variability in amplification was not included. Vertical motions were obtained by applying mean V/H ratios to the horizontal UHRS.

2.7.4.6.1 Site Response Analysis

Site-specific hazard in the 2022 DNNP PSHA (Reference 2.7-41) is computed only using USNRC (2012a) (Reference 2.7-35) Option 2, integration of hazard from M 5 without applying the EPRI (2006) (Reference 2.7-36) CAV filter. Site amplification was computed using NUREG/CR-6728 Approach 3, per McGuire et al., in 2001 (Reference 2.7-20). This approach develops the SRA in which probabilistic site amplification functions defining both median amplification and aleatory variability in amplification were convolved with the reference rock hazard to produce site-specific hazard at the target elevations. Epistemic uncertainty in site amplification was modeled.

The site response model was extended to finish grade to represent anticipated as-built site conditions with reactor basemat elevation for the planned BWRX-300 is located approximately 12 m below the top of rock at the DNNP site while the reactor foundation levels at the existing DNGS site are at or near top of rock. Minimum epistemic uncertainty in site amplification was applied in both studies, with the updated value used for the DNNP study being 50 percent larger than the value used in the 2021 NK38-REP-03611-10041 (Reference 2.7-11). Seismic hazard results for vertical motions were computed by convolving probabilistic V/H ratios with the horizontal hazard rather than applying mean V/H ratios.

Per guidance of NEDO 33914-A (Reference 2.7-27), Section 5.2.2, the site-specific hazard is defined for the following three horizons at:

- The RB foundation bottom elevation 52.93 m CGD
- The soil/rock interface elevation 64 m CGD
- The finished grade elevation 88 m CGD

There are only slight differences between the reference rock and site-specific hazard curves at the RB base and soil/rock interface as presented in the 2022 DNNP PSHA report (Reference 2.7-41). The horizontal mean hazard curves were interpolated to obtain UHRS for an Annual Frequency of Exceedance (AFE) of 1E-2, 1E-3, 1E-4, 1E-5, 1E-6, and 1E-7 for the RB base, soil / rock interface, and finished grade elevations, respectively. The results of the UHRS curves at the horizontal and vertical of the three targeted horizons are provided in Figure 2.7.4.6-1 through Figure 2.7.4.6-8.

Seismic hazard results were produced in the 2022 DNNP PSHA report NK054-REP-03500.8-00001 (Reference 2.7-41) for Design Basis Earthquake (DBE) seismic inputs to design and Beyond Design Basis Earthquake (BDBE) seismic inputs for the evaluations of the Design Extension Conditions (DEC) as per REGDOC-2.5.2 (Reference 2.7-32) and to the Checking Level Earthquake as per CSA N289.1:18 (Reference 2.7-33). Section 9.2 of NK054-REP-03500.8-00001 (Reference 2.7-41) describes the development of DBE and BDBE ground motion response spectra. Figure 2.7.4.6-9 through Figure 2.7.4.6-11 compare the DBE and BDBE horizontal ground motion spectra with the corresponding UHRS with 1E-4 and 1E-5 AFE for the three elevations mentioned above.

Subsection 2.7.5.3.5.1 presents the DBE and BDBE response spectra that define the amplitude and frequency content of the DBE and BDBE ground motion. The DBE horizontal ground motion spectra meet the minimum earthquake requirement by enveloping the CSA N289.3 minimum spectrum as shown in Figure 2.7.4.6-12 and Figure 2.7.4.6-13.

Subsection 2.7.5.2.5.2 presents the hazard-consistent, strain-compatible dynamic soil properties used as input for the seismic response analysis and design of BWRX-300 RB that were also developed for both the DBE and BDBE levels of motion using the results of the site response analyses.

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Augmentations were applied to the DBE and BDBE RB base motions, as described in Section 9.4 of NK054-REP-03500.8-00001 (Reference 2.7-41), to produce foundation input response spectra which meet the requirements of the 2010 USNRC DC/COL ISG-017 (Reference 2.7-37) for hazard consistency of foundation input response spectra for SSI analyses following guidance of NEDO 33914-A, Section 5.3.4.1. Finally, sets of recorded ground motions were recommended for use as seed motion in developing time histories for seismic analyses.

Table 2.7-4 identifies the figures which present UHRS based on the mean hazard results, reproduced from the 2022 PSHA NK054-REP-03500-.8-00001 (Reference 2.7-41).

Table 2.7-4: Figures Presenting UHRS Based on Mean Hazard Results.

Elevation (m CGD)	Orientation	Figure
52.93	Horizontal	2.7.4.6-1
52.93	Vertical	2.7.4.6-2
64	Horizontal	2.7.4.6-3
64	Vertical	2.7.4.6-4
88	Horizontal	2.7.4.6-5
88	Vertical	2.7.4.6-6
Reference Rock	Horizontal	2.7.4.6-7
Reference Rock	Vertical	2.7.4.6-8
52.93	Horizontal DBE and BDBE	2.7.6.4-9
64	Horizontal DBE and BDBE	2.7.6.4-10
88	Horizontal DBE and BEBE	2.7.6.4-11

2.7.4.7 Potential Seismically Related Hazards

Several geological hazards and seismicity-related phenomena that could potentially affect the suitability of the DNNP site and the plant design are evaluated.

2.7.4.7.1 *Volcanism*

A methodology for initial investigation of volcanism suggests evaluating within a 150 km radius of the site, per the 2009 DNNP Flood Hazard Assessment NK054-REP-01210-00012 (Reference 2.7-23). The methodology states that if there is no evidence of Cenozoic era (i.e., within the last 65 million years), volcanic rocks or volcanism in the region, no further investigations are required. Geological Map 1860a from Natural Resources Canada in the 2009 DNNP NK054-REP-01210-00012 (Reference 2.7-23) does not identify Cenozoic era formations within 150 km of the site. Hence, volcanism at the DNNP site is considered an improbable hazard with no associated seismic activity.

2.7.4.7.2 *Tsunami*

Tsunamis are long period gravity waves generated in oceans or lakes by seismic disturbances or landslides resulting in a sudden displacement of the water surface. The resulting wave energy spreads across the ocean or lake at high speed. Tsunami occurrences in Canada are rare, with the Pacific Coast at greatest risk due to the higher occurrence rate of earthquake and landslide activity. The magnitude 7.2 Grand Banks earthquake of 1929 produced tsunami effects on the Burin Peninsula of Newfoundland. The Great Lakes are on the edge of the Canadian Shield, a

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geologically stable, mid-continental region where the rate of occurrence of earthquakes is about one tenth of that at tectonic plate boundaries.

The Lake Ontario shorelines are not generally susceptible to shore slope failure or landslide. Review of U.S. National Geophysical Data Center Lake Ontario bathymetry gave no evidence of submarine landslides or other surface disturbance in the post-glacial period, per the 2009 NK054-REP-01210-00012 (Reference 2.7-23). Around the perimeter of Lake Ontario, "Quaternary sediments are relatively thin or absent, and bedrock exposures are common, possibly reflecting the effects of sub-glacial erosion and subsequent abrasion by lacustrine waves and currents."

The Natural Hazards Database at the U.S. National Geophysical Data Center reports one 1755 "tsunami run-up event" in Lake Ontario, though this appears to have been a seiche-like event. The event, for a location about 50 km northwest of Rochester, N.Y. is coded as "an event that only caused a seiche or disturbance in an inland river", source "unknown." "In Lake Ontario the water repeatedly rose in an unusual way to the height of about 1.5 m, no shock is mentioned. Exact latitude and longitude are unknown."

In the absence of tsunami reports in Lake Ontario and the lack of shoreline or lakebed evidence of tsunami initiators, tsunamis are considered improbable events with no associated flood hazard potential at the site.

2.7.4.7.3 Seiches

Storm surge and seiche effects in Lake Ontario resulting for various scenario storms were considered in the 2009 NK054-REP-01210-00012 (Reference 2.7-23). The maximum storm-induced surge and seiche at the Darlington shore is 0.75 m. The 1755 event where 1.5 m high seiche-like oscillations in Lake Ontario were reported may not have been seismically-induced as no shock is mentioned. A review of historical earthquake records in the 2009 DNNP PSHA NK054-REP-01210-00014 DNNP PSHA (Reference 2.7-4) identified an event on January 9th, 1847, in Grafton Harbour where with "Lake Ontario calm under a north wind, suddenly the lake level descended, exposing the lakebed for upwards of about 107 m". In moments it recoiled, rushing towards the shore in one unbroken wave about 1.2 m above normal. This wave accompanied by a heavy noise crashed over the wharf and washed inland about 91 m. This happened about 8 or 9 times, each with "diminishing force." The editor of the Cobourg Star reminded his readers that something similar had occurred in Cobourg and Port Hope in 1845. An apparently related report described "some commotion" at Rice Lake about 19 km north of Grafton Harbour, during which the 0.46 m of ice on Rice Lake began "to undulate". Eventually the ice burst with "a noise like thunder" and chunks in the center of the lake were tossed into a pile about 3.1 m high. These reports do not mention ground shaking, although noise is mentioned.

Based on the historical evidence, seiche events have occurred in Lake Ontario; therefore, shoreline protection at DNNP is considered in the design as discussed in Subsection 2.4.2.

2.7.4.7.4 Dams and Landslides

There are no human-made water retaining structures within the Darlington Creek watershed or other site vicinity watersheds, as described in the 2009 Flood Hazard Assessment NK054-REP-01210-00012 (Reference 2.7-23). Hence, there are no flooding hazards associated with seismically-induced failure of human-made water retaining structures. Additionally, the flooding threat due to seismically-induced landslide at the site is minimal, per the 2009 NK054-REP-01210-00012 (Reference 2.7-23). These conclusions are validated in Section 5.6 of the 2022 DNNP Site Evaluation Update Summary Report NK054-REP-01210-00142 (Reference 2.7-43).

2.7.4.7.5 Surface Faulting

At present, there is no known evidence of larger, pre-historic earthquakes that have resulted in surface fault rupture because such earthquakes have not occurred, or the evidence for surface rupture or coseismic damage is not preserved, or the studies needed to identify past large earthquakes is insufficient to recognize these events.

Given the relatively stable geological setting of the region surrounding the Darlington site, the recency of the post-glacial landscape that might preserve past large earthquake effects, it is expected that evidence for large earthquakes if they have occurred, would be difficult to identify. The 2022 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.7-39) confirms the absence of historical evidence for surface rupture within the Darlington site, including any absence of faults within the boreholes as logged.

The onshore and offshore boreholes and mapping of the DNGS excavations did not indicate offsets in the stratigraphic units, shear zones, or deep depressions in the bedrock surface, hence no near surface faulting has occurred in the bedrock at the site, as described in the 1981 DNGS Geotechnical Mapping of Bedrock Excavation NK38-02004P (Reference 2.7-24). There is no evidence of post-glacial fault-related scarps in the overburden or of solution-weathered cavities in the bedrock, as reported in the 1977 DNGS Geology and Seismicity - Hydro Geotechnical Engineering Dept. Report 77110 (Reference 2.7-25).

The stratigraphic continuity of the upper Paleozoic bedrock in the site vicinity conformed to the regional dip of about 5 m/km to the south. Minor changes in thickness and position of marker units were evident, but the differences were well within the limits of variation expected for sedimentary rock formations in southern Ontario. No vertical dislocation or displacement was evident in the upper Paleozoic bedrock formations, indicating that faulting has not propagated through the sedimentary rock strata from the Precambrian basement rock.

Mapping of marker units in the DNGS intake and discharge tunnels that extend over 1 km south of the site showed continuity consistent with the regional dip. Jointing in the rock is tight and water ingress is insignificant.

Regional geologic maps, e.g., Ontario Geological Survey, 1991, indicate that the Paleozoic rocks are, with few exceptions, relatively flat-lying and laterally continuous, indicating that no large-scale, major faulting has occurred in the region since they were deposited.

The 2022 DNNP Geotechnical Investigation NK054-REP-01210-00175 (Reference 2.7-39) reaffirmed the conclusions from the 2009 investigations and it is concluded that there is no evidence of surface faulting in the overburden or bedrock at the site or site vicinity.

2.7.4.7.6 Liquefaction Potential of Foundations

The RB foundation is to be founded on sound limestone bedrock. Foundations of other structures are to be founded on dense to very dense till deposits, and/or engineered fill. As such, the liquefaction potential of foundations will be low.

The 2022 DNNP Liquefaction Assessment Report NK054-REP-03500.8-00002 (Reference 2.7-42) assessed seismically-induced liquefaction hazards of foundation soils for the DNNP to support the Licence to Construct (LTC) application. The assessment considered the latest seismic hazard values reported in the 2022 DNNP PSHA NK054-REP-03500.8-00001 DNNP PSHA (Reference 2.7-41). The detailed liquefaction assessment of foundation soils was performed for the structures No. 1 to No. 6, namely, (1) RB, (2) TB, (3) RWB, (4) CB, (5) Reactor Auxiliary Bay, and (6) Independent Spent Fuel Storage Installation (ISFSI), as labelled in Figure 2.7.4.7-1 and Figure 2.7.4.7-2. In addition, for the potential Emergency Mitigating Equipment (EME) Access Routes at the site, all boreholes within the project boundary were evaluated for liquefaction potential.

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The following conclusions were derived from the results of the liquefaction assessment (Reference 2.7-42).

- For the DBE event, foundation soil liquefaction is not expected for the structures within the power block including the RB, TB, RWB, CB, and Reactor Auxiliary Bay, based on available soil data and the plan for the power block area to be over-excavated approximately to elevation 81 m CGD and then backfilled to plant grade elevation of 88 m CGD. For the foundation soil below the structures No.2 to No.5, the estimated seismically-induced settlement is typically less than 5 mm with a maximum of 8 mm, and the seismically-induced lateral displacement is expected to be up to 28 mm under DBE event
- For the DBE event, soil in the vicinity of the ISFSI structure (Structure No. 6) is expected to experience liquefaction, particularly in the surficial glaciolacustrine deposit (Unit 2 from the expected finished grade at elevation 88 m down to about 5 m depth). The estimated seismically-induced settlement is up to 154 mm and the lateral spreading displacement is up to 1.67 m.
- For the BDBE event, foundation soil liquefaction is not expected for the following structures:
 - RB (Structure No. 1), founded directly on bedrock
 - TB (Structure No. 2)
 - RWB (Structure No. 3)
- For the BDBE event, liquefaction potential exists at only one data point (isolated and limited extent of zones) for foundation soils in the vicinity of the following structures:
 - CB (Structure No. 4) - The liquefaction data point is at about elevation 69.1 m CGD, about 18.9 m depth from the finished grade.
 - Reactor Auxiliary Bay (Structure No. 5) - The liquefaction data point is at about elevation 69.9 m CGD, about 18.1 m depth from the finished grade.
- For the BDBE event, the foundation soil of the structures No.2 and No.5 in the power block area is calculated to have typically less than 17 mm and up to 27 mm of seismically-induced settlement, and the displacement due to lateral spreading that is calculated to be typically less than about 0.05 m and up to about 0.09 m displacement, as per the detailed liquefaction assessment of the available geotechnical data.
- For the BDBE event, significant liquefaction and seismically-induced deformation is expected in the vicinity of the proposed location for the ISFSI structure (Structure No.6).
- For the EME access routes, liquefaction susceptibility and screening assessment was performed considering all boreholes (forty-eight in total) at the site except for those within the power block area. Figure 2.7.4.7-1 and Figure 2.7.4.7-2 show the locations of the boreholes which are susceptible to liquefaction for the DBE and BDBE events respectively.

In Section 7.2 of the 2022 DNNP Geotechnical Investigation Report (Reference 2.7-39), it is indicated the upper clayey, sandy, and silty deposits (i.e., Units 2a and 2b) are potentially liquefiable during the 10,000-year design earthquake event. However, approximately 8 m of soil will be removed from beneath the power block and replaced by engineered fill. Excavating the aforementioned soil units by the specified 8 m will mitigate the potential for liquefaction. It is therefore concluded that the soil under the power block is considered non-liquefiable under the 10,000-year design earthquake for the RB, TB, RWB, CB, and the Reactor Auxiliary Bay.

2.7.5 Geotechnical and Seismological Parameters

Subsection 2.7.5 describes the site-specific information used for developing the geotechnical and seismological parameters for the in-situ site conditions prior to construction of and the anticipated as-built conditions after the construction of the BWRX-300 facility. The in-situ conditions are characterized based on the information described in Subsection 2.7.3, including the results reported in the 2022 NK054-REP-01210-00175 Phase I Geotechnical Investigation Report (Reference 2.7-39) and the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

Subsection 2.7.5 is divided into the following subsections:

- Subsection 2.7.5.1: Assessment of As-Built Conditions at the DNNP Site, including a description of the over-excavation and fill replacement, evaluation of bearing capacity and time-dependent deformation for the proposed foundations, and evaluation of the anticipated earth pressure on structures.
- Subsection 2.7.5.2: Geotechnical and Seismological Site Properties, including subgrade stratigraphic profiles, static and dynamic properties of rock and soil; and groundwater level
- Subsection 2.7.5.3: Geotechnical Variability and Uncertainty, including potential sampling bias, inherent variability of samples and possible measurements errors consideration, including the main source of epistemic and aleatory uncertainties

2.7.5.1 Assessment of As-Built Conditions at DNNP Site

The site geotechnical investigations, presented in Subsection 2.7.3, are used to characterize the stratigraphy of subsurface materials at the area of the DNNP site where the first BWRX-300 unit is to be constructed. The data collected from the 2022 geophysical investigations NK054-REP-01210-00175 (Reference 2.7-39) provide comprehensive understanding of the subsurface soil and the deep bedrock conditions at the site.

The DNNP site subsurface soil and rock profiles are presented in Subsection 2.7.3.2. The DNNP site consists of approximately 25 m of soil deposits overlaying bedrock. Both the soil and bedrock materials are characterized as flat laying to slightly dipping toward the south. The top and surficial soil deposits may not have the required capacity to support the near surface mounted foundations of the BWRX-300 RWB, TB, CB and Reactor Auxiliary Bay (refer to Chapter 1, Figure A1.1-2, Figure A1.4-1 and Figure A1.5-1 for site and BWRX-300 Unit 1 layouts). Bearing capacity and settlement confirmatory calculations were performed, as part of the 2022 geotechnical work NK054-REP-01210-00175 (Reference 2.7-39) and the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38), considering approximate dimensions, bearing pressure demands and stratigraphy of the soil materials under the RWB, TB, and CB and the Reactor Auxiliary Bay foundations.

The results of the geotechnical investigations that are reported in the 2012 NK054-REF-01210-0418696 (Reference 2.7-28), the 2013 NK054-REP-01210-00098 (Reference 2.7-29), and the 2022 geotechnical investigations and tests (Reference 2.7-39) do not indicate the presence of rock cavities, voids, large open fractures, significant eroded zones, shear zones, or joint configurations that would have a potential for causing rock instability and thus jeopardizing the integrity or the safety functions of the deeply embedded BWRX-300 RB.

2.7.5.1.1 Over-excavation and Fill Replacement

The range of SPT blow count numbers (as low as 6) and laboratory tests results indicate that the topsoil and fill materials may contain organic clays and be soft or very loose sands, which is not suitable for supporting the near surface mounted foundations of RWB, TB, CB, and Reactor

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Auxiliary Bay. As described in Subsection 2.7.3.1, beneath the topsoil and fill materials, two layers of surficial lacustrine soil materials that differ in clay content and plasticity were identified:

- The layer at the top (Unit 2a) consists of sandy lean clay to lean clay soil with soft to very stiff consistency
- The layer below (Unit 2b) consists of cohesionless silty gravel to silty sand materials, with compactness varying from very loose to very dense

The SPT blow counts taken for the two surficial lacustrine soil layers (Units 2a and 2b) show low values indicating that these materials may not be suitable for supporting the RWB, TB, CB, and Reactor Auxiliary Bay foundations and may liquefy during a DBE level event. The results of field and laboratory tests performed for the upper till (unit 3), intermediate glaciolacustrine (Units 4a and 4b), and lower till (Unit 5) indicate dense and stiff materials surrounding the deeply embedded RB that have no potential for liquefaction during a DBE event and are suitable for supporting the foundations of the RWB, TB, CB, and other power block structures.

As a result, site preparation for construction of the BWRX-300 SMR is anticipated to include excavation at the power block area of the weaker surficial soils to an elevation between 80 m and 82 m CGD. The excavated surface soils will be replaced with engineered fill to bring the site grade back to elevation 88 m CGD. The dense upper till, intermediate glaciolacustrine and lower till soils below elevations 80 m to 82 m CGD would remain in place. The BWRX-300 RB would then be constructed in a vertical right cylinder shaft excavation that extends to a depth of about 35.2 m or elevation 52.8 m CGD. At this depth, the bottom of deeply embedded BWRX-300 RB is anticipated to extend through the compacted or engineered fill and in-situ soils and into the underlying bedrock.

The RWB, TB, CB, and other power block structures surrounding the deeply embedded RB are anticipated to be supported by shallow foundations on the engineered fill.

Information detailed in the 2021 licensing topical report on BWRX 300 Advanced Civil Construction and Design Approach, NEDO-33914-A (Reference 2.7-27) describes the approach to be used for monitoring the effects of excavation and construction on the properties of subsurface materials; specifically in its Subsection 3.4 Field Instrumentation Plan, and Section 4.0 Foundation Interface Analysis.

2.7.5.1.2 Bearing Capacity Evaluation for Proposed Foundations

2.7.5.1.2.1 Shallow Foundation

As documented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38), based on engineering assessment, conventional spread and strip footings located in the power block area which are founded on engineered fill can be designed using ultimate bearing capacities (q_u):

- 1.0 m wide with depths of 1.3 to 2.5 m: 1857 to 3642 kPa
- 2.0 m wide with depths of 1.3 to 2.5 m: 1854 to 3493 kPa
- 3.0 m wide with depths of 1.3 to 2.5 m: 1834 to 3509 kPa
- 4.0 m wide with depths of 1.3 to 2.5 m: 1854 to 3422 kPa
- 5.0 m wide with depths of 1.3 to 2.5 m: 1891 to 3393 kPa

Raft foundations can be used for heavily loaded structures where conventional spread or strip footings are not adequate to support. Raft foundation founded on engineered fill can be designed for the following ultimate bearing capacities (q_u):

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- 68 x 70 m TB raft foundation with depths of 1.3 to 2.5 m: 5672 to 6917 kPa
- 30 x 48 m RWB raft foundation with depths of 1.3 to 2.5 m: 3986 to 4978 kPa
- A factor of safety of 3.0 is recommended to be used for the service limit state, and a resistance factor of 0.5 is recommended to calculate the ultimate limit state

2.7.5.1.2.2 Reactor Building Deeply Embedded Foundation

The proposed elevation for the RB foundation is at elevation of approximately 53 m CGD, corresponding to a depth of about 35 m below grade. At this elevation/depth, the Lindsay Formation has Rock Quality Designation values ranging from 90% to 100% and discontinuity spacing is considered to be 1 m to 3 m, per the 2022 Power Block geotechnical investigations (Reference 2.7-39).

Considering a mean UCS of 75 MPa and 48 MPa (Reference 2.7-39), the allowable bearing capacity (qa) for the RB is 7.5 MPa and 4.8 MPa, respectively.

For a conservative bearing capacity estimate, using a minimum UCS of 48 MPa and bearing capacity factor (Ksp) of 0.1, an allowable bearing capacity of 4.8 MPa will be used for the Reactor Building foundation design.

2.7.5.1.2.3 Pile Foundation

Pile foundations may also be considered for other heavily loaded power block structures. These structures may be supported on drilled caissons founded on competent undisturbed very dense/hard glacial till (with minimum 1 m embedment) or bedrock (with 1 m embedment recommended) with the over-excavation and backfill for soil deposits above elevation 80 m to 82 m CGD. End-Bearing Caissons founded on native undisturbed lower till deposit (Unit 5) at about 20 m depth can be designed for a factored geotechnical compression resistance 1100 kN. Alternatively, end-bearing caissons advanced to about 25 m depth, at least 1.0 m socket into bedrock (Unit 6a – Blue Mountain Formation), can be designed using a factored geotechnical compression resistance of 620 kN. The ultimate end-bearing resistance in bedrock is estimated to be approximately 20 MPa and a resistance factor of 0.4 is used to calculate the factored geotechnical compression resistance. These will form predominantly end-bearing foundations and therefore larger diameters (minimum 0.76 m in diameter) are recommended. Relatively undisturbed (clean) caisson bases should be ensured prior to concrete placement to minimize any potential settlement under maximum applied loads. The end-bearing caissons with at least 1 m embedment below weathered and/or fractured bedrock is estimated and presented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

Uplift forces of cast-in place concrete caissons will be resisted by the weight of the foundation and friction along its embedment surface area. Estimation of uplift resistance of 1.0 m diameter caissons are presented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

2.7.5.1.3 Earth Pressure

The anticipated earth pressure considering the in-situ stress, ground conditions, soil shoring system, RB stiffness, and loads from surrounding buildings along the depth of the RB has been conservatively evaluated based on results of non-linear FIA, as presented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38) and is displayed in Figure 2.7.5.1.3-1.

The horizontal pressure was found higher in bedrock compared to the soil. This is due to the higher in-situ stress locked in the bedrock as a result of past tectonic activities. The earth pressure at the interface of the RB wall in the bedrock presented in Figure 2.7.5.1.3-1 represents a bounding post-construction stage scenario that assumes no stress release occurs in the bedrock

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during excavation so all the in-situ stresses locked in the rock would be fully transferred to the RB wall. Reinforcement in the bedrock is to be incorporated in updates to the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38) that estimates stress release in the bedrock at the end of the excavation once the rock reinforcement is designed. A field instrumentation plan is to be implemented, per guidance in Section 3.4 of NEDO-33914-A (Reference 2.7-27), to monitor the deformations of the rock during the excavation. These measurements will be used to calibrate the FIA model.

2.7.5.1.4 Time-Dependent Deformation for Proposed Foundations

2.7.5.1.4.1 Elastic Settlement Method

The elastic settlement is presented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38) and summarized in Table 2.7-5

Table 2.7-5 Deformation for Proposed Foundations (Reference 2.7-38)

Building Structures	Structural Bearing Pressure, Upper Bound (kPa)	Proposed Foundation (Width, Depth) (m)	Estimated Elastic Settlement (mm)
Control Building	28.7	Spread footing (3, 1.3)	1
Turbine Building	270	Raft Foundation (68X70, 1.3)	41
	150		23
	80		12
RAD Waste Building	162	Spread Footing (3, 1.3)	5
	162	Raft Foundation (48X30, 1.3)	16
Reactor Auxiliary Bay	36.8	Spread Footing (3, 1.3)	1

The expected settlement of raft foundation was analysed for the non-uniform structural load as documented in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

2.7.5.1.4.2 Consolidation Settlement Method

As detailed in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38), it is anticipated that much of the consolidation settlement occurs in the lean clay deposit (Unit 4b). Given the Over-Consolidation-Ratio for Unit 4b is between 1.8 and 2.2, the lean clay deposit is over consolidated. Since the final effective pressure caused by the structural pressure is estimated to be lower than the pre-consolidation pressure in the deposit, the consolidation settlement is therefore estimated using the reconsolidation index (Cr). Annual secondary (creep) consolidation settlement is negligible.

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The estimated consolidation settlement of different building structures is summarized in Table 2.7-6.

Table 2.7-6 Consolidation Settlement Method (Reference 2.7-38)

Building Structures	Structural Bearing Pressure, Upper Bound (kPa)	Proposed Foundation (Width, Depth) (m)	Estimated Consolidated Settlement (mm)
Control Building	28.7	Spread Footing (3, 1.3)	5
Turbine Building	270	Raft Foundation (68X70, 1.3)	51
	150		31
	80		17
RAD Waste Building	162	Spread Footing (3, 1.3)	9
	162	Raft Foundation (48X30, 1.3)	45
Reactor Auxiliary Bay	162	Spread Footing (3, 1.3)	2

The expected settlement of raft foundation was analysed for the non-uniform structural load. The maximum total settlement (elastic and consolidated settlement) of the TB is approximately 92 mm, and the differential settlement is approximately 61 mm.

The settlement of raft foundations is also dependent on the rigidity of the foundation, homogeneity of the subgrade material and the construction method. Following the guidance of Section 4.0 of NEDO-33914-A (Reference 2.7-27), a 3-D non-linear FIA is to be performed to develop settlement contours of the raft foundations at a later design stage.

2.7.5.2 Geotechnical and Seismological Site Design Parameters

Subsection 2.7.5.2 presents the geotechnical and seismological properties for the seismic and structural analysis, and design, including:

- Subgrade profiles – Subsection
- Equivalent linearized static properties of soil and engineered fill materials – Subsection 2.7.5.2.2
- Equivalent linearized static properties of rock – Subsection 2.7.5.2.3
- Dynamic subgrade properties – Subsection 2.7.5.2.4
- Seismic Design Parameters – Subsection 2.7.5.2.5
- Groundwater Level – Subsection 2.7.5.2.6

2.7.5.2.1 Subgrade Profiles Stratigraphy

The design analyses of the deeply embedded BWRX-300 RB consider subgrade profiles to account for the variations of the soil and rock properties with depth at the DNNP site. The soil profiles represent “as-built” conditions at the DNNP site after construction of the BWRX-300 facility, where the engineered fill replaces the excavated top in-situ upper lacustrine or fill units. The stratigraphy of the as-built subgrade profiles consists of:

- Engineered fill that is for the upper 6 m to 8 m from elevation 80 m to 82 m CGD, as required to the final grade at elevation 88 m CGD.
- In-situ soils consisting of upper till (Unit 3), intermediate glaciolacustrine soils (Units 4a and 4b), and the lower till unit (Unit 5).
- Rock units including Blue Mountain (Unit 6a), Lindsay (Unit 6b), Verulam (Unit 6c), Bobcaygeon, Gull River, Shadow Lake and Genesis formations.

The engineered fill will comprise either commercial crusher run, or pit run granular fill or select excavated material meeting the requirements of engineered fill described under “Planned As-Built Soil Profile” in Subsection 2.7.3.2. Placement of the fill will be controlled based on in-situ testing and monitoring by the geotechnical engineer as described in the 2023 DNNP FIA Report NK054-REP-03500.8-00003 (Reference 2.7-38).

The BWRX-300 RB vertical cylindrical shaft deep excavation is to be extended through the Blue Mountain Formation (Unit 6a) and founded in the Lindsay Formation (Unit 6b). The Gneiss formation – the deepest investigated unit - is taken as the hard rock basement with shear wave velocities that are greater than or equal to 3000 m/s, per the 2012 Field Work – Geology and Geological Evaluation NK054-REF-01210-0418696 (Reference 2.7-28).

The pre-excavation in-situ site stratigraphy for soil layers are presented in Table 2.7-2. The adopted in-situ soil layer thicknesses are based on the 2022 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.7-39).

The stratigraphy of the rock units at the DNNP site including rock formations and thicknesses are presented in Table 2.7-3. The bedrock stratigraphy is based on the discussion presented in Subsection 2.7.3.2. The elevation of top of upper rock unit, the Blue Mountain (Whitby) Formation, considered as “top of rock” is expected to be about 64.2 m CGD with a variability of ± 2 m. The variation in the thickness layer of ± 3 m is based on the results of the 2022 DNNP Geotechnical Investigation reported in NK054-REP-01210-00175 (Reference 2.7-39).

2.7.5.2.2 Equivalent Linearized Static Properties of Soil and Engineered Fill Materials

Upper Bound and Lower Bound equivalent linearized properties representing the pressure of the soil and rock materials under long-term (static) loads are established based on measurements obtained from the different field and laboratory tests executed during the 2022 geotechnical investigation NK054-REP-012010-00175 (Reference 2.7-39). Upper and lower values are directly from the measured values. Further statistical analysis is completed to account for uncertainty as required during detailed design.

The static Elastic Modulus E_{st} values for soil materials are obtained from the results of field and laboratory tests. Initial Tangent Elastic Modulus values for the soil materials are established by Triaxial Compression Testing and Pressuremeter Testing, respectively. Initial Tangent Elastic Modulus is interpreted from consolidated anisotropic drained triaxial testing of reconstituted specimen. This is representative of in-situ conditions where the specimen is consolidated to approximate in-situ vertical effective stress.

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Poisson's ratio ν_{st} values are determined by compression and shear wave velocities measured during triaxial compression testing. Effective friction angle and Coefficient Lateral Earth Pressure at Rest is determined by Triaxial Compression Testing and Pressuremeter Testing during the 2022 geotechnical investigation NK054-REP-01210-000175 (Reference-2.7-39).

A summary of linearized static properties for engineered fill and in-situ soil layers in the as-built profiles are provided in the Table 2.7-7.

2.7.5.2.3 Equivalent Linearized Static Properties of Rock

The 2022 geotechnical investigation NK054-REP-01210-00175 (Reference 2.7-39) studied the linearized static properties of rock on the DNNP site, with focus around the BWRX-300 power block area. The linearized E_{st} and ν_{st} values of the rock masses are evaluated from UCS testing and triaxial compression testing. The intact rock modulus was measured through UCS testing of intact rock samples. The intact rock modulus was then adjusted to evaluate the rock mass deformation modulus by two different methods:

- Evaluation of the Geologic Strength Index (two separate ways) and further calculation
- Directly measured through pressuremeter testing.

Total or bulk unit weight was measured during the uniaxial and triaxial compressive strength testing. Table 2.7-8 presents a summary of linearized static properties for the rock layers.

Table 2.7-7: As-Built Linearized Static Properties for Soil Layers (Reference 2.7-39)

Stratigraphic Unit	Layer Thickness (m)	Total Unit Weight (kN/m ³)	Effective Friction Angle (degrees)		Elastic (Young's) Modulus (MPa)		Estimated Coefficient of Lateral Earth Pressure at rest	
			Ave.	Range	Lower	Upper	Ave.	Range
Unit 2a	0.00 - 5.72	21.5	32 ^(a)	25 – 37 ^(a)	13	49	0.61 ^(a) - 0.67 ^(b)	0.41 – 0.95 ^(a) 0.65-0.68 ^(b)
Unit 2b		non-plastic	34 ^(b)	29 – 41 ^(b)	32	80	0.73 ^(b)	0.49 – 0.91 ^(b)
Unit 3	<1.00 - 13.1	24.3	37 ^(a) 41 ^(b)	36 – 38 ^(a) 31 - 48 ^(b)	31	613	0.69 ^(b)	0.53 – 1.02 ^(b)
Unit 4a	0.00 - 17.7	22.1	40 ^(a) 39 ^(b)	39 – 41 ^(a) 35 - 45 ^(b)	52	600	0.57 ^(b)	0.42 – 0.73 ^(b)
Unit 4b		22.2	30 ^(a) 34 ^(b)	30 ^(a) 29 – 42 ^(b)	136	413	0.83 ^(a) 0.53 ^(b)	0.43 – 1.15 ^(a) 0.34 - 0.70 ^(b)
Unit 5	0.00 – 6.4	23.7	35 ^(a) 31 ^(b)	32-38 ^(a) 26 – 36 ^(b)	110	330	0.58 ^(a) 0.49 ^(b)	0.39 – 0.74 ^(a) 0.39-0.58 ^(b)

(a) From Triaxial Compression Testing

(b) From Pressuremeter Testing

Table 2.7-8: Summary of Linearized Static Rock Properties (Reference 2.7-39)

Stratigraphic Unit	Average Bulk Density (kgN/m ³)	Mean Intact Modulus (GPa)	Rock Mass Deformation Modulus (GPa)		Poisson's Ratio
			GSI	Pressuremeter Tests	
Unit 6a - Blue Mountain Shale / Shale+Limestone / Limestone	2641	26.6	17.9	5.91	0.32/0.28/0.00
Unit 6b – Lindsay Formation Shale / Shale+Limestone / Limestone	2681	43.4	38.7	9.75	0.00/0.22/0.36
Unit 6c – Verulam Formation Shale / Shale+Limestone / Limestone	2679	25.0	22.3	12.29	0.21/0.29/0.25

2.7.5.2.4 Dynamic Subgrade Properties

The measured values for dynamic properties of rock are presented in Table 2.7-9a and Table 2.7-9b. The measured small-strain in-situ soil dynamic properties are listed in Table 2.7-10a and Table 2.7-10b. The compression wave velocities, shear wave velocities for the soil and bedrock rock units are obtained from the measurements during the 2022 geotechnical investigation NK054-REP-01210-00175 (Reference 2.7-39). Poisson's Ratio, Young's Modulus, Shear Modulus and Bulk Modulus are presented as calculated in the 2022 Geotechnical Investigation Report NK054-REP-01210-00175 (Reference 2.7-39).

Table 2.7-9a: Rock Dynamic Properties (Reference 2.7-39)

Stratigraphic Unit	Total Unit Weight (kN/m ³)	Shear Wave Velocity (m/s)		Compression Wave Velocity (m/s)		Poisson's Ratio	
		Average	Range	Average	Range	Average	Range
Unit 6a – Blue Mountain Formation	26.4	2405	1841 - 2953	4283	3073 - 5935	0.26	0.20 - 0.38
Unit 6b -Lindsay Formation	26.6	2640	1934 - 3024	4792	3202 - 5773	0.28	0.27 - 0.30
Unit 6c – Verulam Formation	26.6	2559	2128 - 2801	4570	3772 - 5443	0.27	0.26 - 0.28

Table 2.7-9b: Rock Dynamic Properties (Reference 2.7-39)

Stratigraphic Unit	Dynamic Shear Modulus (MPa)		Dynamic Young's Modulus (MPa)		Dynamic Bulk Modulus (MPa)	
	Average	Range	Average	Range	Average	Range
Unit 6a – Blue Mountain (Whitby) Formation	15320	12772 - 18186	38674	34068 - 45959	0.26	0.20 - 0.38
Unit 6b – Lindsay Formation	18696	17099 - 19458	47931	43539 - 49978	0.28	0.27 - 0.30
Unit 6c – Verulam Formation	17544	16534 - 18041	44614	42363 - 45762	0.27	0.26 - 0.28

Table 2.7-10a: In-situ Soil Dynamic Properties (Reference 2.7-39)

Stratigraphic Unit	Total Unit Weight (kN/m ³)	Shear Wave Velocity (m/s)		Compression Wave Velocity (m/s)		Poisson's Ratio	
		Average	Range	Average	Range	Average	Range
Unit 2a – Surficial Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay	21.5	304	215 - 451	1087	560 - 2200	0.43	0.15 - 0.48
Unit 2b – Surficial Glaciolacustrine Deposits – Silty Clayey Sand to Silty Sand/Sandy Silt	0.00	351	255 - 483	1769	800 - 2200	0.48	0.45 - 0.49
Unit 3 – Upper Till	24.3	489	240 - 705	1845	700 - 2400	0.46	0.42 - 0.48
Unit 4a – Intermediate Glaciolacustrine Deposits – Silty Sand to Sandy Silt	22.1	659	362 - 1078	2107	1600 - 2400	0.44	0.30 - 0.48
Unit 4b – Intermediate Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay	22.2	656	440-994	2118	1800-2400	0.44	0.37-0.47
Unit 5 (Lower Till)	23.7	875	683-1344	2470	2000-3400	0.42	0.24-0.47

Table 2.7-10b: In-situ Soil Dynamic Properties (Reference 2.7-39)

Stratigraphic Unit	Dynamic Shear Modulus (MPa)		Dynamic Young's Modulus (MPa)		Dynamic Bulk Modulus (MPa)	
	Average	Range	Average	Range	Average	Range
Unit 2a – Surficial Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay	198	94-415	550	277-950	2374	446 - 6271
Unit 2b – Surficial Glaciolacustrine Deposits – Silty Clayey Sand to Silty Sand/Sandy Silt	298	181-450	879	525-1331	7304	1658 - 10750
Unit 3 – Upper Till	577	265 - 878	1678	783 - 2489	7340	4066 - 10472
Unit 4a – Intermediate Glaciolacustrine Deposits – Silty Sand to Sandy Silt	1025	454 - 2607	2915	1338 - 6752	8636	5497 - 10914
Unit 4b – Intermediate Glaciolacustrine Deposits – Sandy Lean Clay to Lean Clay	947	432 - 1808	2719	1272 - 4941	8386	6038 - 10869
Unit 5 (Lower Till)	1848	1133 - 3670	5217	3273 - 10435	12180	5093 - 22215

2.7.5.2.5 Seismic Design Parameters

Two sets of seismic design parameters were developed based on the results of the site response analyses described in Subsection 2.7.4.6.1 for DBE seismic design and BDBE DEC seismic evaluations:

- Ground motion spectra defining the amplitude and frequency content of the DBE and BDBE ground motion at the DNNP site
- Hazard-Consistent, Strain-Compatible (HCSC) profiles defining the variation with depth of the dynamic properties of DNNP subgrade materials compatible to the strains generated by DBE and BDBE levels

2.7.5.2.5.1 Ground Motion Spectra

Per guidance of NEDO 33914-A, Section 5.2.2, three sets of response spectra are developed, as described in Section 9.2 of the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference 2.7-41), defining the amplitude and frequency content of the DNNP site-specific DBE and BDBE ground motion:

1. Foundation Input Response Spectra at RB foundation bottom elevation of 52.93 m CGD presented in Figure 2.7.5.2.5-1
2. Performance Based Intermediate Response Spectra at the soil / rock interface elevation 64 m CGD, located 24 m below planned finished grade presented in Figure 2.7.5.2.5-2
3. Performance Based Surface Response Spectra at the finished grade elevation of 88 m CGD presented in Figure 2.7.5.2.5-3

2.7.5.2.5.2 Strain-Compatible Soil Properties

Profiles of HCSC dynamic subgrade properties, needed for the SSI analyses, are developed based on the results from the site response analyses described in Section 2.7.4.6. The profiles defining the variation with depth of subgrade shear wave velocities compatible to the DBE and BDBE strain levels are presented in Figure 2.7.5.2.5-4. The profiles of subgrade compression wave velocities for the DBE and BDBE strain levels are presented in Figure 2.7.5.2.5-5. Figure 2.7.5.2.5-6 presents the subgrade damping profiles representing the dissipation of energy in the subgrade materials for the DBE and BDBE levels. The presented HCSC dynamic subgrade properties are per Section 9.3 (Table 9-40 through Table 9-45) of the 2022 DNNP PSHA NK054-REP-03500.8-00001 (Reference 2.7-41).

2.7.5.2.6 Groundwater Level

The groundwater elevations are listed in Table 3-7 in Volume 2 of 2 of the 2022 Phase 1 Geotechnical Investigation - DNNP (Reference 2.7-39), and is replicated, in part, in Table 2.7-11, which provides samples of groundwater elevation and hydraulic vertical gradient at BH12 area which is located on the western side of the RB perimeter, as shown in Figure 2.7.3.1-6.

Table 2.7-11 Samples of Groundwater Elevation and vertical Hydraulic Gradients, BH12 Area (Reference 2.7-39, Table 3-7)

Date	Groundwater Elevation (m)				Vertical Gradient (m/m)		
	Unit 3 ¹	Unit 4a ²	Unit 5 ³	Unit 6 ⁴	Unit 3 to 4a (down)	Unit 4a to 5 (Down)	Unit 5 to 6 (Down)
	BH12-1	BH12-2	BH12-3	BH14			
29NOV21	85.47	83.89	79.47	77.09	-0.25	-0.71	-0.28
05JAN22	85.72	84.03	82.47	78.56	-0.27	-0.25	-0.46
07FEB22	85.24	83.66	79.18	78.41	-0.25	-0.72	-0.09
17FEB22	85.46	83.81	79.29	78.52	-0.26	-0.73	-0.09

1. Upper Till
2. Surficial Glaciolacustrine Deposits
3. Lower Till
4. Bedrock

Based on the groundwater information at the DNNP site presented in Subsection 2.7.3.2 and Table 2.7-11, an upper bound groundwater level at elevation of 85.74 m CGD (or approximately 86 m CGD) corresponding to a depth of 2.26 m (or approximately 2 m) below the plant grade at elevation 88 m CGD is to be used for design.

2.7.5.3 Geotechnical Variability and Uncertainty

Geotechnical variability and uncertainty are considered in detail in the 2022 Geotechnical Investigation NK054-REP-01210-00175 (Reference 2.7-39).

When sampling the soil and rock there can be sampling bias that is introduced in the sample selection process. In general, DNNP project samples were selected based on predetermined testing requirements for each borehole and samples were selected from a variety of depths within each borehole. In some cases, such as the shale from the Blue Mountain Formation, it is not possible to test the weaker rock as intact samples of this material cannot properly be prepared for testing (typically breaking apart along weaker bedding planes). In these cases, sensitivity analysis and engineering judgement are required during design to account for the fact that the range in the data may not capture the minimum values.

When in-situ and laboratory methods are used to measure soil and rock attributes, the inherent variability along with measurement error typically led to data scatter. Measurement error may result from equipment errors and procedural or operator errors. Measurement error is minimized through equipment calibration, standardized procedures, laboratory accreditation, etc.

In-situ and laboratory methods are also subject to statistical uncertainty, which may be reduced by increasing the sampling frequency. Further, certain in-situ and laboratory measurements are transformed for design purposes through empirical or other correlation methods.

Geotechnical variability and uncertainty are addressed using a two-pronged approach:

- Reduction in uncertainty through the use of reliable, calibrated equipment, precision in measurement and testing procedures and sufficient quantity of sampling/testing.

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- Consideration of total variability associated with each geotechnical property/parameter, including evaluation of statistical parameters and identification of sources of uncertainty particular to each property/parameter.

2.7.6 References

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Legend

- OPG Ownership
- Hydro One Switchyard
- Building Footprint
- Contour Line - 2m Interval*
- Protected Area Fence
- Utility Power Lines
- Tower
- Waterbody
- Infrastructure
- Road
- Railway

OPG NEW NUCLEAR AT DARLINGTON EXISTING TOPOGRAPHIC DRAWING FOR ILLUSTRATION PURPOSES

ONTARIOPOWER Real Estate Services

DATE 2019/12/12 **D. COYE** **NAD 1983 CSRS**

PROJECT 1-10.000 **NEWNUCDEV** **UTM 17N**

DRAWN BY N. BRYAN **DATE** 2019/12/12 **SCALE** 1:10,000

PROJECT NO. 1-10.000 **PROJECT NAME** NEW NUCLEAR DEVELOPMENT

PROJECT LOCATION 48°58'00" N 76°58'00" W

PROJECT DESCRIPTION This drawing is a topographic drawing of the OPG New Nuclear Station at Darlington, showing existing infrastructure, proposed facilities, and surrounding terrain. The drawing includes a legend, a north arrow, a scale bar, and a title block.

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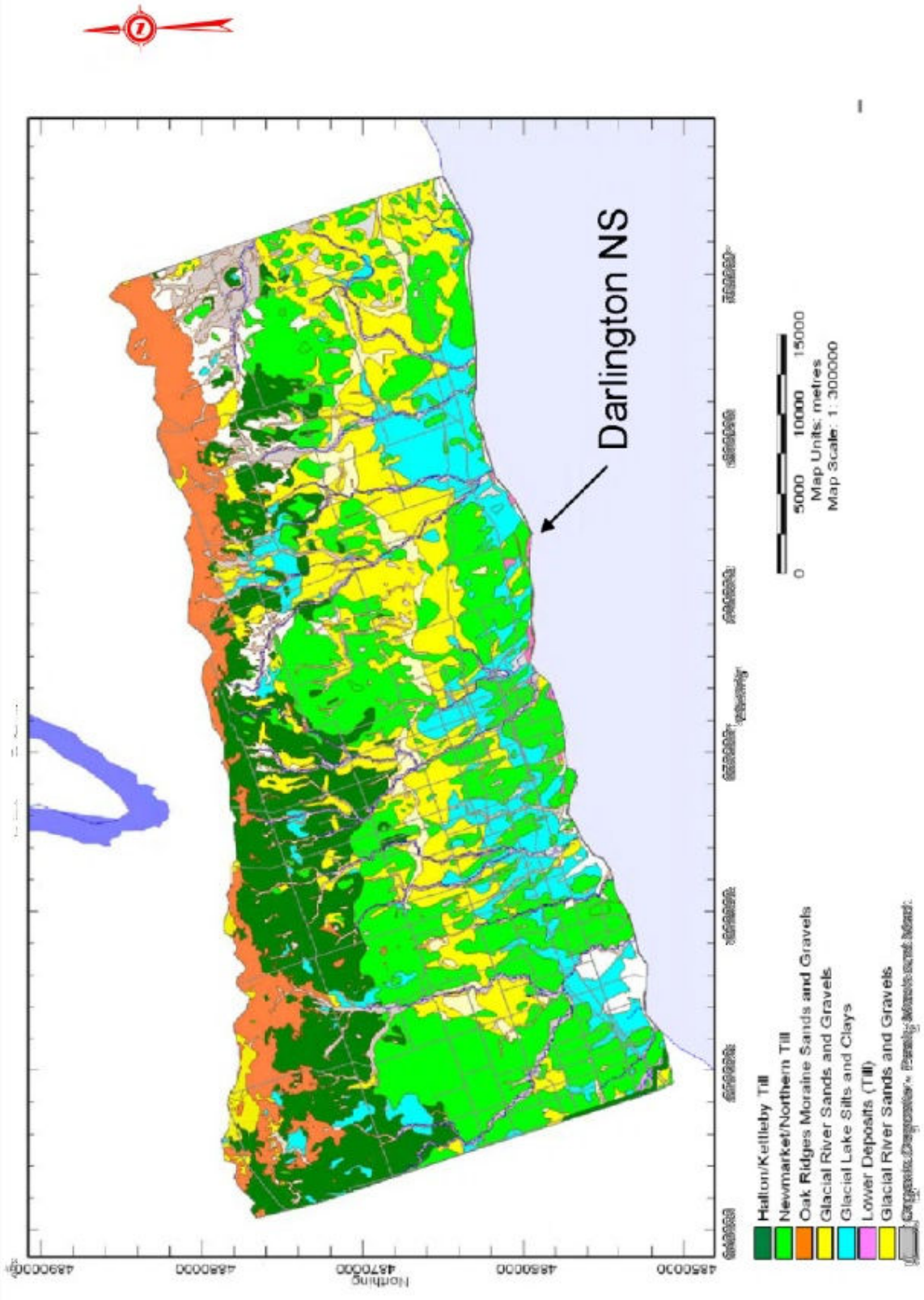


Figure 2.7.2-1: Darlington Nuclear Site - Regional Surficial Geology (Reference 2.7-1)

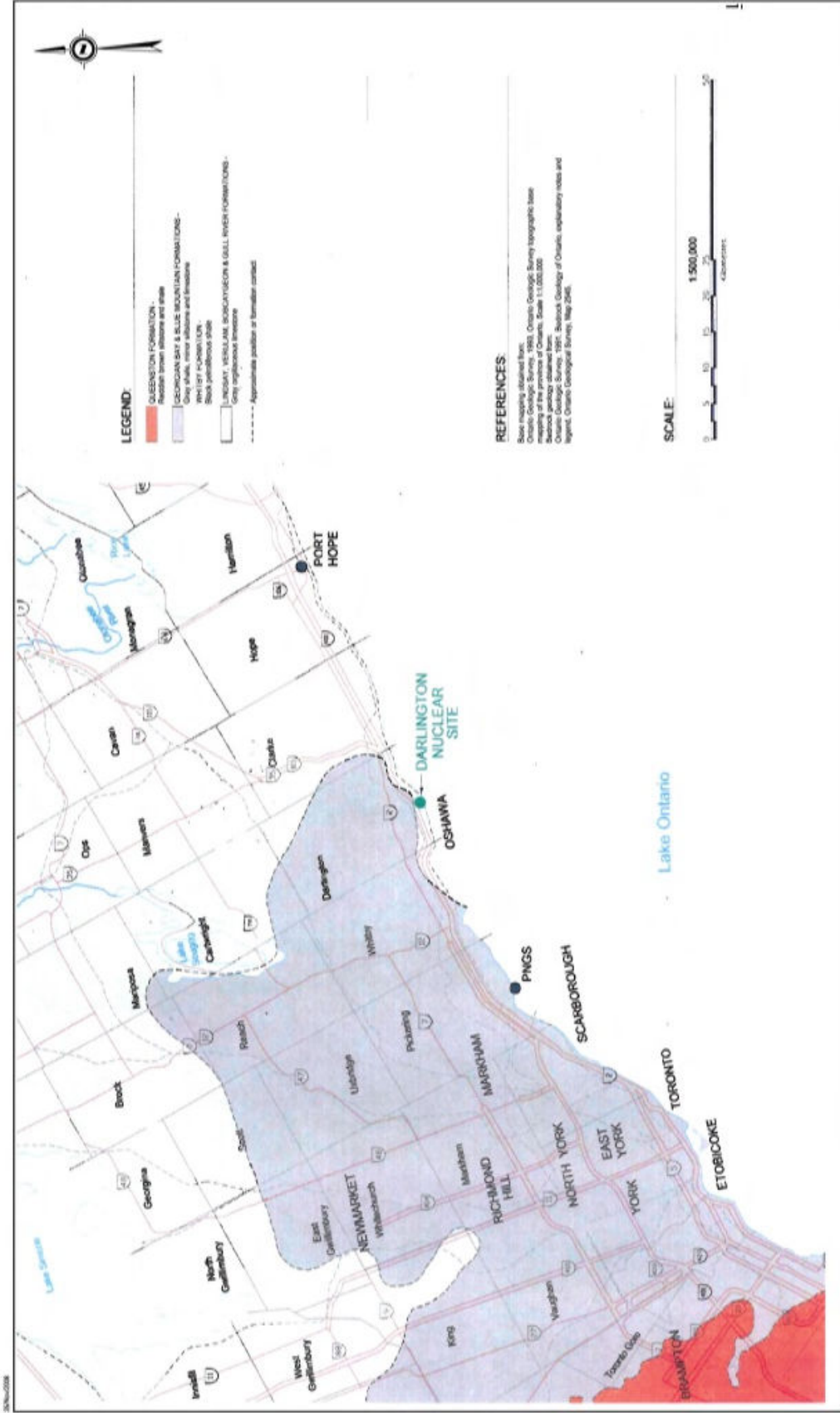


Figure 2.7.2-2: Darlington Nuclear Site - Regional Bedrock Geology (Reference 2.7-1)

The figure displays a topographic map of the study area, showing elevation contours and a key map of the project location. The map is oriented with North at the top. The vertical axis is labeled 'NORTHING (m)' and ranges from 4857600 to 4860200. The horizontal axis is labeled 'EASTING (m)' and ranges from 683600 to 686000. The map shows a large, irregularly shaped area outlined in yellow, representing the study area. The terrain is characterized by steep slopes and a network of roads and trails. A key map in the bottom right corner shows the project location relative to the town of Bowmanville and Lake Ontario.

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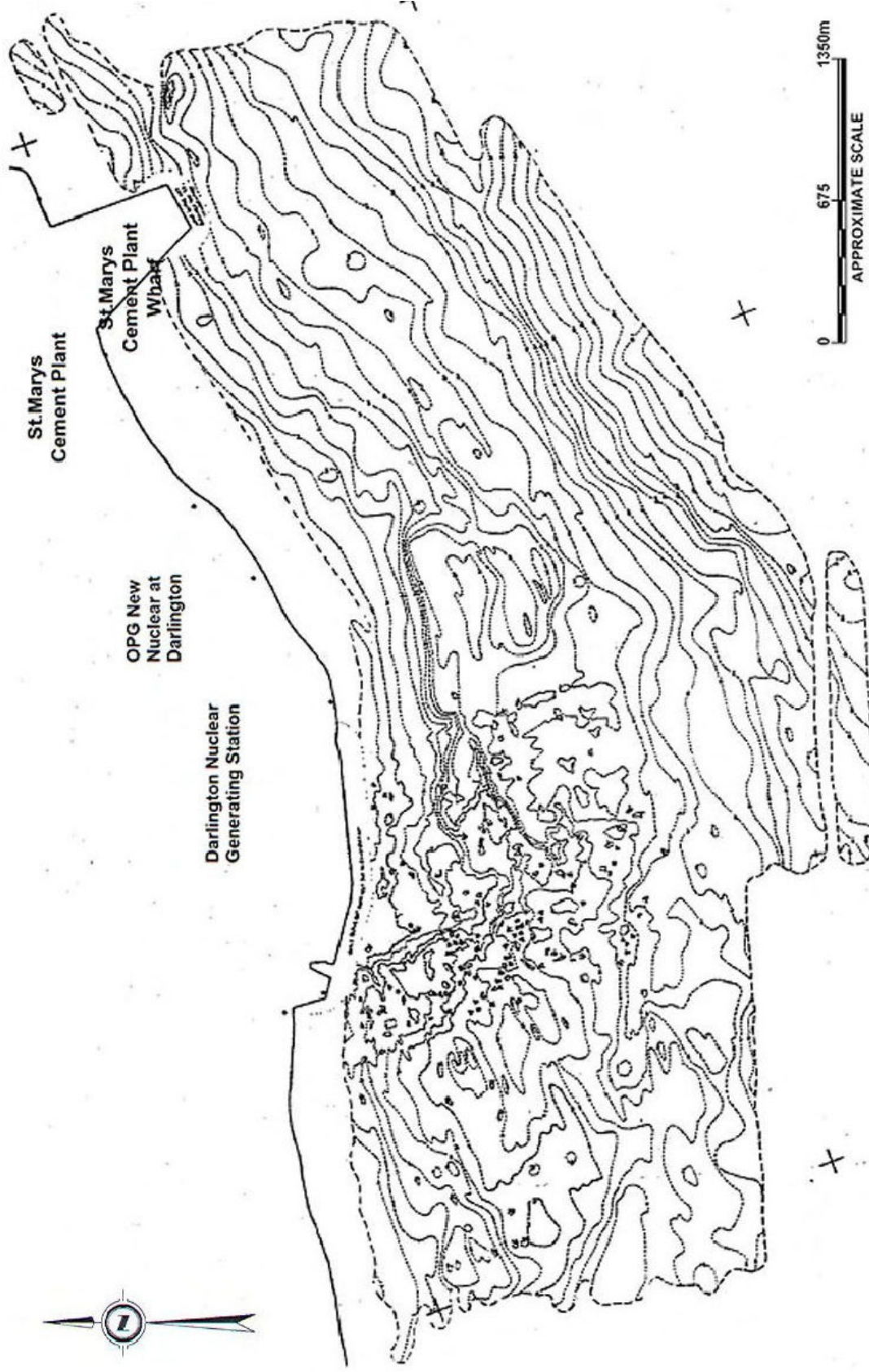


Figure 2.7.2-5: Lakebed Bathymetric Contours along DNNP Site's Shoreline (Reference 2.7-2)

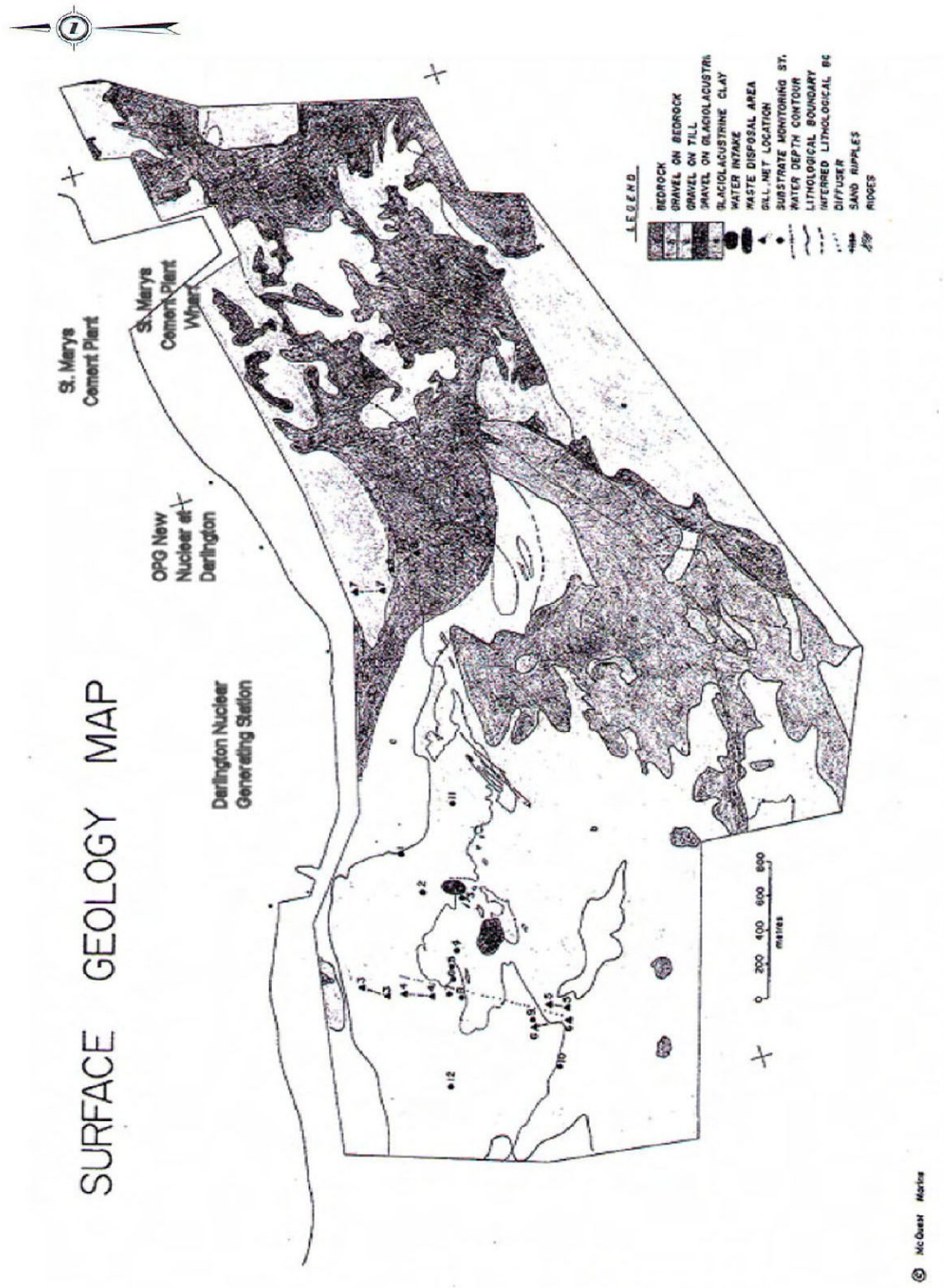


Figure 2.7.2-6: Lakebed Surface Geology Map along DNNP Site's Shoreline (Reference 2.7-2)

Figure 2.7.3.1-1: Locations of CH2MHILL (2009) Monitoring Wells/Boreholes (Reference 2.7-1)

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Figure 2.7.3.1-2: Locations of AMEC (2012) Boreholes (Reference 2.7-35)

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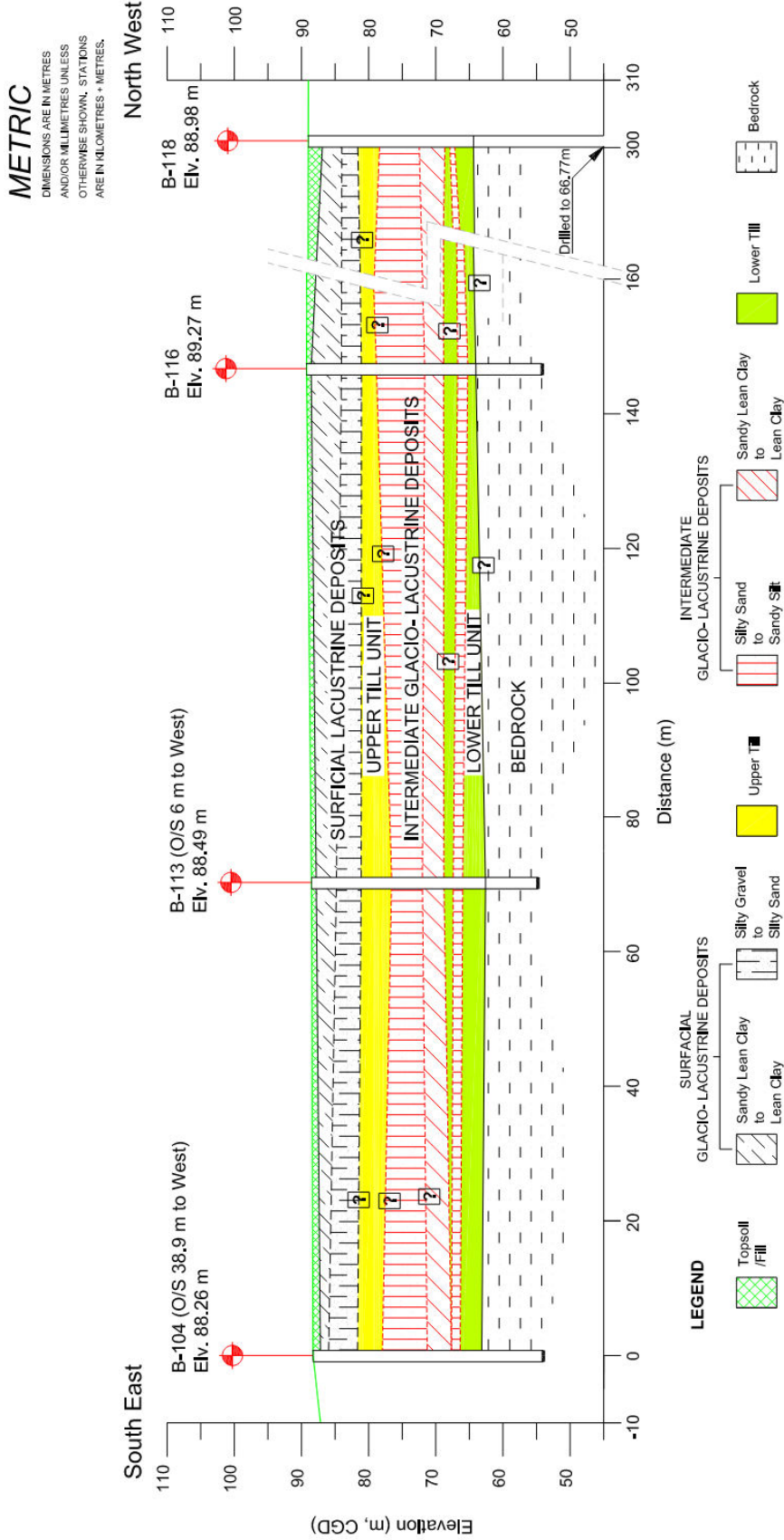


Figure 2.7.3.1-4: Stratigraphic Profile near BWRX-300 Location (Reference 2.7-36)

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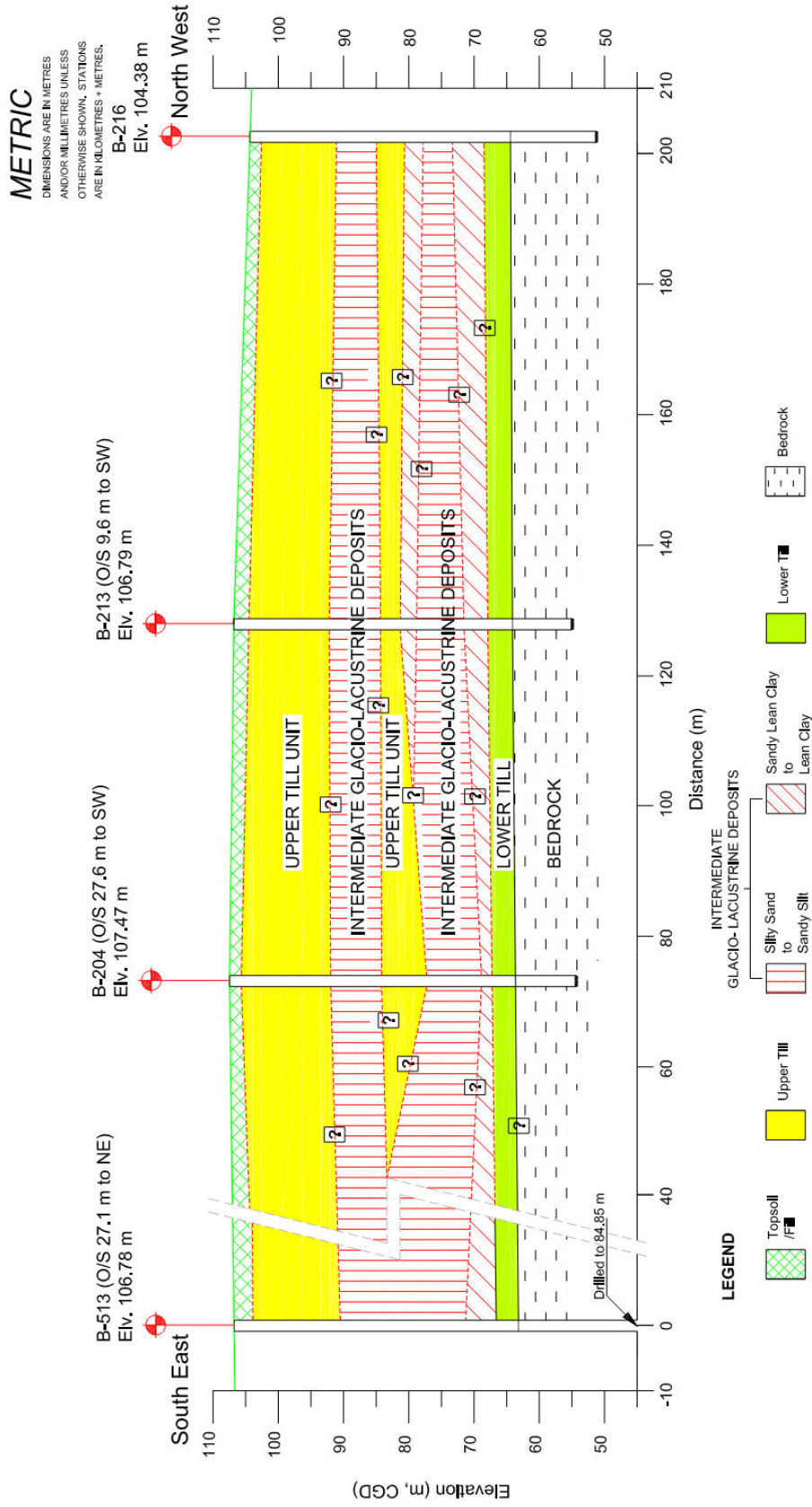


Figure 2.7.3.1-5: Stratigraphic Profile near Topographic Ridge East of the BWRX-300 Location (Reference 2.7-36)

The figure is a detailed geotechnical investigation plan map. It shows a large area with various monitoring wells and boreholes. The wells are labeled with codes such as MW044-34, MW045-10, MW066-8, MW065-25, MW066-29, MW066-44, MW066-45, MW066-46, MW066-47, MW066-48, MW066-49, MW066-50, MW066-51, MW066-52, MW066-53, MW066-54, MW066-55, MW066-56, MW066-57, MW066-58, MW066-59, MW066-60, MW066-61, MW066-62, MW066-63, MW066-64, MW066-65, MW066-66, MW066-67, MW066-68, MW066-69, MW066-70, MW066-71, MW066-72, MW066-73, MW066-74, MW066-75, MW066-76, MW066-77, MW066-78, MW066-79, MW066-80, MW066-81, MW066-82, MW066-83, MW066-84, MW066-85, MW066-86, MW066-87, MW066-88, MW066-89, MW066-90, MW066-91, MW066-92, MW066-93, MW066-94, MW066-95, MW066-96, MW066-97, MW066-98, MW066-99, MW066-100. The boreholes are labeled with codes such as BH01, BH02, BH03, BH04, BH05, BH06, BH07, BH08, BH09, BH10, BH11, BH12, BH13, BH14, BH15, BH16, BH17, BH18, BH19, BH20, BH21, BH22, BH23, BH24, BH25, BH26, BH27, BH28, BH29, BH30, BH31, BH32, BH33, BH34, BH35, BH36, BH37, BH38, BH39, BH40, BH41, BH42, BH43, BH44, BH45, BH46, BH47, BH48, BH49, BH50, BH51, BH52, BH53, BH54, BH55, BH56, BH57, BH58, BH59, BH60, BH61, BH62, BH63, BH64, BH65, BH66, BH67, BH68, BH69, BH70, BH71, BH72, BH73, BH74, BH75, BH76, BH77, BH78, BH79, BH80, BH81, BH82, BH83, BH84, BH85, BH86, BH87, BH88, BH89, BH90, BH91, BH92, BH93, BH94, BH95, BH96, BH97, BH98, BH99, BH100. The map also shows a project location near Lake Ontario, with a scale bar indicating 0 to 200 meters. A legend in the top right corner defines the symbols used for monitoring wells, boreholes, and other features. The map is titled 'SECTION LOCATION PLAN (PHASE 1)' and is part of a larger project titled 'NEW NUCLEAR PROJECT, BOWMANVILLE, ONTARIO'.

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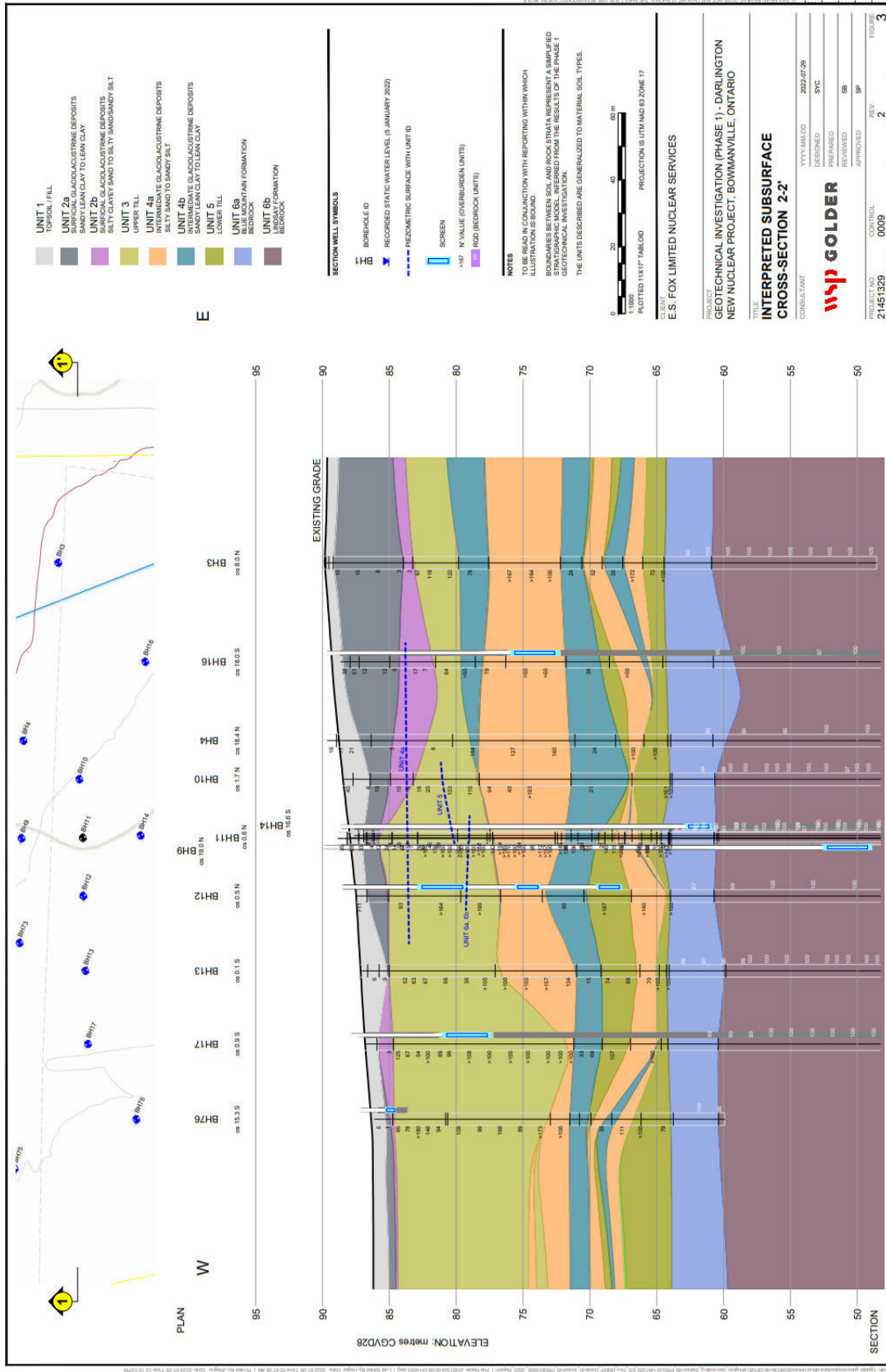


Figure 2.7.3.2-1: Subsurface Stratigraphic Profile at Cross-Section 1-1 (Reference 2.7-39)

Figure 2.7.3.2-2: Subsurface Stratigraphic Profile at Cross-Section 2-2 (Reference 2.7-39)

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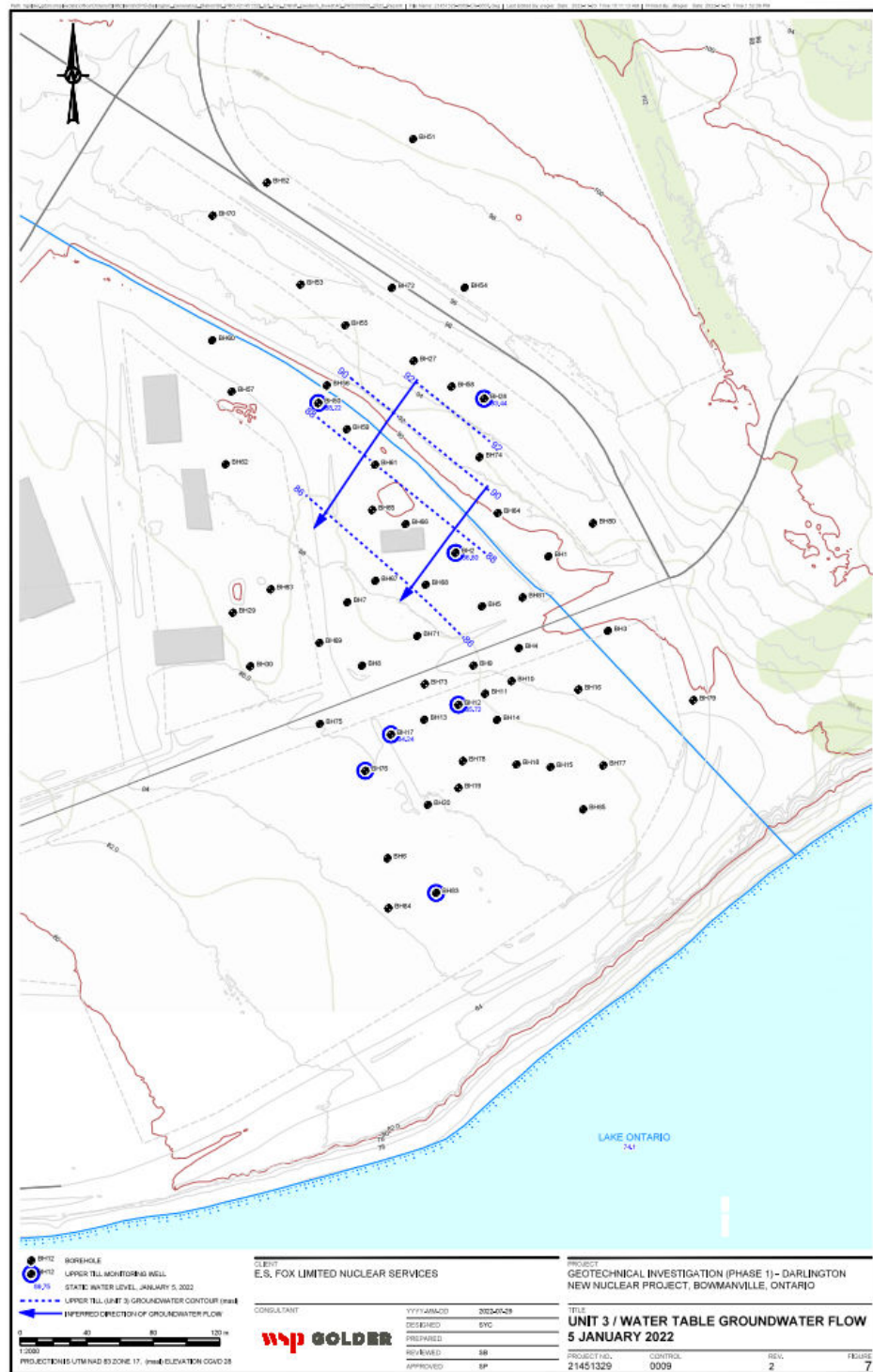


Figure 2.7.3.2-3: Unit 3 Groundwater Levels – Shallow/Water Table (Reference 2.7-39)

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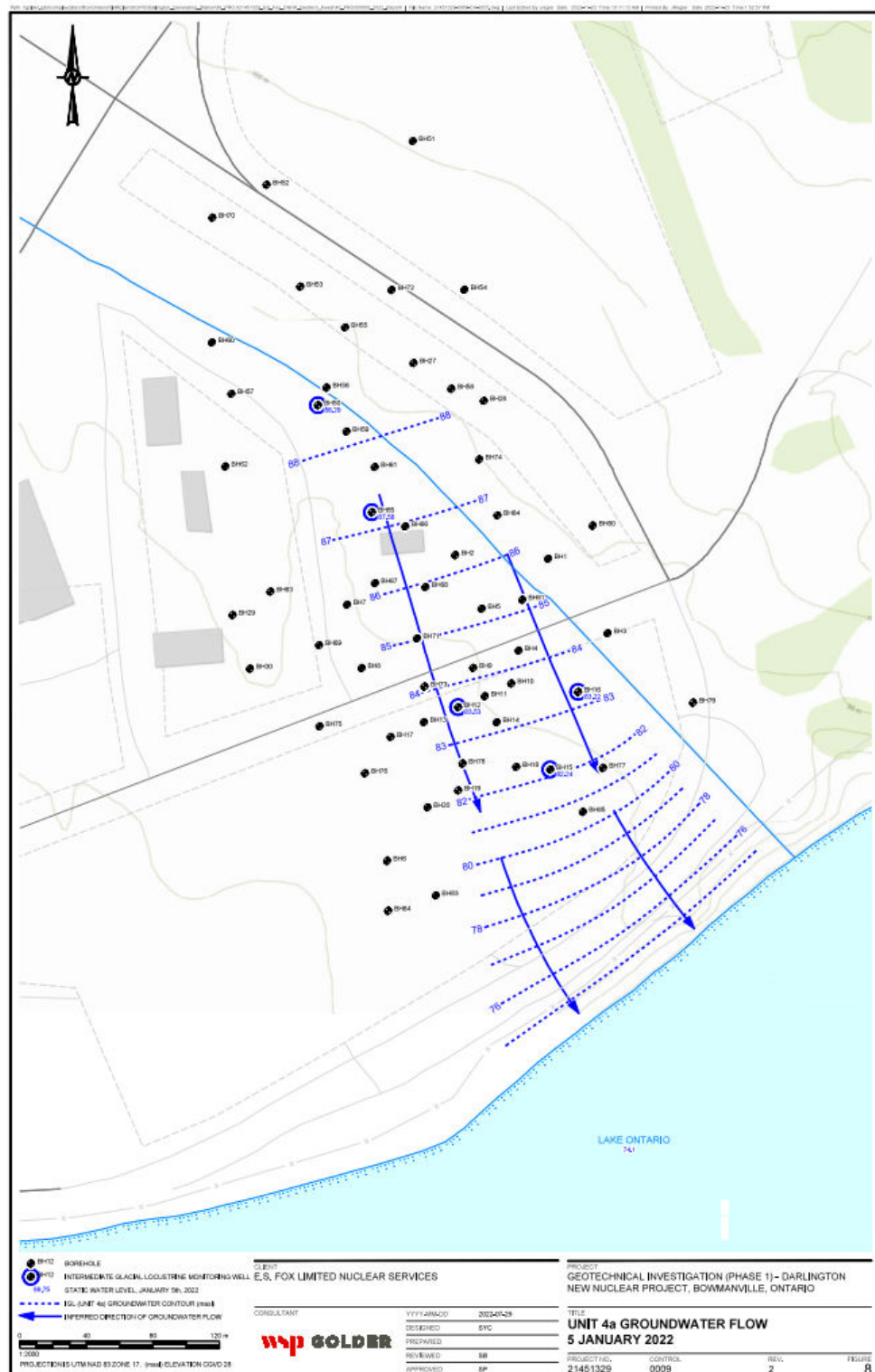


Figure 2.7.3.2-4: Unit 4a Groundwater Flow – Inter-glacial Deposits (Reference 2.7-39)

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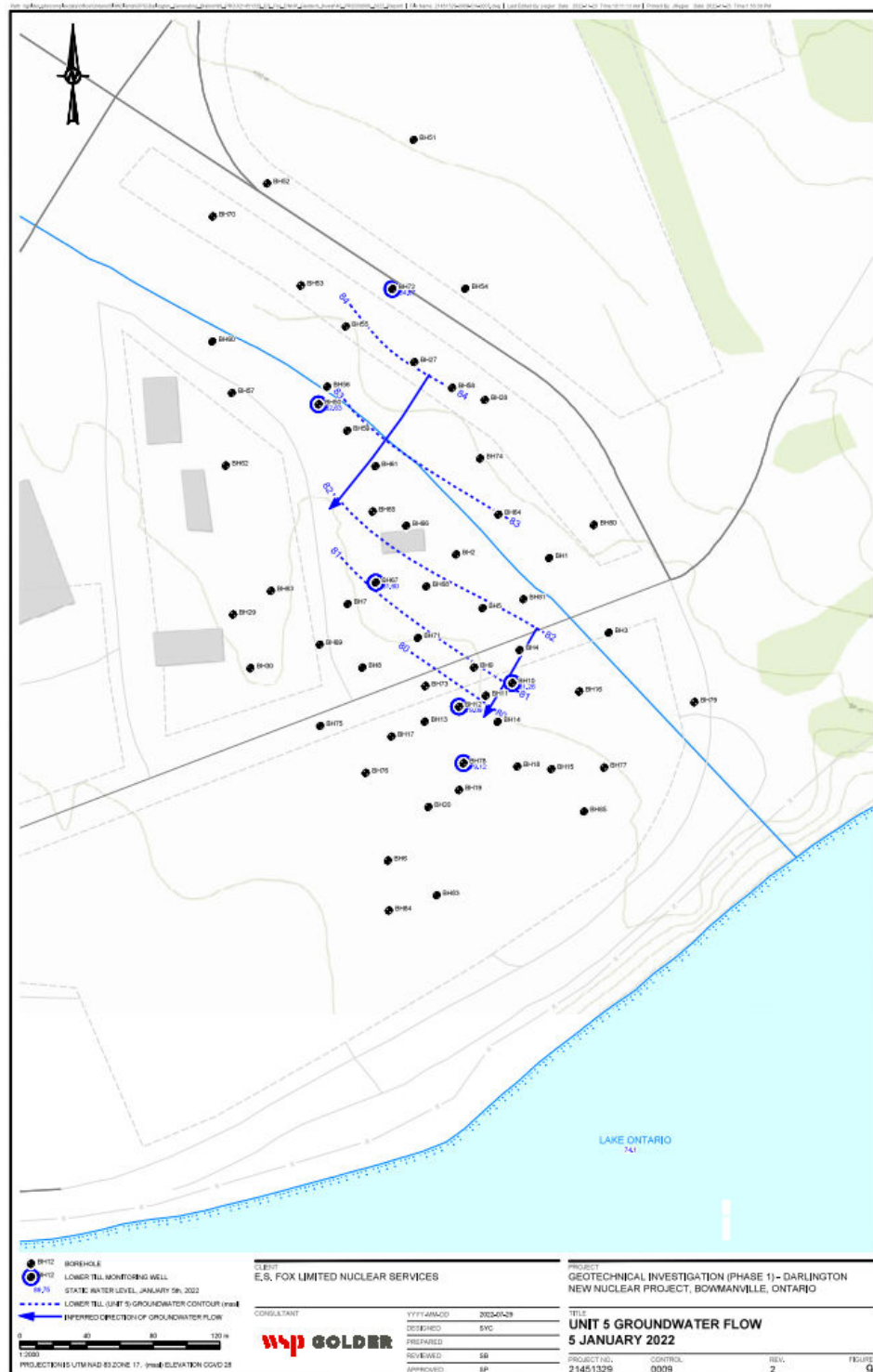


Figure 2.7.3.2-5: Unit 5 Groundwater Flow – Shallow Bedrock (Reference 2.7-39)

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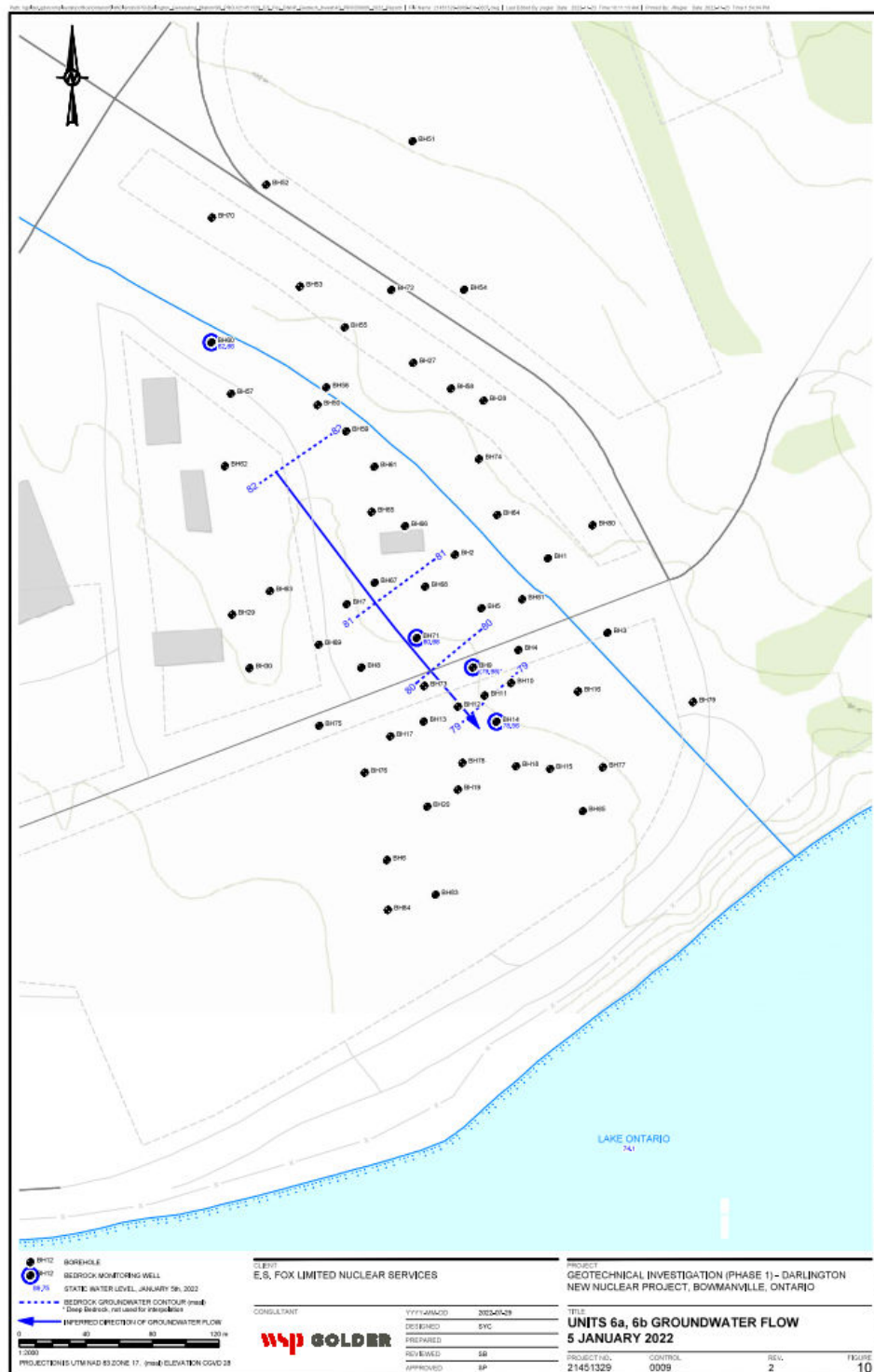


Figure 2.7.3.2-6: Units 6a - 6b Groundwater Flow – Shallow Bedrock (Reference 2.7-39)

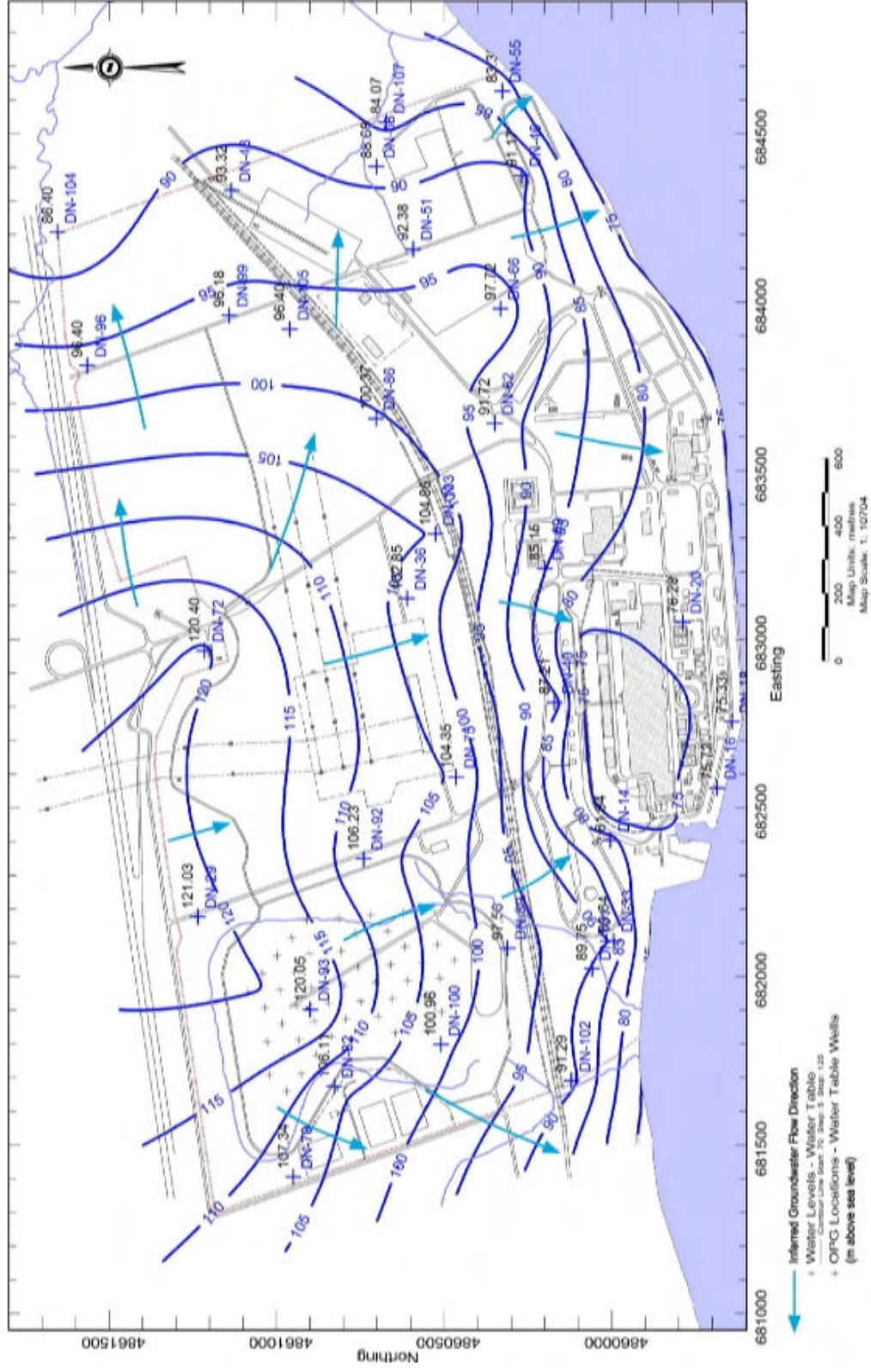


Figure 2.7.3.2-7: Regional Groundwater Levels – Shallow/Water Table (Reference 2.7-1)

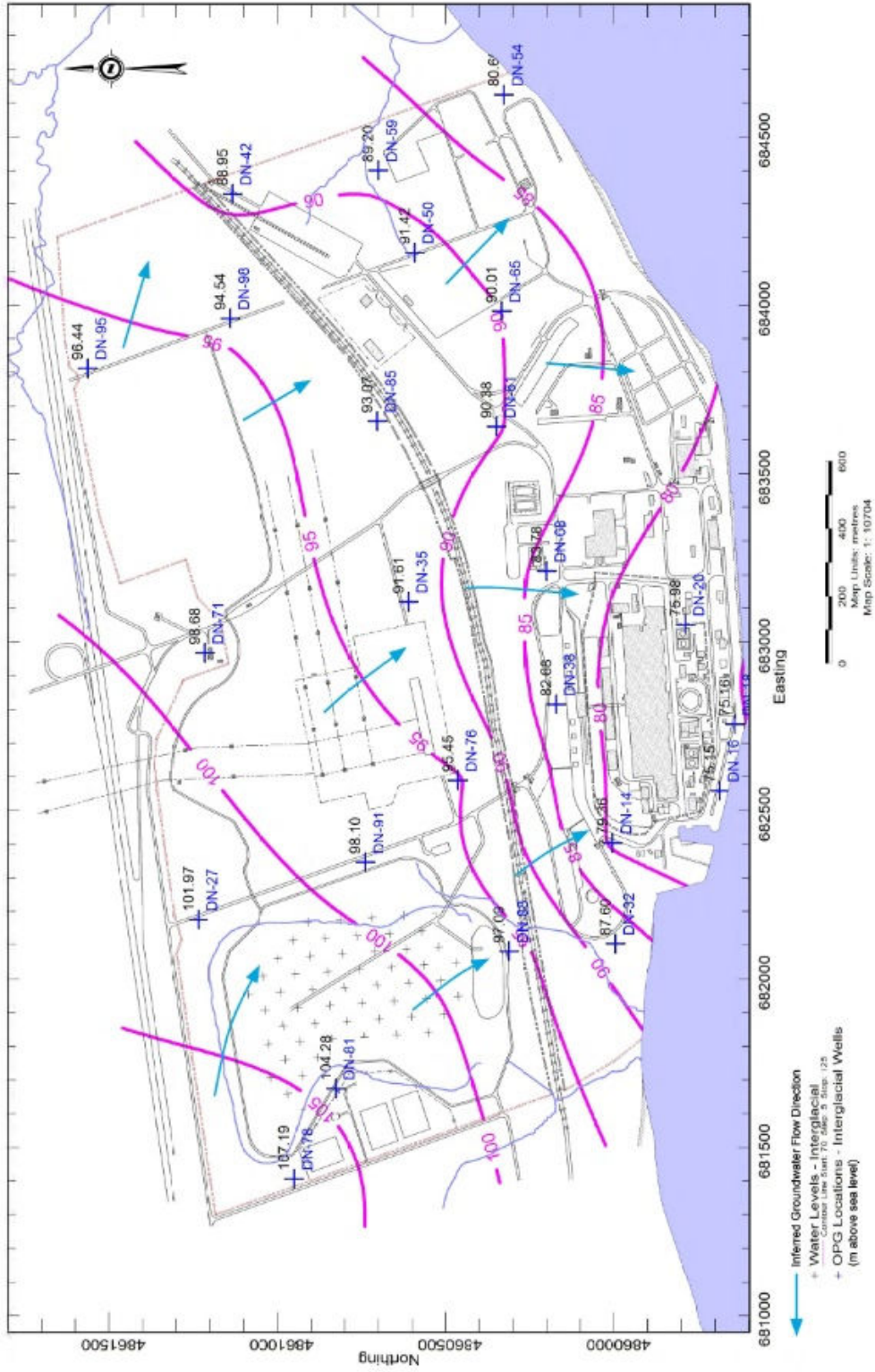


Figure 2.7.3.2-8: Regional Groundwater Flow - Interglacial Deposits (Reference 2.7-1)

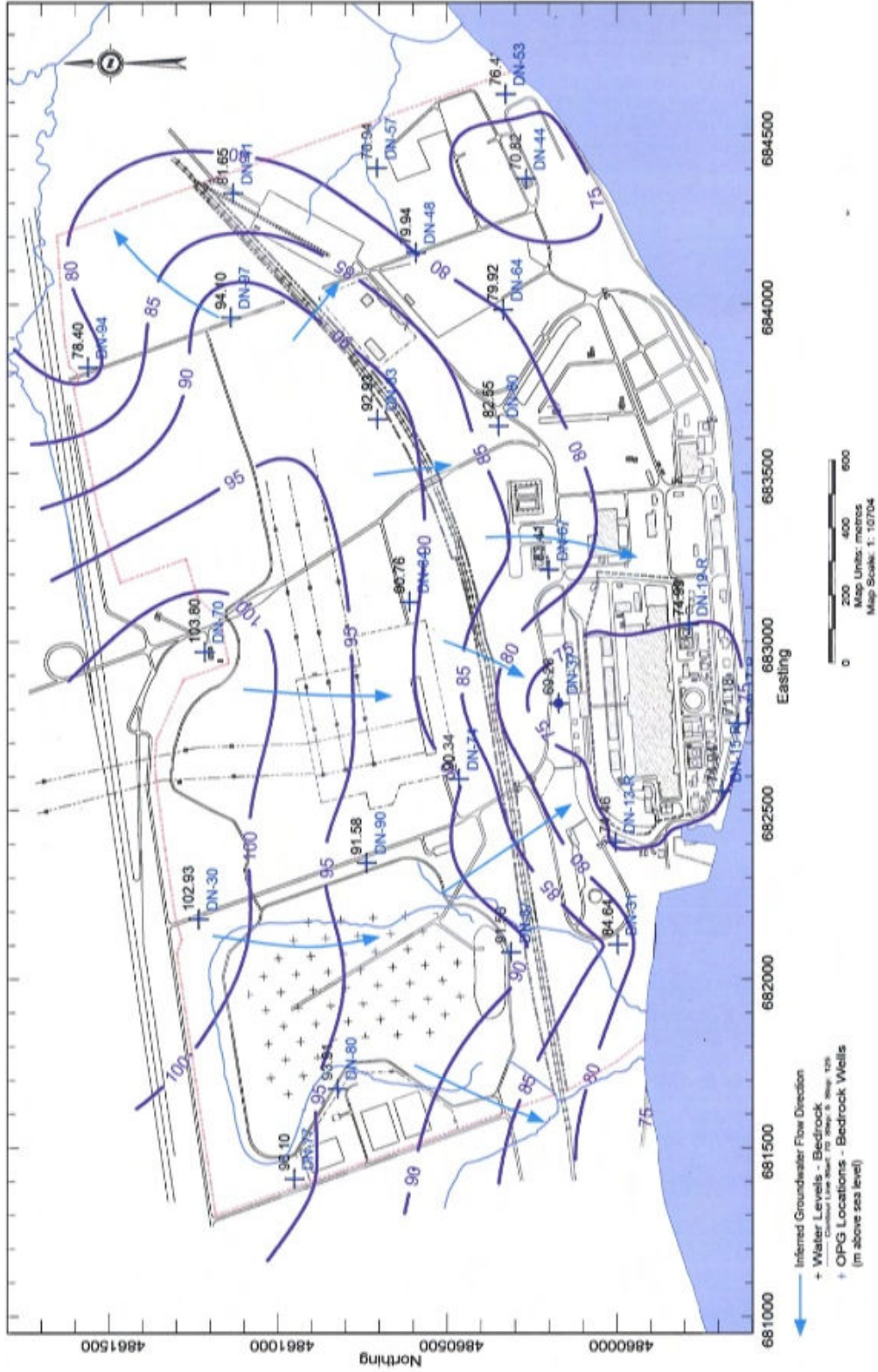


Figure 2.7.3.2-9: Regional Groundwater Flow – Shallow Bedrock (Reference 2.7-1)

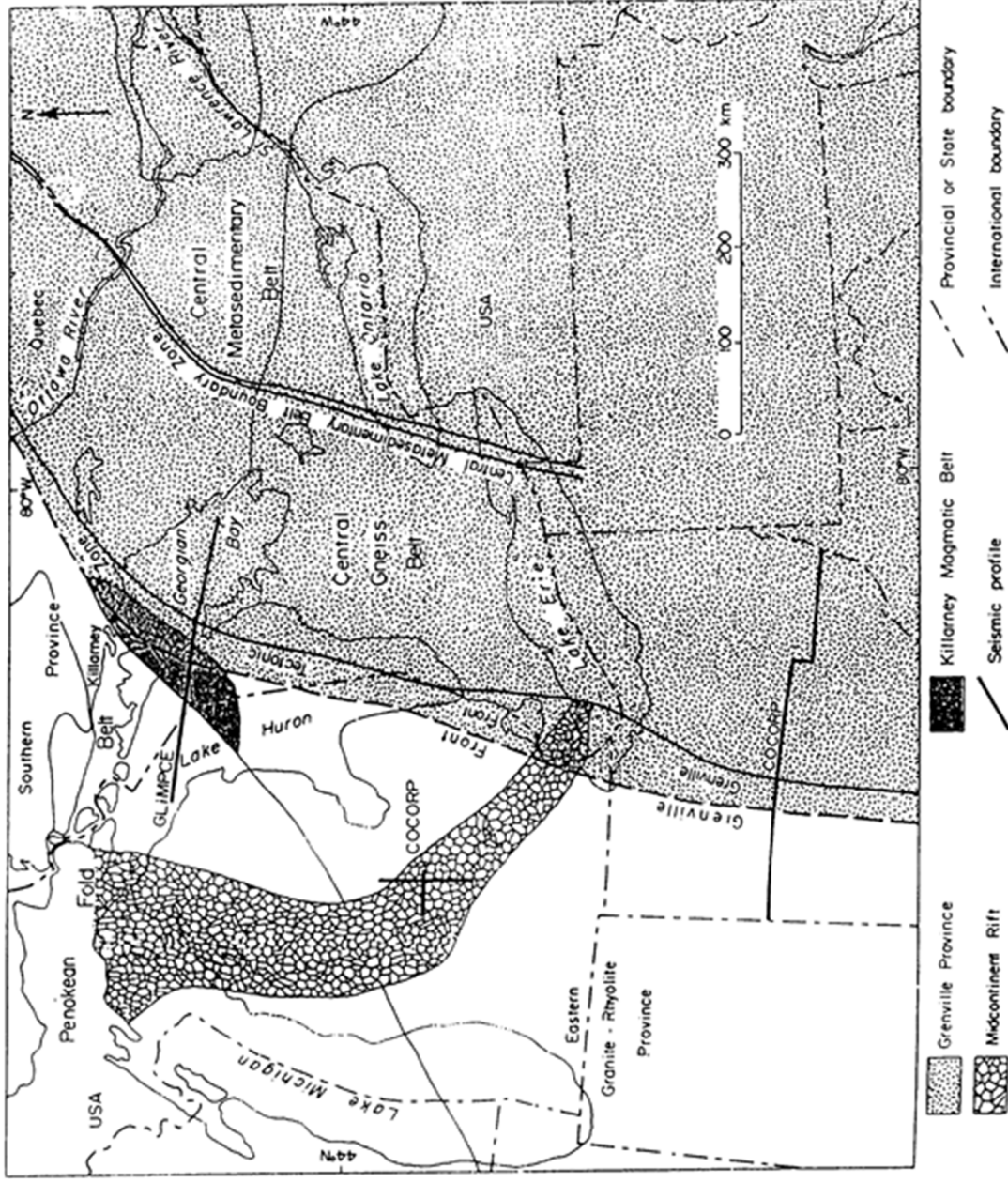
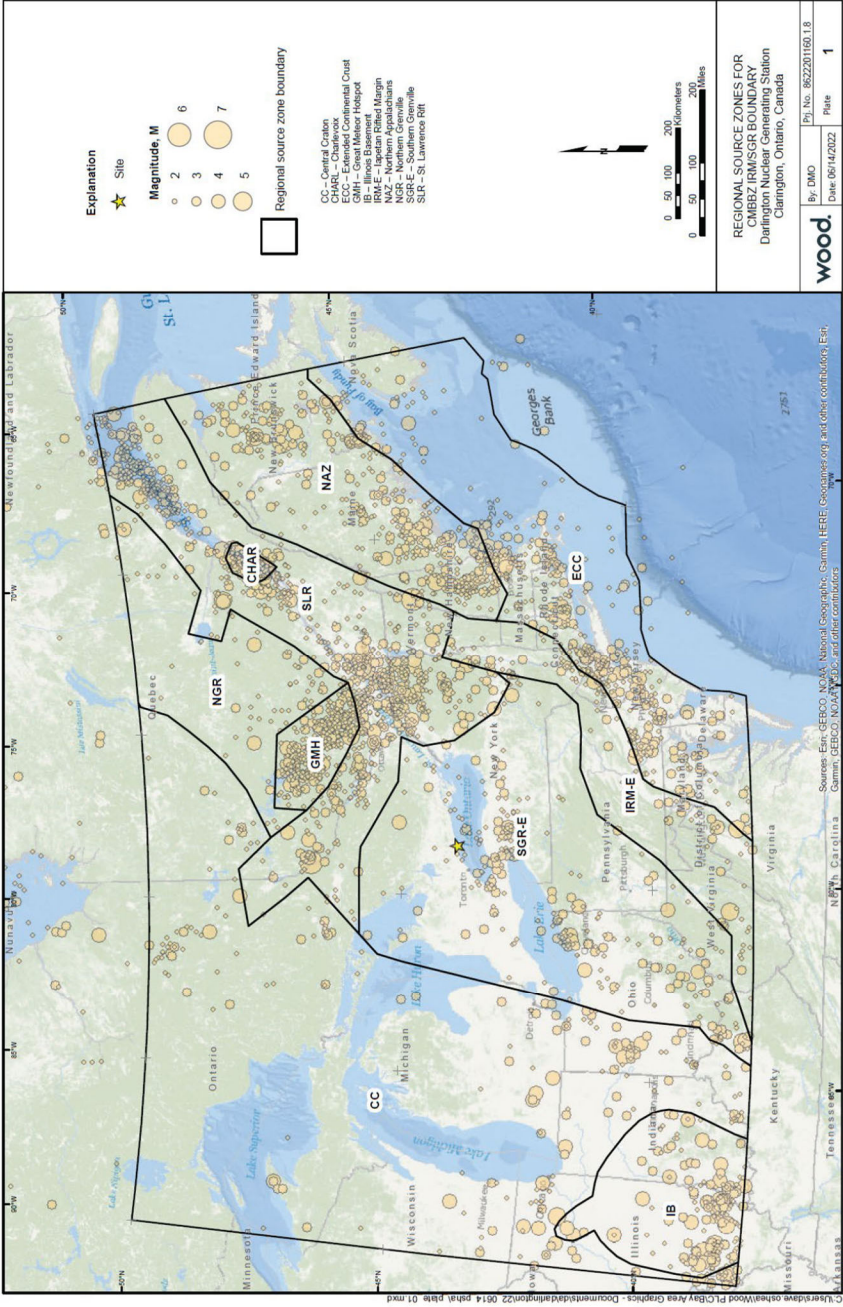


Figure 2.7.4.2-1: Principal Subdivisions of Precambrian Rocks in the Great Lakes Region (Reference 2.7-5)

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Abbreviations:

CC – Central Craton
CHAR – Charlevoix
ECC – Extended Continental Crust
GMH – Great Meteor Hotspot
IB – Illinois Basin

IRM-E – Iapetan Rifted Margin
NAZ – Northern Appalachians
NGR – Northern Grenville
SGR-E – Southern Grenville
SLR – St. Lawrence Rift

Figure 2.7.4.2-2: Regional Source Zones for IRM/SGR Boundary Eastern Boundary (Reference 2.7-41)

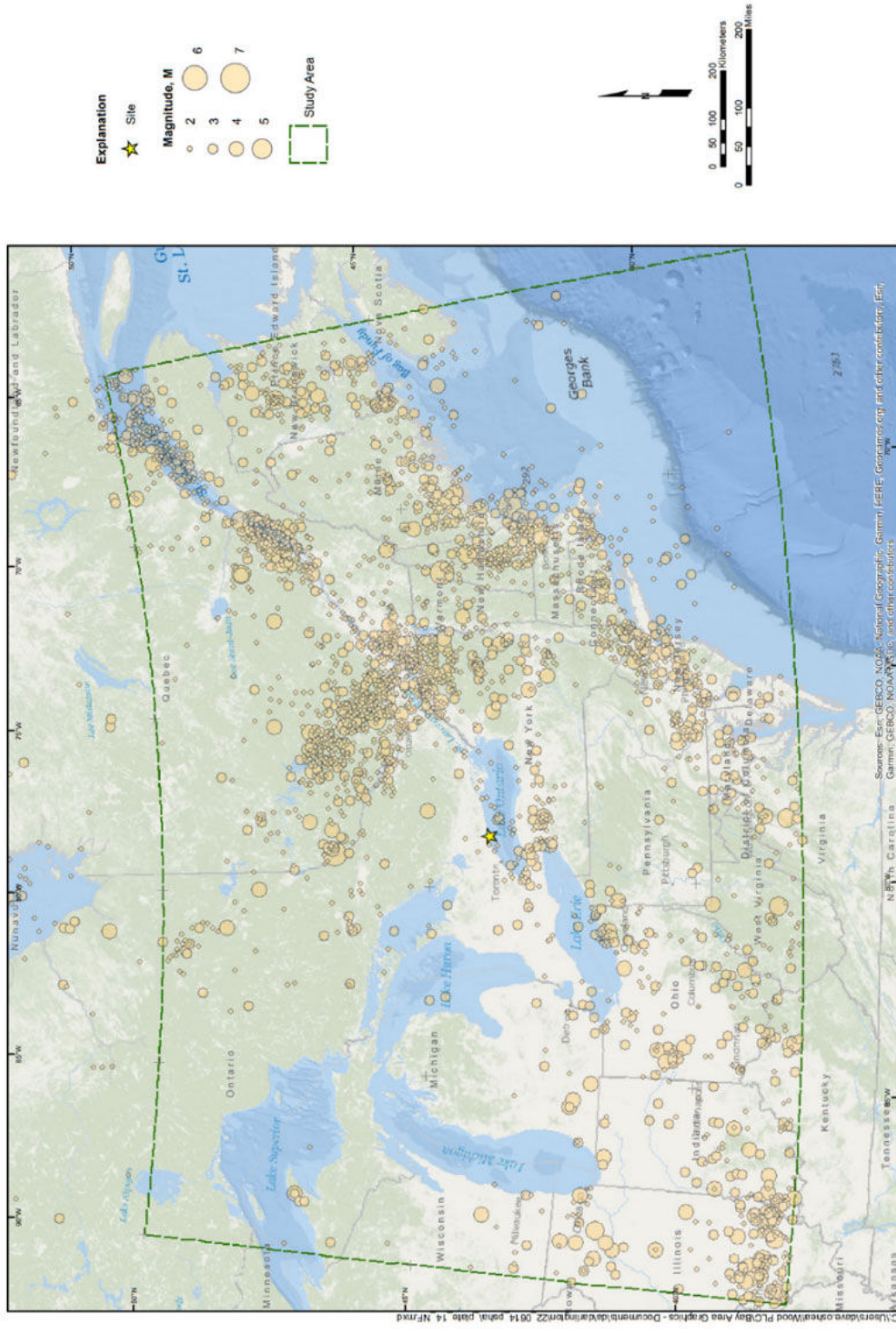


Figure 2.7.4.3-1: Map of Independent Earthquakes in the Updated Earthquake Catalogue for the Study Region
(Reference 2.7-41)

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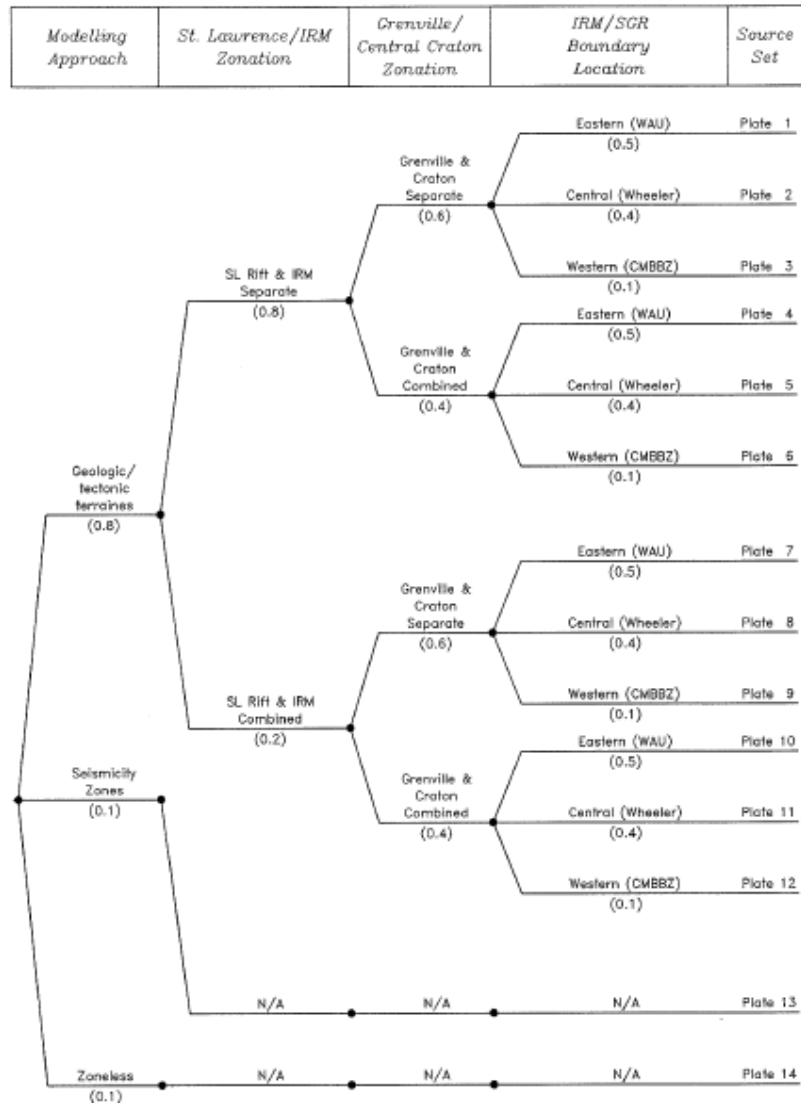


Figure 2.7.4.4-1: Logic Tree for Distributed Seismicity Sources (Reference 2.7- 42)

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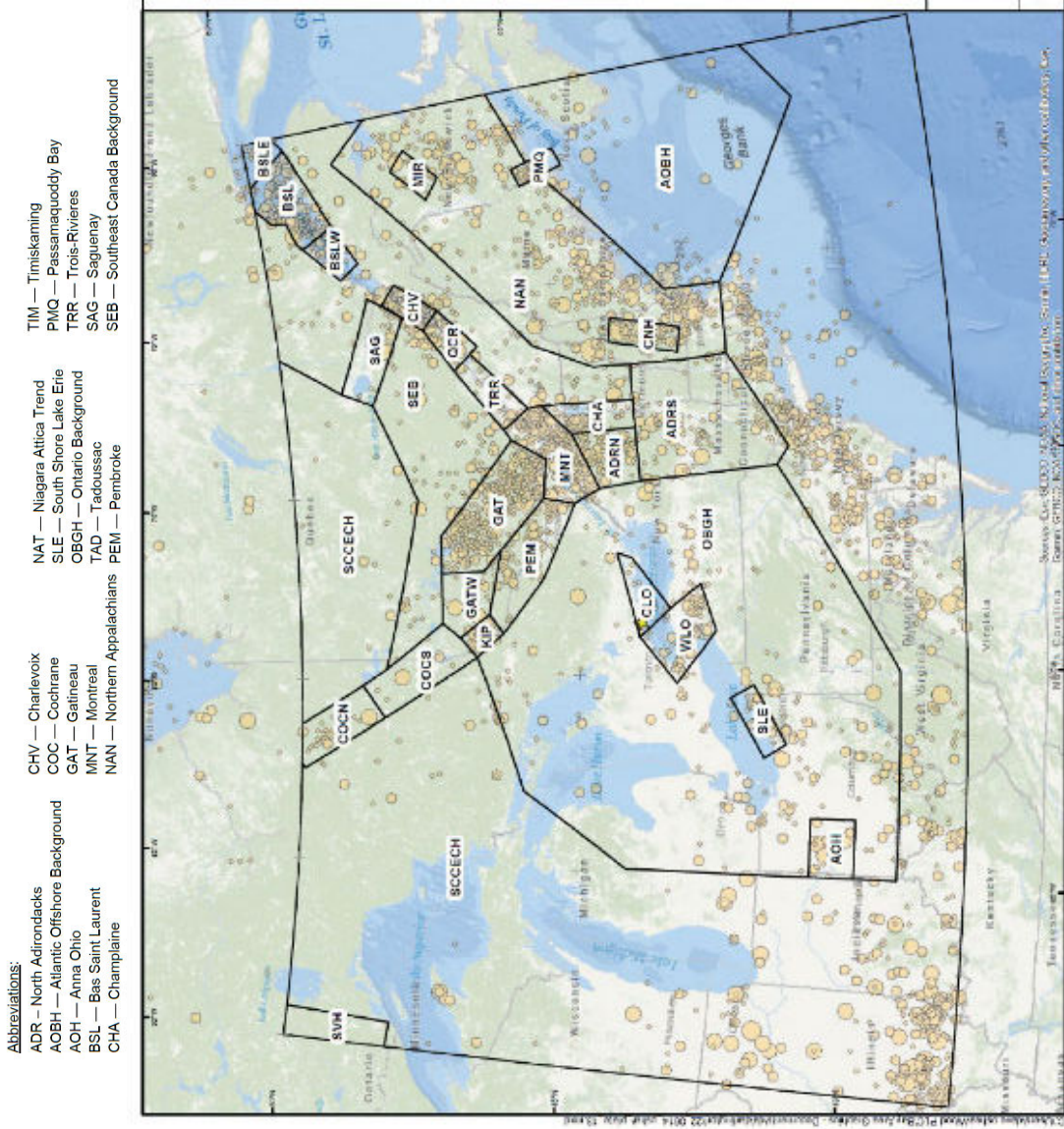


Figure 2.7.4.4-2: Regional Seismicity Source Zones from the Geological Survey of Canada's Sixth Generation H Model
(Plate 13 in Reference 2.7-41)

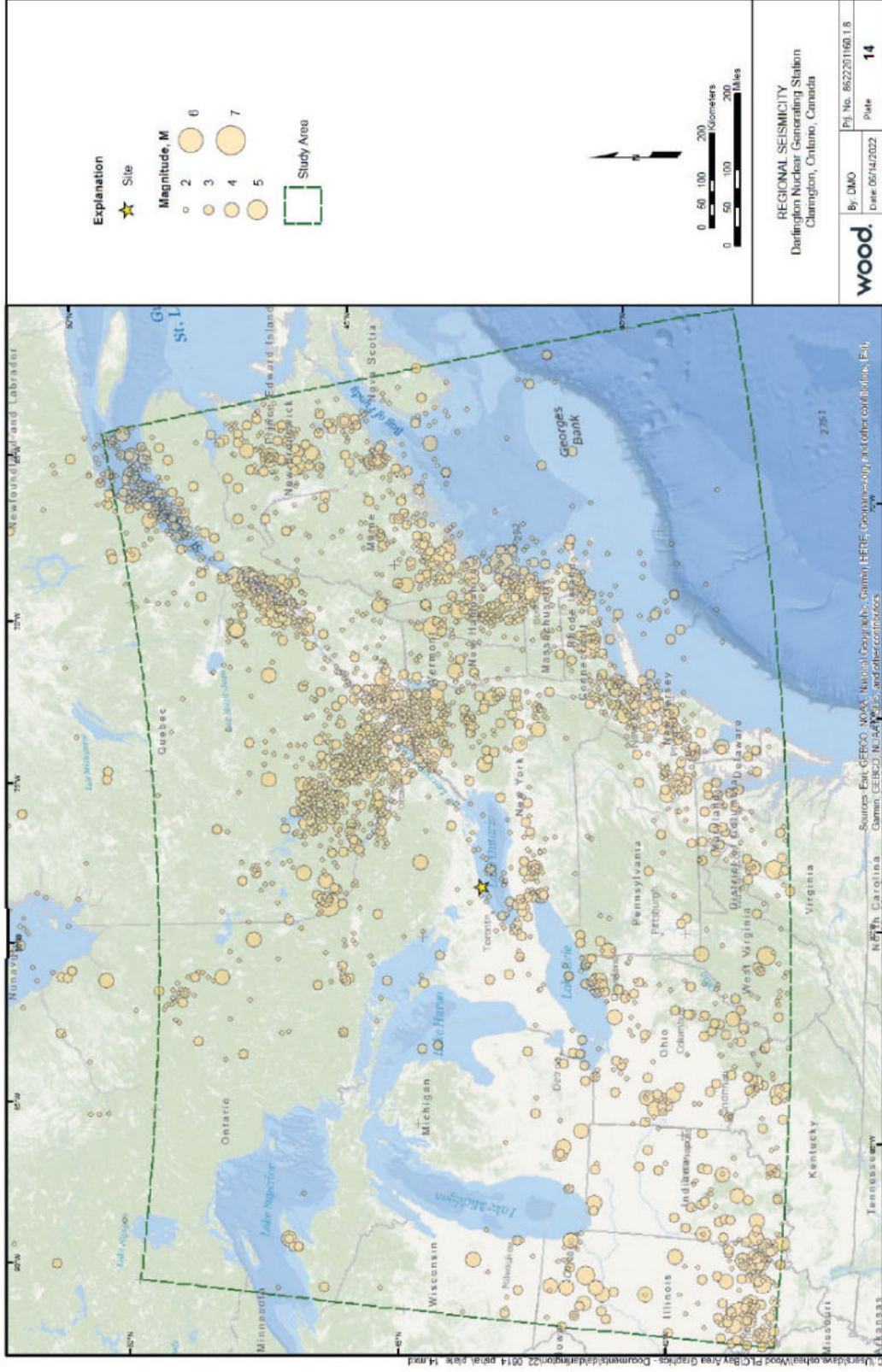
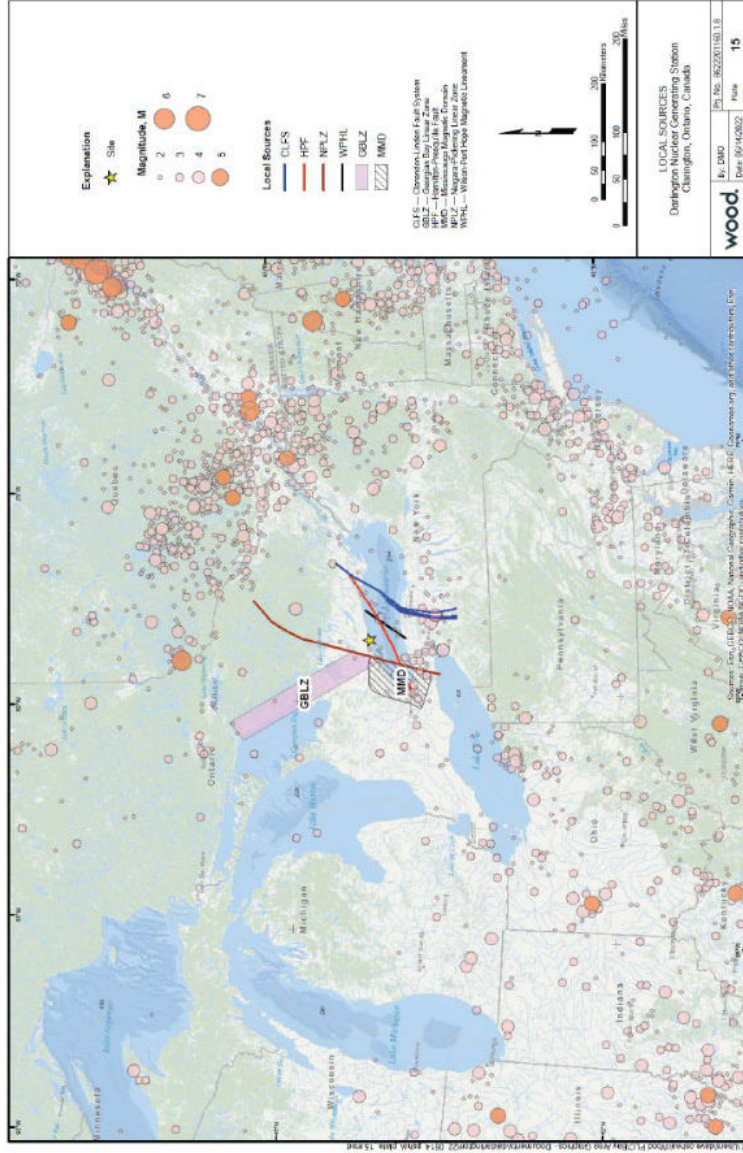


Figure 2.7.4.4-3: Regional Seismicity for Zoneless Model (Plate 14 in Reference 2.7-41)



Abbreviations:

- CLFS — Clarendon-Linden Fault System
- GBLZ — Georgian Bay Linear Zone
- HPF — Hamilton-Presqu'ile Fault
- MMD — Mississauga Magnetic Domain
- NPLZ — Niagara-Pickering Linear Zone
- WPHL — Wilson-Port Hope Magnetic Lineament

Figure 2.7.4-4: Local Source Zones (Plate 15 in Reference 2.7-41)

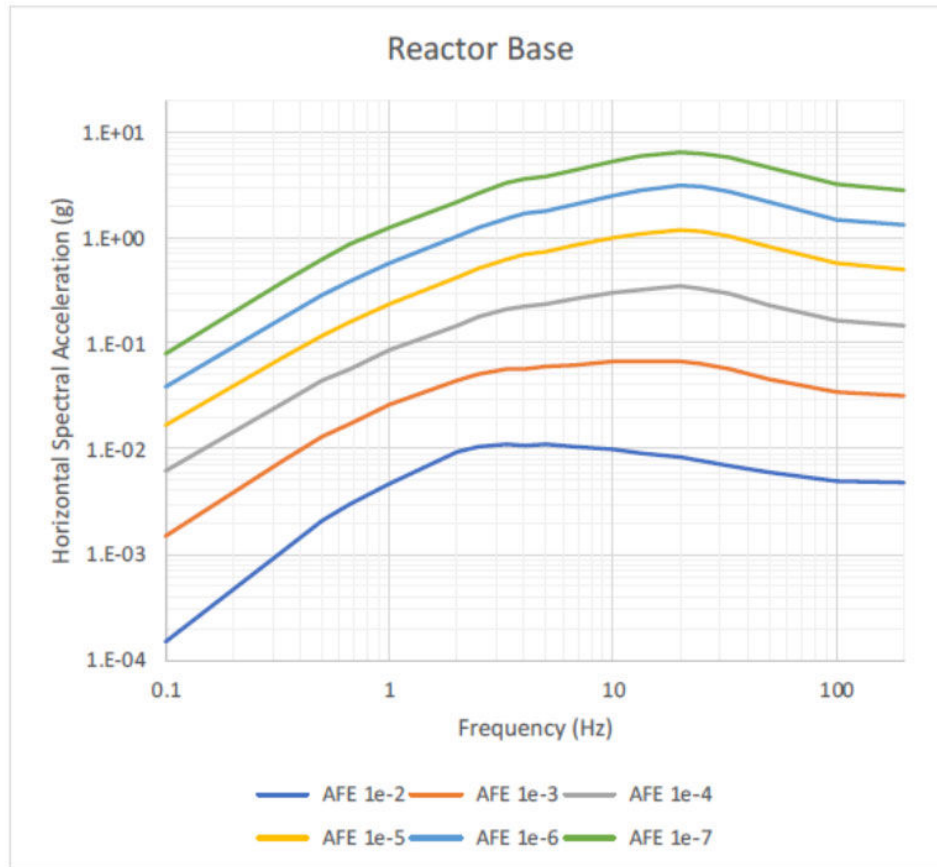


Figure 2.7.4.6-1: Horizontal UHRS at Elevation 52.93 m Based on Mean Hazard (Reference 2.7-41)

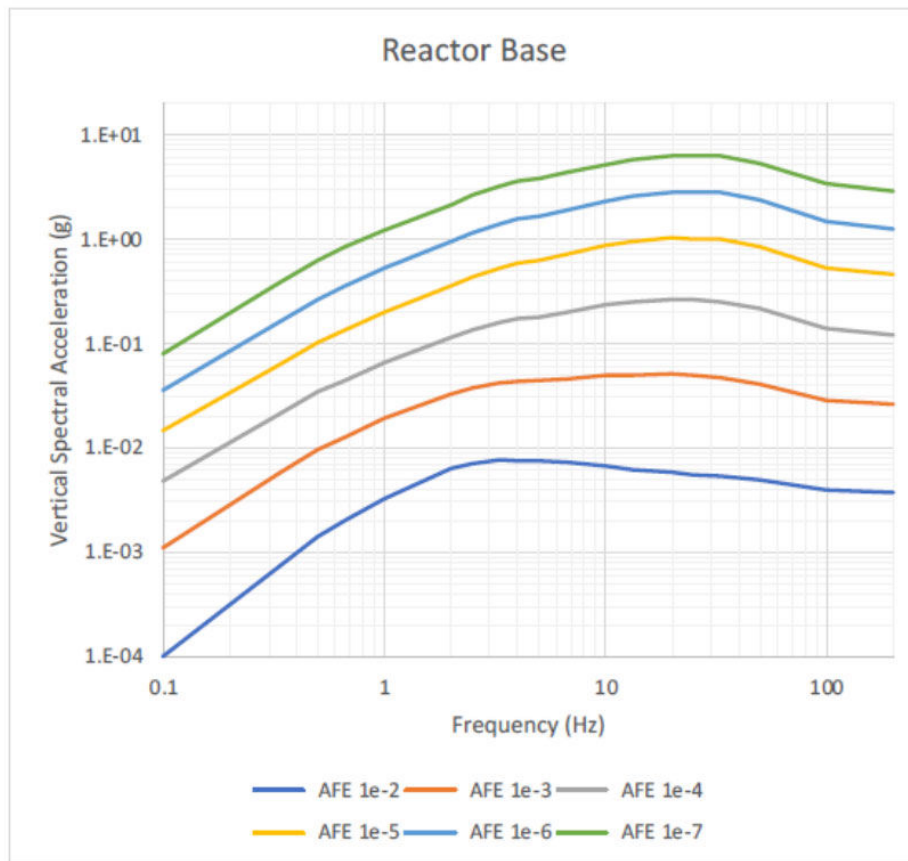
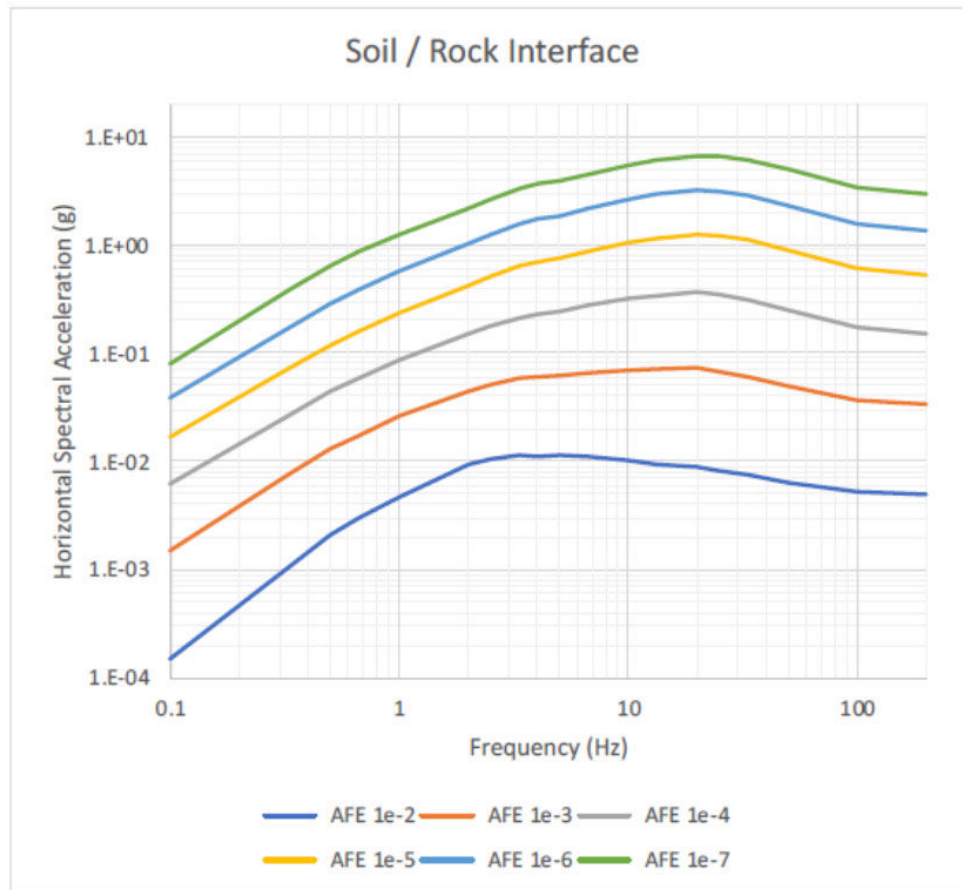
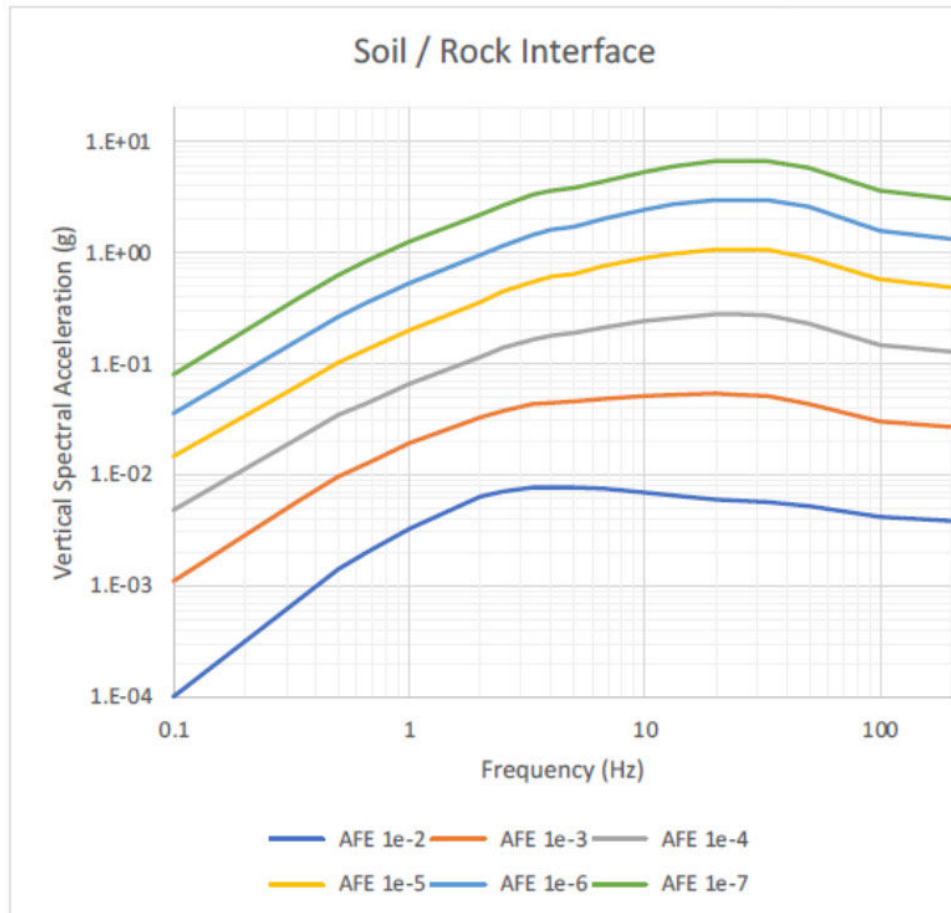


Figure 2.7.4.6-2: Vertical UHRS at Elevation 52.93 m Based on Mean Hazard (Reference 2.7-41)



**Figure 2.7.4.6-3: Horizontal UHRS at Elevation 64 m Based on Mean Hazard
(Reference 2.7-41)**



**Figure 2.7.4.6-4: Vertical UHRS at Elevation 64 m Based on Mean Hazard
(Reference 2.7-41)**

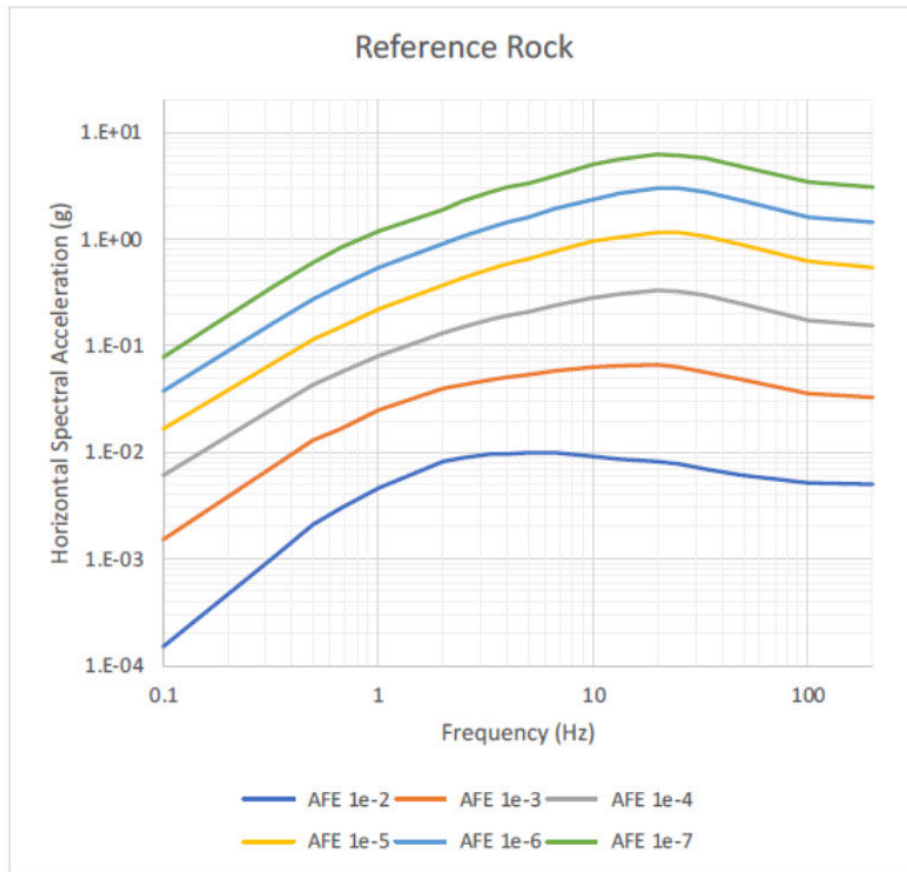


Figure 2.7.4.6-5: Horizontal UHRS for Finished Grade at Elevation 88 m Based on Mean Hazard (Reference 2.7-41)

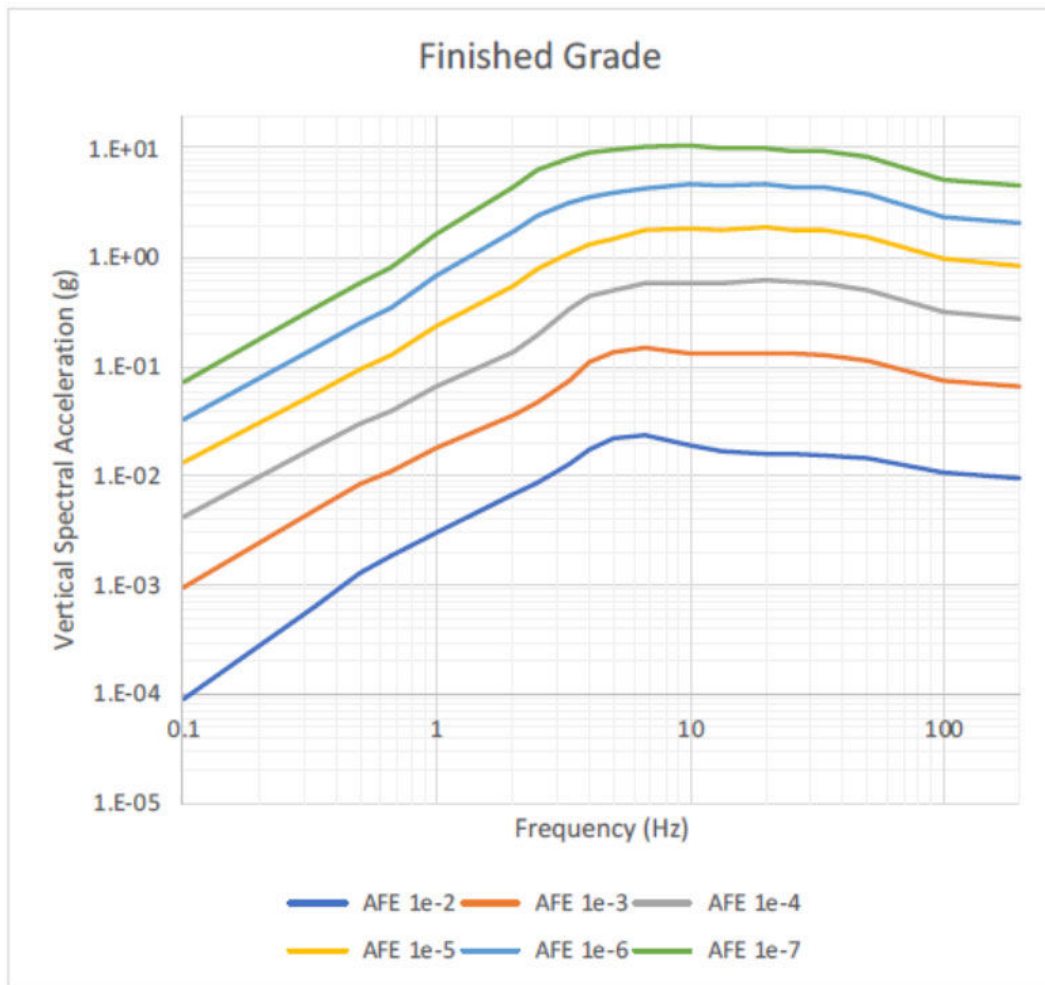


Figure 2.7.4.6-6: Vertical UHRS for Finished Grade at Elevation 88 m Based on Mean Hazard (Reference 2.7-41)

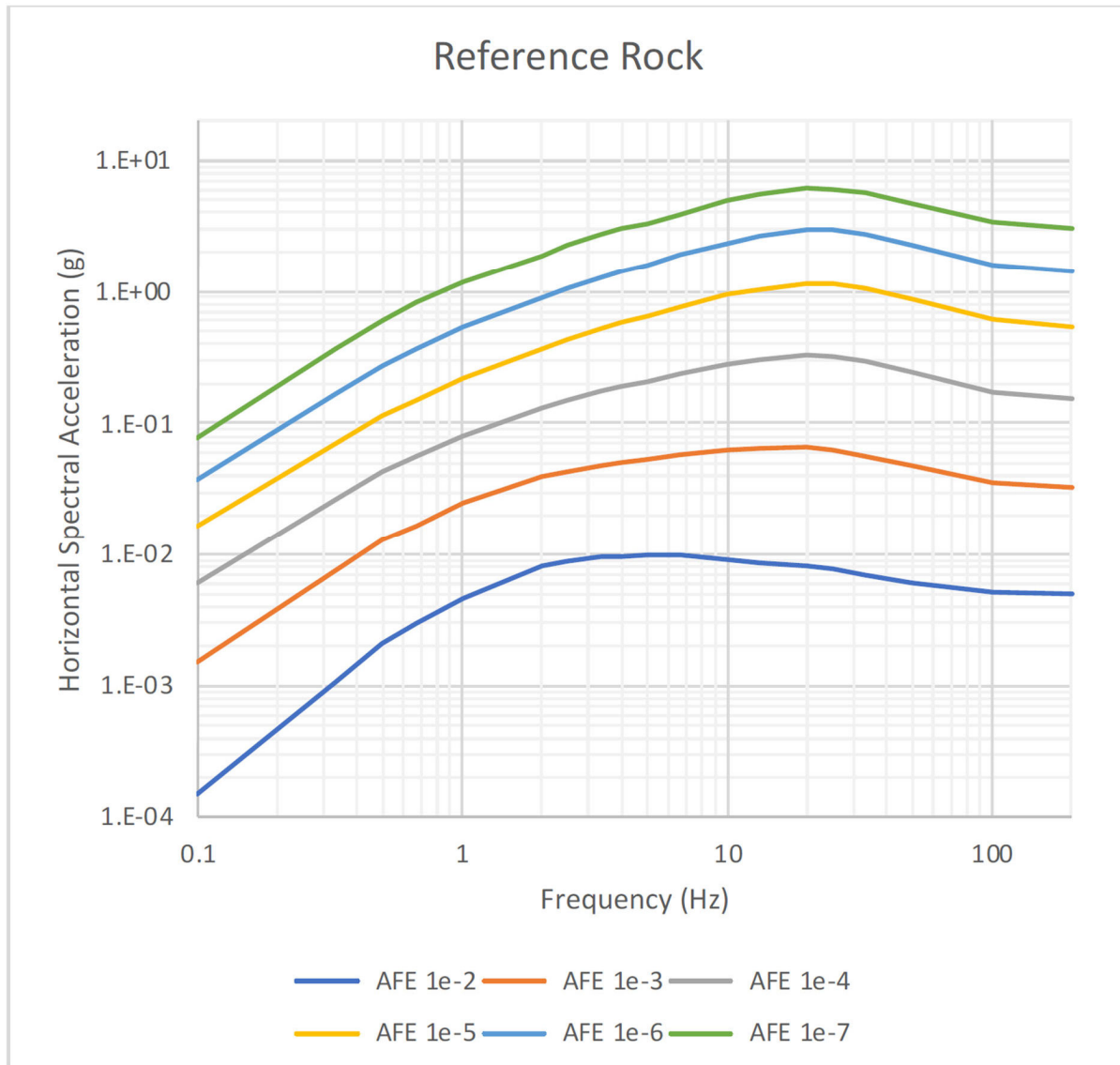
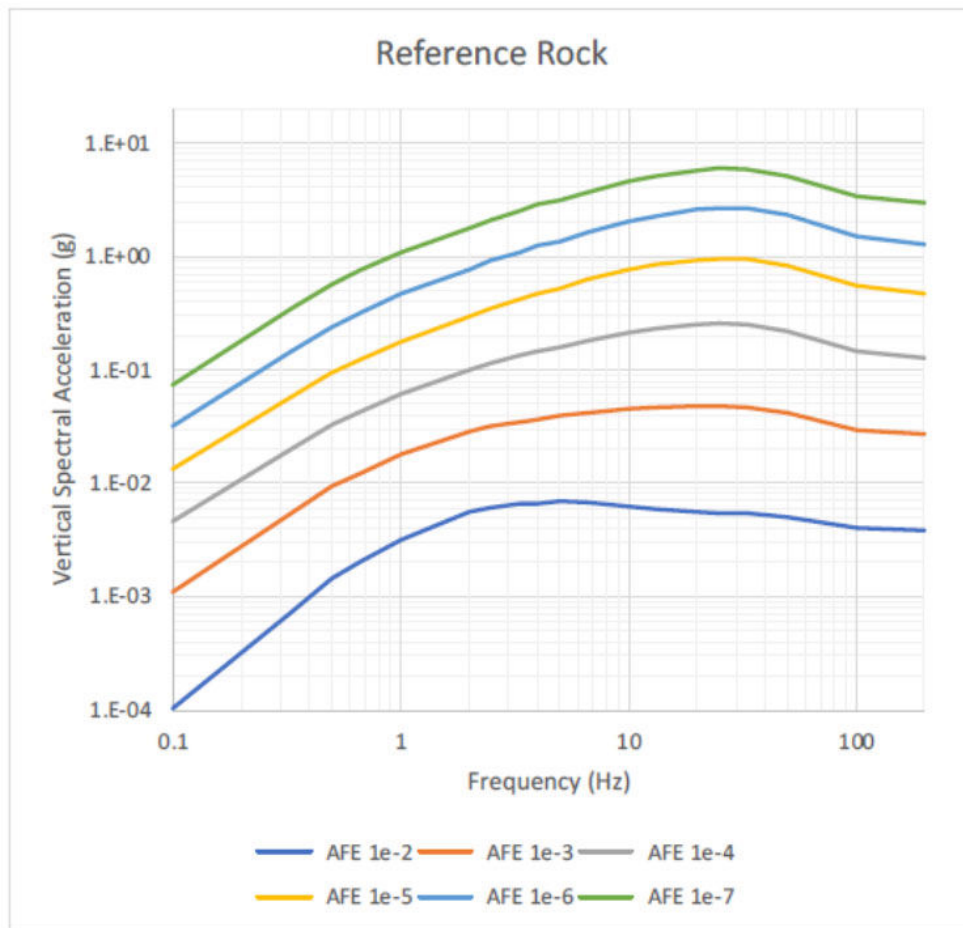
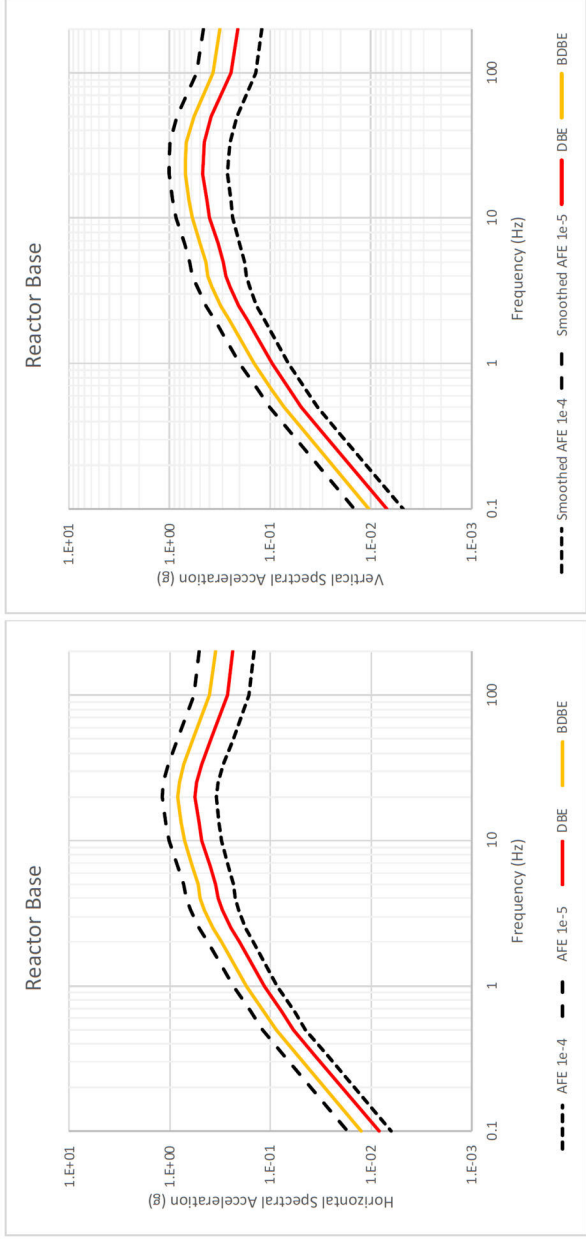


Figure 2.7.4.6-7: Horizontal UHRS for Reference Rock Based on Mean Hazard (Reference 2.7-41)



**Figure 2.7.4.6-8: Vertical UHRS for Reference Rock Based on Mean Hazard
(Reference 2.7-41)**



**Figure 2.7.4.6-9: Initial Horizontal DBE and BDBE Spectra for Reactor Base (Elevation 52.93 m)
(Reference 2.7-41)**

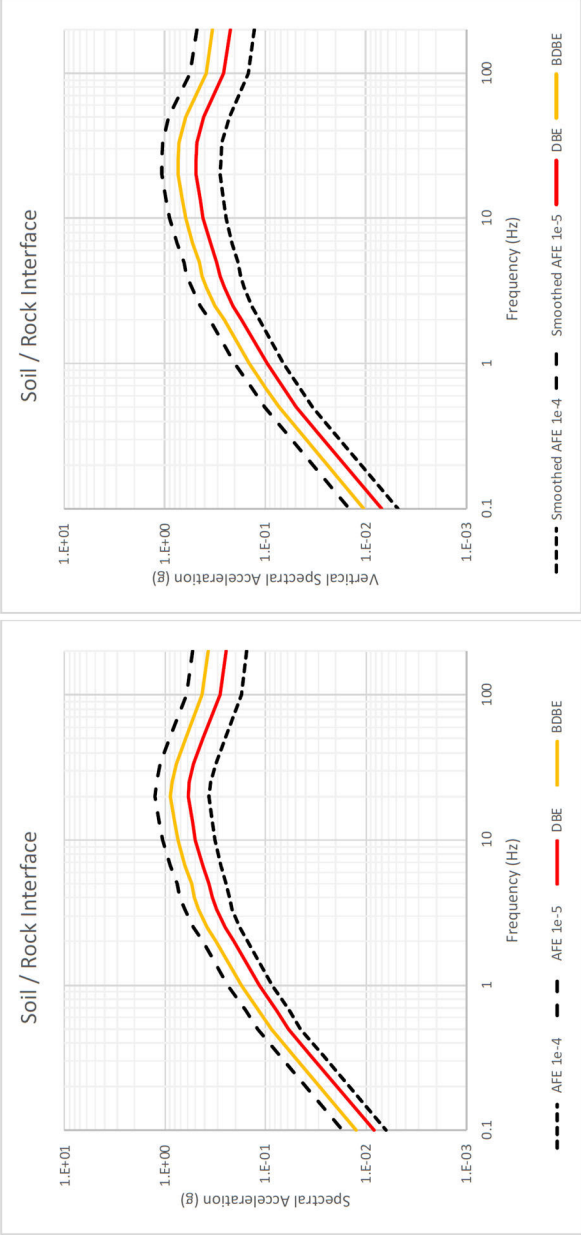


Figure 2.7.4.6-10: Initial Horizontal DBE and BD8E Spectra for Soil-Rock Interface (Elevation 64 m) (Reference 2.7.41)

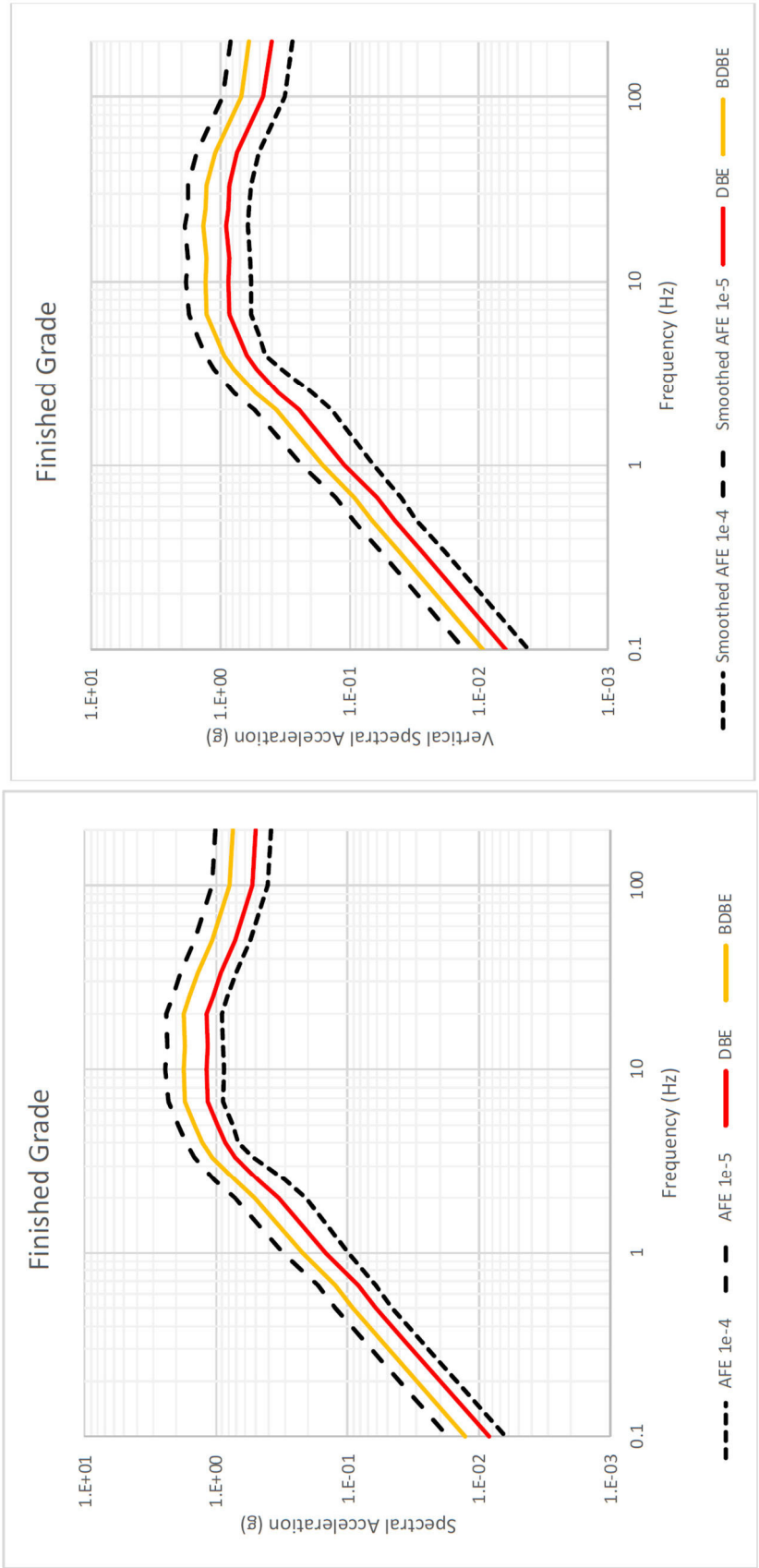


Figure 2.7.4.6-11: Initial Horizontal DBE and BDBE Spectra for Finished Grade (Elevation 88 m) (Reference 2.7-41)

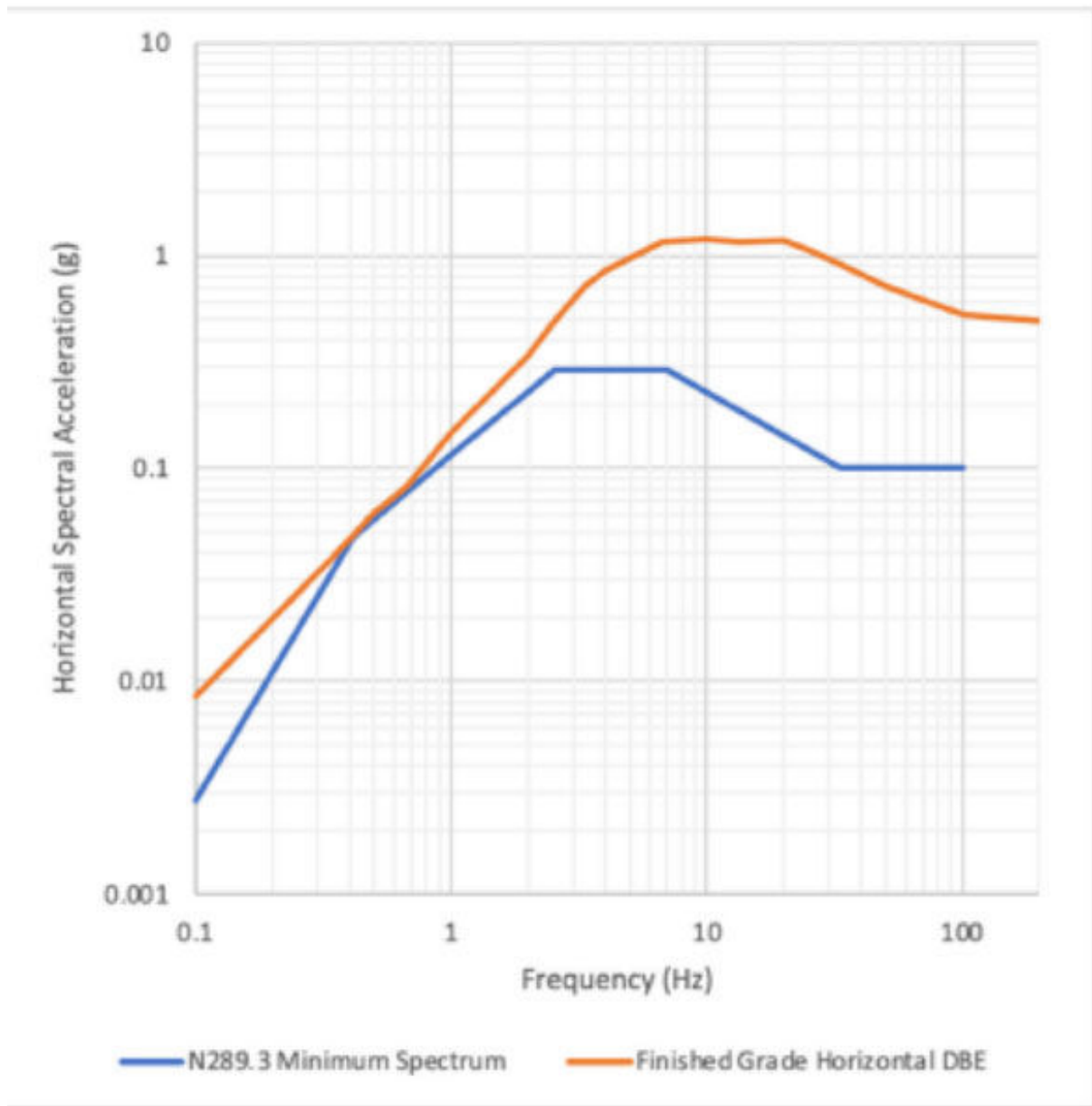


Figure 2.7.4.6-12: Comparison of Finished Grade Horizontal DBE with CSA N289.3 Minimum Spectrum (Reference 2.7-41)

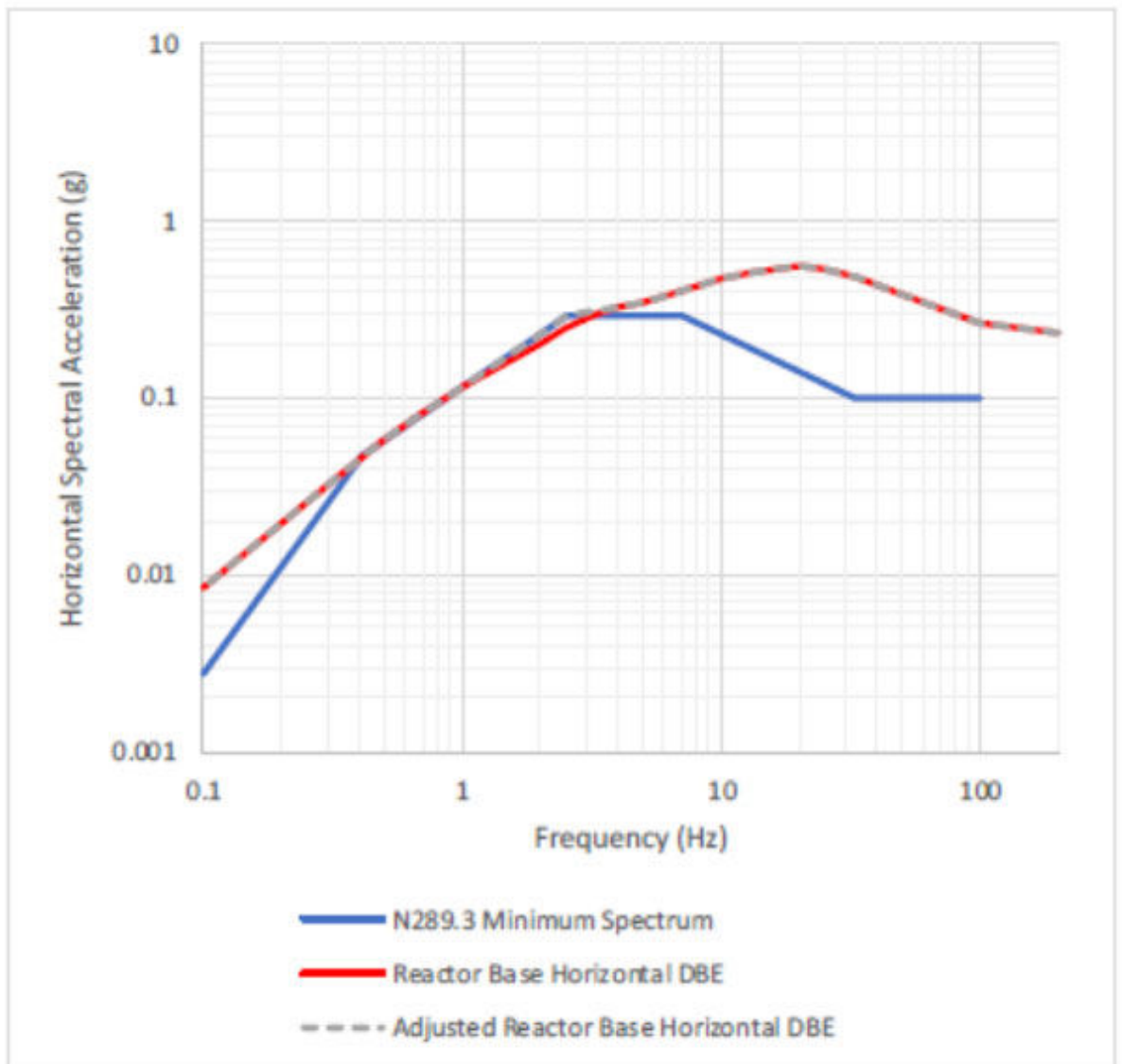


Figure 2.7.4.6-13: Comparison of Reactor Building Base Horizontal DBE with CSA N289.3 Minimum Spectrum (Reference 2.7-41)

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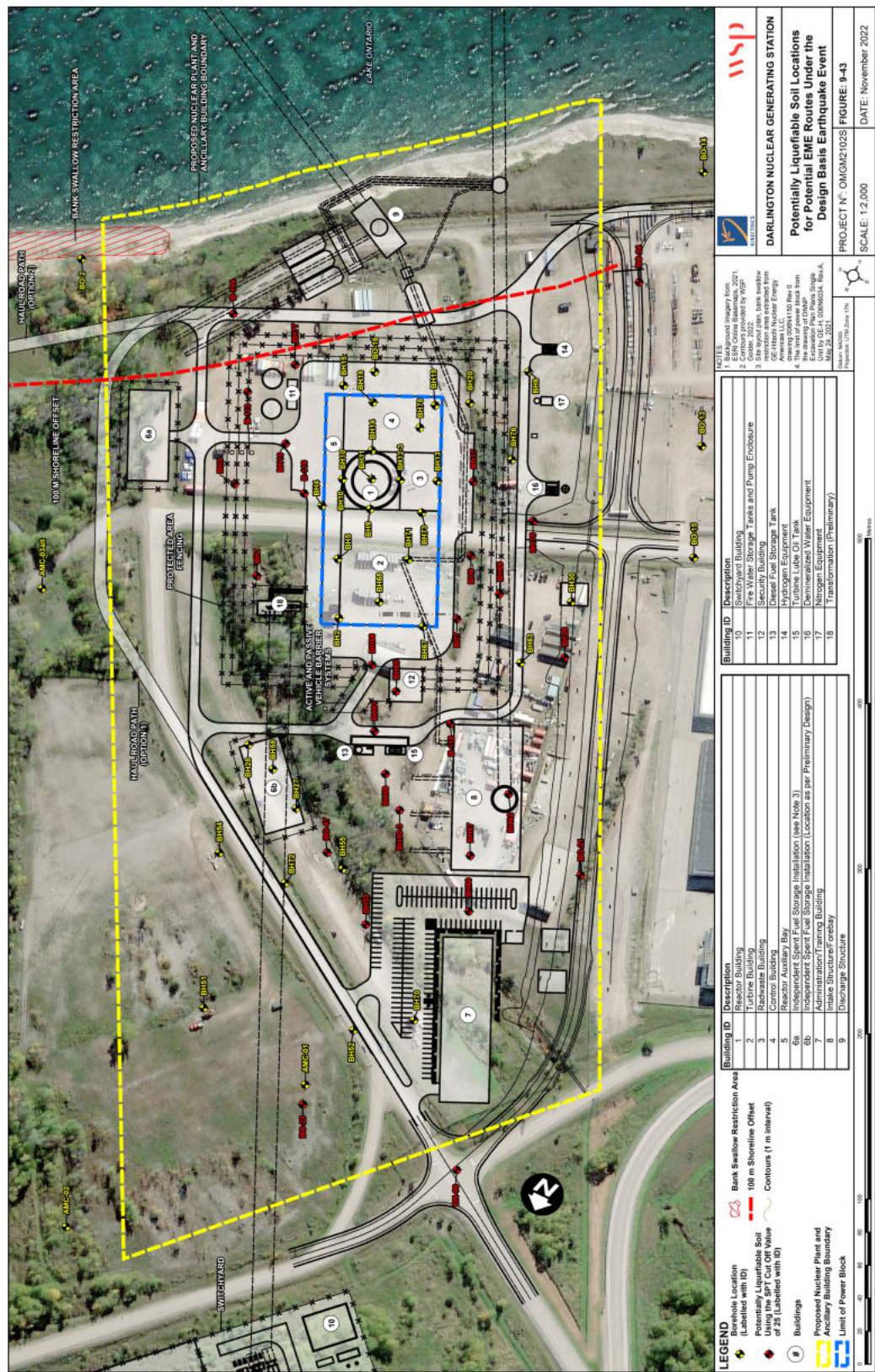


Figure 2.7.4.7-1: Locations of Boreholes Potentially to Liquefaction Under the DBE Event (Reference 2.7-42)

LEGEND

- Buildings (Labelled with ID)
- Potentially Liquefiable Soil (Using the SPT Cut Off Value of 30 (Labelled with ID))
- Bank Swallow Restriction Area
- 100 m Shoreline Offset
- Contours (1 m Interval)
- Buildings
- Proposed Nuclear Plant and Ancillary Building Boundary
- Limit of Power Block

Building ID | **Description**

1	Reactor Building
2	Turbine Building
3	Control Building
4	Control Building
5	Reactor Auxiliary Bay
6a	Independent Spent Fuel Storage Installation (see Note 3)
6b	Independent Spent Fuel Storage Installation (Location as per Preliminary Design)
7	Administration Training Building
8	Administration Training Building
9	Discharge Structure

Building ID | **Description**

10	Switchyard Building
11	Fire Water Storage Tanks and Pump Enclosure
12	Hydrogen Equipment
13	Disposal Fuel Storage Tank
14	Hydrogen Equipment
15	Turbine Lube Oil Tank
16	Demineralized Water Equipment
17	Nitrogen Equipment
18	Transformer (Preliminary)
19	Transformer (Preliminary)

NOTES

- Background imagery from Google Earth, 2021.
- Contours provided by WSP.
- Greater 2020, 2021, 2022, 2023, 2024, 2025, 2026, 2027, 2028, 2029, 2030, 2031, 2032, 2033, 2034, 2035, 2036, 2037, 2038, 2039, 2040, 2041, 2042, 2043, 2044, 2045, 2046, 2047, 2048, 2049, 2050, 2051, 2052, 2053, 2054, 2055, 2056, 2057, 2058, 2059, 2060, 2061, 2062, 2063, 2064, 2065, 2066, 2067, 2068, 2069, 2070, 2071, 2072, 2073, 2074, 2075, 2076, 2077, 2078, 2079, 2080, 2081, 2082, 2083, 2084, 2085, 2086, 2087, 2088, 2089, 2090, 2091, 2092, 2093, 2094, 2095, 2096, 2097, 2098, 2099, 2100, 2101, 2102, 2103, 2104, 2105, 2106, 2107, 2108, 2109, 2110, 2111, 2112, 2113, 2114, 2115, 2116, 2117, 2118, 2119, 2120, 2121, 2122, 2123, 2124, 2125, 2126, 2127, 2128, 2129, 2130, 2131, 2132, 2133, 2134, 2135, 2136, 2137, 2138, 2139, 2140, 2141, 2142, 2143, 2144, 2145, 2146, 2147, 2148, 2149, 2150, 2151, 2152, 2153, 2154, 2155, 2156, 2157, 2158, 2159, 2160, 2161, 2162, 2163, 2164, 2165, 2166, 2167, 2168, 2169, 2170, 2171, 2172, 2173, 2174, 2175, 2176, 2177, 2178, 2179, 2180, 2181, 2182, 2183, 2184, 2185, 2186, 2187, 2188, 2189, 2190, 2191, 2192, 2193, 2194, 2195, 2196, 2197, 2198, 2199, 2200, 2201, 2202, 2203, 2204, 2205, 2206, 2207, 2208, 2209, 2210, 2211, 2212, 2213, 2214, 2215, 2216, 2217, 2218, 2219, 2220, 2221, 2222, 2223, 2224, 2225, 2226, 2227, 2228, 2229, 2230, 2231, 2232, 2233, 2234, 2235, 2236, 2237, 2238, 2239, 2240, 2241, 2242, 2243, 2244, 2245, 2246, 2247, 2248, 2249, 2250, 2251, 2252, 2253, 2254, 2255, 2256, 2257, 2258, 2259, 2260, 2261, 2262, 2263, 2264, 2265, 2266, 2267, 2268, 2269, 2270, 2271, 2272, 2273, 2274, 2275, 2276, 2277, 2278, 2279, 2280, 2281, 2282, 2283, 2284, 2285, 2286, 2287, 2288, 2289, 2290, 2291, 2292, 2293, 2294, 2295, 2296, 2297, 2298, 2299, 2300, 2301, 2302, 2303, 2304, 2305, 2306, 2307, 2308, 2309, 2310, 2311, 2312, 2313, 2314, 2315, 2316, 2317, 2318, 2319, 2320, 2321, 2322, 2323, 2324, 2325, 2326, 2327, 2328, 2329, 2330, 2331, 2332, 2333, 2334, 2335, 2336, 2337, 2338, 2339, 2340, 2341, 2342, 2343, 2344, 2345, 2346, 2347, 2348, 2349, 2350, 2351, 2352, 2353, 2354, 2355, 2356, 2357, 2358, 2359, 2360, 2361, 2362, 2363, 2364, 2365, 2366, 2367, 2368, 2369, 2370, 2371, 2372, 2373, 2374, 2375, 2376, 2377, 2378, 2379, 2380, 2381, 2382, 2383, 2384, 2385, 2386, 2387, 2388, 2389, 2390, 2391, 2392, 2393, 2394, 2395, 2396, 2397, 2398, 2399, 2400, 2401, 2402, 2403, 2404, 2405, 2406, 2407, 2408, 2409, 2410, 2411, 2412, 2413, 2414, 2415, 2416, 2417, 2418, 2419, 2420, 2421, 2422, 2423, 2424, 2425, 2426, 2427, 2428, 2429, 2430, 2431, 2432, 2433, 2434, 2435, 2436, 2437, 2438, 2439, 2440, 2441, 2442, 2443, 2444, 2445, 2446, 2447, 2448, 2449, 2450, 2451, 2452, 2453, 2454, 2455, 2456, 2457, 2458, 2459, 2460, 2461, 2462, 2463, 2464, 2465, 2466, 2467, 2468, 2469, 2470, 2471, 2472, 2473, 2474, 2475, 2476, 2477, 2478, 2479, 2480, 2481, 2482, 2483, 2484, 2485, 2486, 2487, 2488, 2489, 2490, 2491, 2492, 2493, 2494, 2495, 2496, 2497, 2498, 2499, 2500, 2501, 2502, 2503, 2504, 2505, 2506, 2507, 2508, 2509, 2510, 2511, 2512, 2513, 2514, 2515, 2516, 2517, 2518, 2519, 2520, 2521, 2522, 2523, 2524, 2525, 2526, 2527, 2528, 2529, 2530, 2531, 2532, 2533, 2534, 2535, 2536, 2537, 2538, 2539, 2540, 2541, 2542, 2543, 2544, 2545, 2546, 2547, 2548, 2549, 2550, 2551, 2552, 2553, 2554, 2555, 2556, 2557, 2558, 2559, 2560, 2561, 2562, 2563, 2564, 2565, 2566, 2

2-170

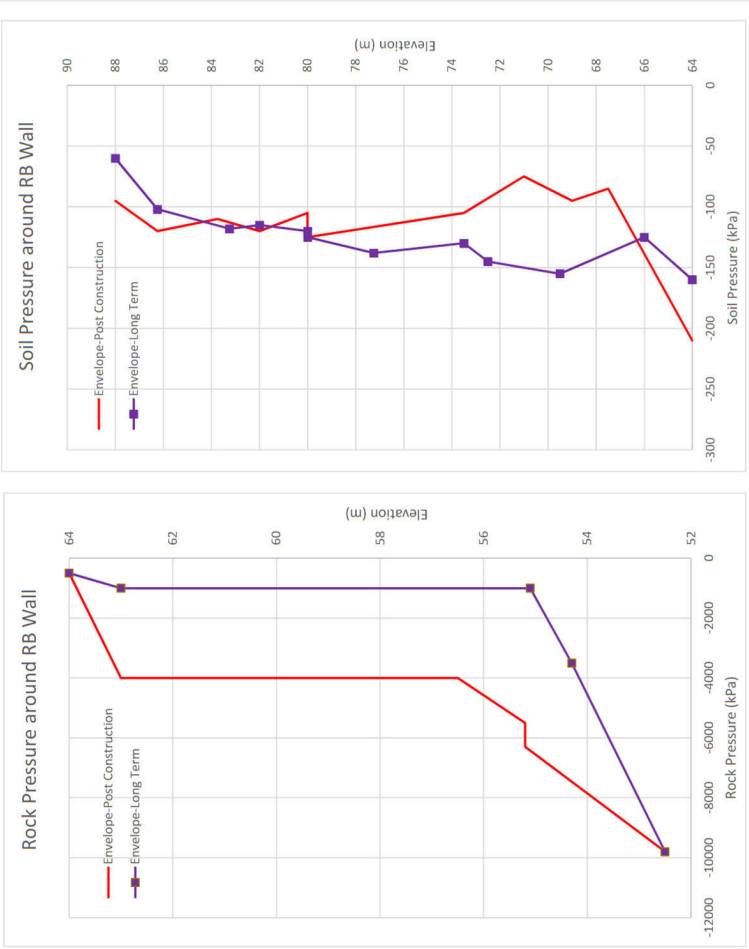


Figure 2.7.5.1.3-1: Rock and Soil Pressures Around RB Wall (Reference 2.7-38)

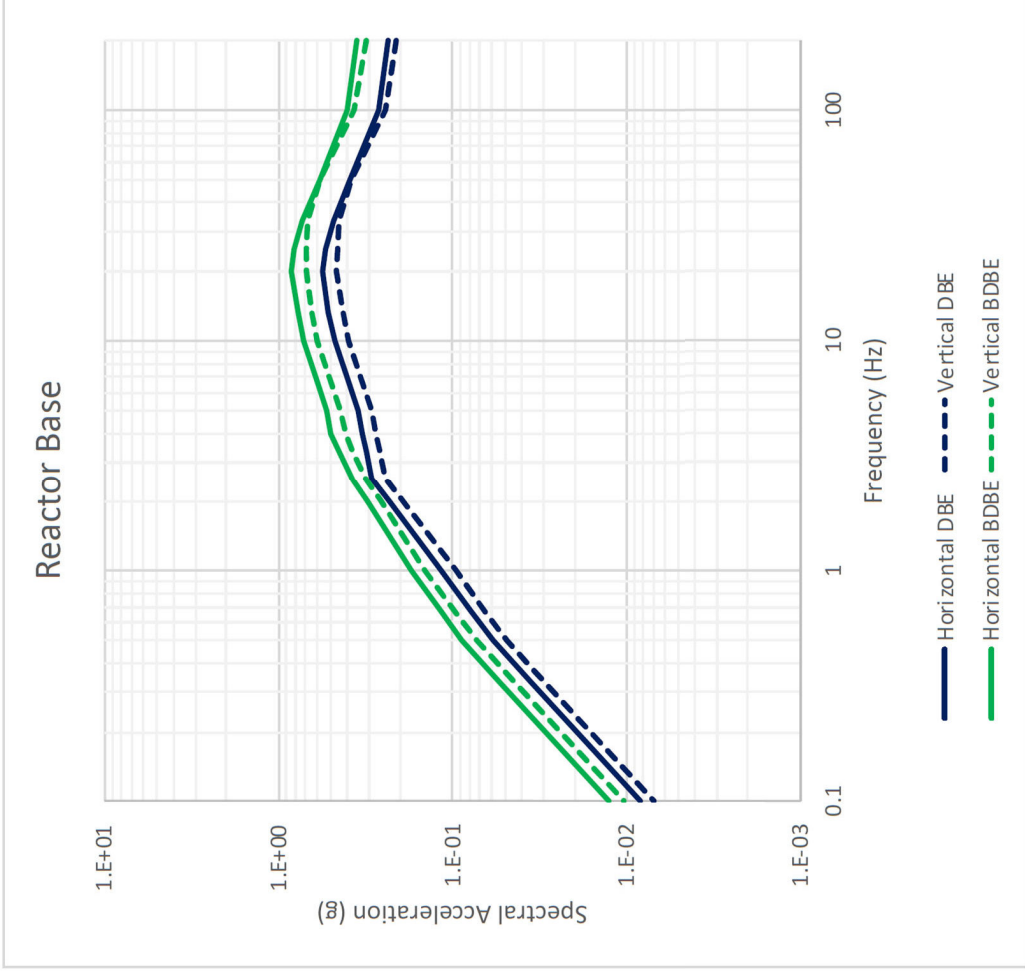


Figure 2.7.5.2.5-1: DBE and BDBE Foundation Input Response Spectra for Reactor Base (Elevation 52.93 m) (Reference 2.7-41)

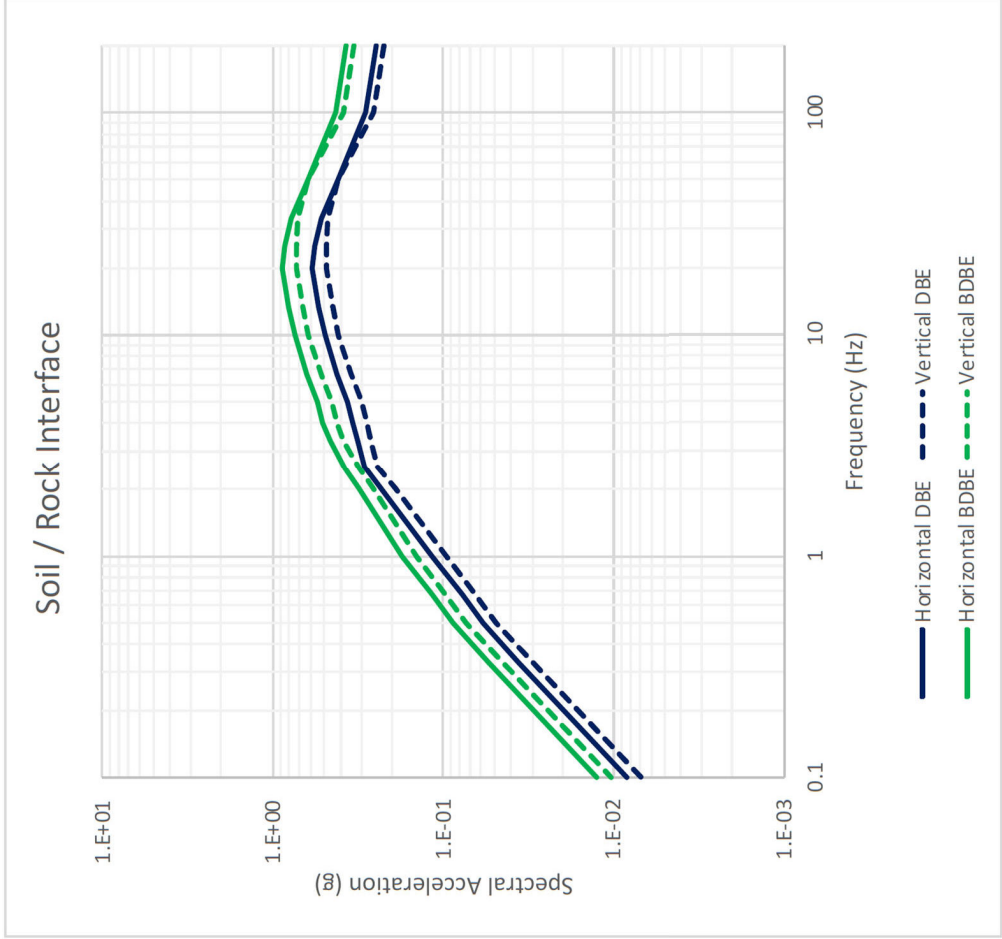


Figure 2.7.5.2.5-2: DBE and BDBE Performance Based Intermediate Response Spectra for Soil/Rock Interface (Elevation 64 m) (Reference 2.7-41)

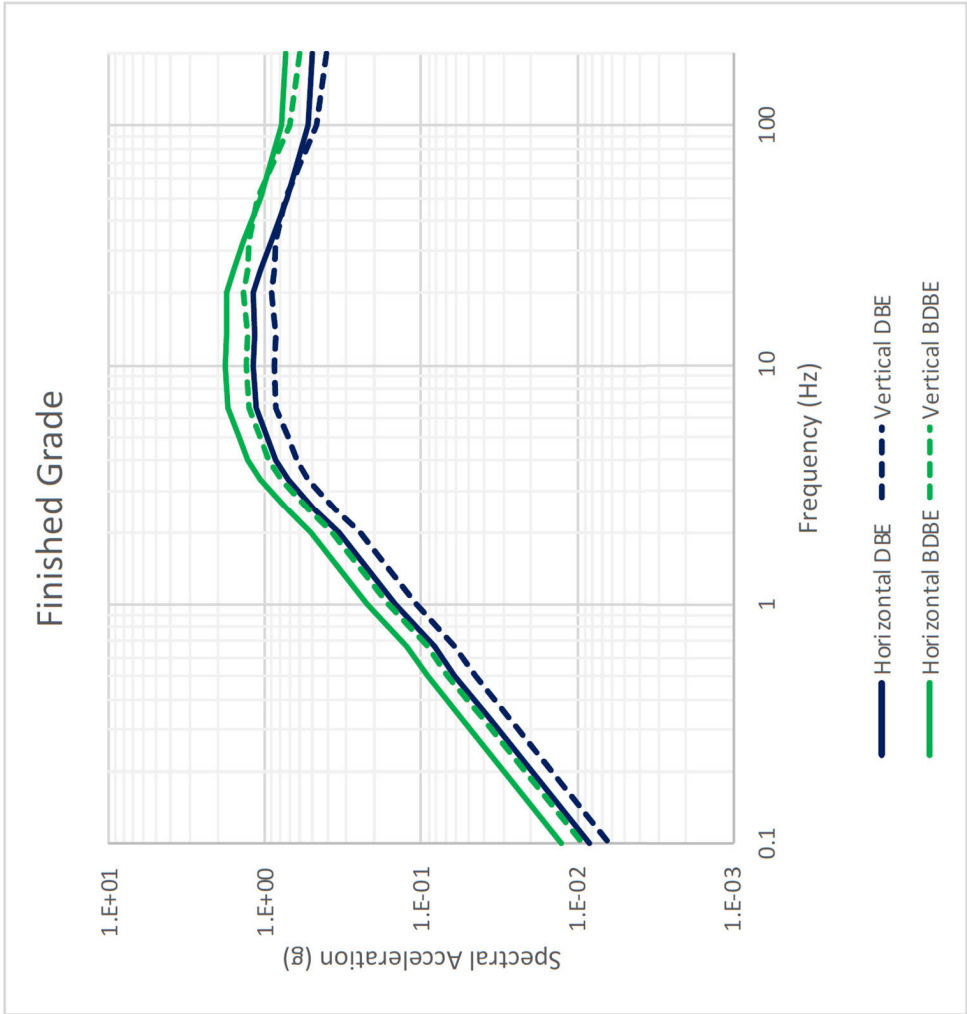


Figure 2.7.5.2.5-3: DBE and BDBE Performance Based Surface Response Spectra for Finished Grade (Elevation 88 m)
(Reference 2.7-41)

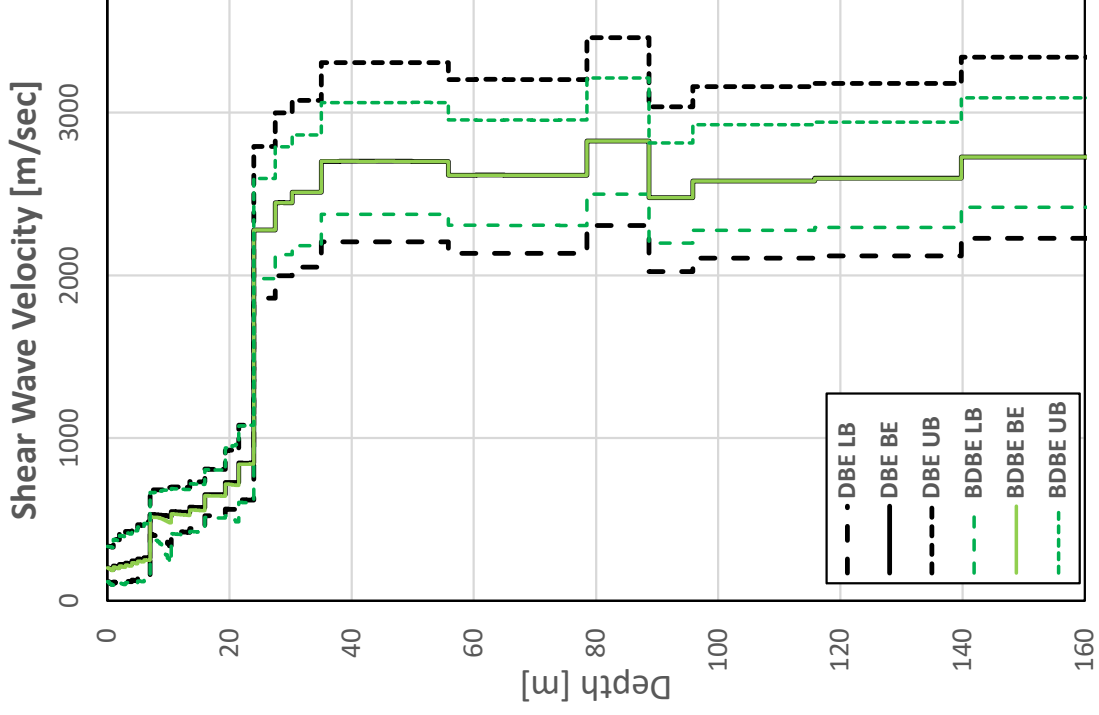


Figure 2.7.5.2.5-4: Subgrade Profiles of DBE and BDBE HCSC Shear Wave Velocities (Reference 2.7.41)

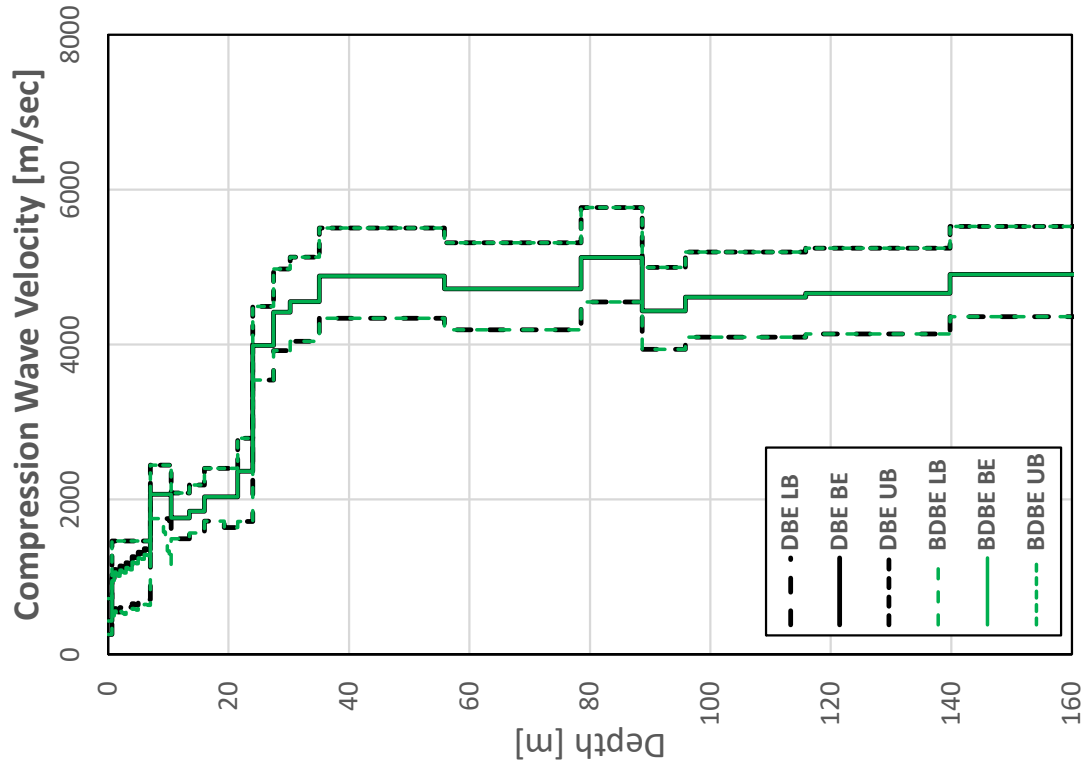


Figure 2.7.5.2.5-5: Subgrade Profiles of DBE and BDBE HCSC Compression Wave Velocities (Reference 2.7-41)

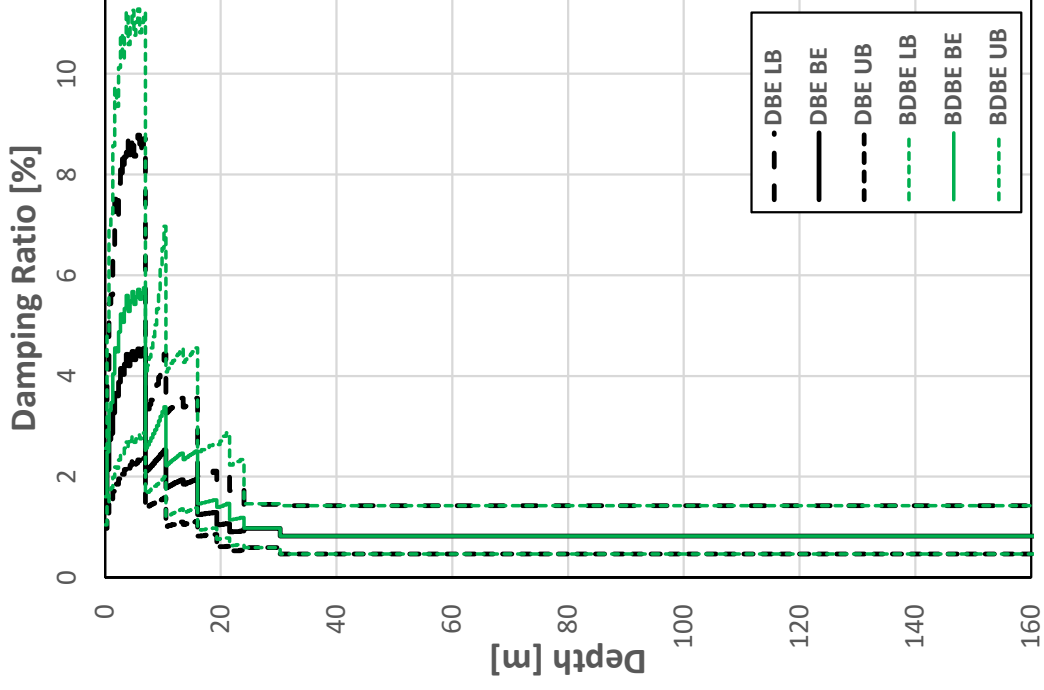


Figure 2.7.5.2.5-6: Subgrade Profiles of DBE and BDBE HCSC Damping Ratios (Reference 2.7-41)

2.8 Site Characteristics Impact on Dispersion of Radioactive Material

The dispersion of radioactive material in water, air, and soil is affected by natural and physical characteristics of the site and the surrounding environment, including meteorology and climate, hydrological and hydrogeological parameters, as well as land cover and use (e.g., vegetation and structures). Population and receptors also influence the potential effects of dispersion of radioactive material. The baseline conditions for these characteristics are established in the:

1. Darlington New Nuclear Project (DNNP) Environmental Impact Statement (EIS), completed in 2009 in NK054-REP-07730-00029 (Reference 2.8-1)
2. Updates to the baseline conditions since the EA was conducted, as discussed in detail in documentation including the 2020 Environmental Risk Assessment (ERA) for the Darlington Nuclear Site, D-REP-07701-00001 (Reference 2.8-2)
3. Yearly Environmental Monitoring Program (EMP) reports, per N-REP-03443-10027 (Reference 2.8-3)
4. DNNP – Site Preparation Licence Renewal Activity Report – Environment, completed in 2020 in NK054-REP-01210-00110 (Reference 2.8-4)
5. Darlington New Nuclear Project Supporting Environment Studies – Environment, completed in 2020, NK054-REP-01210-0001 (Reference 2.8-5)
6. Darlington New Nuclear Project Environmental Impact Statement (EIS) Review Report For Small Modular Reactor BWRX-300, completed in October 2022, per NK054-REP-07730-00055 (Reference 2.8-10)

The 2020 DNNP Site Preparation Licence Renewal Activity Report NK054-REP-01210-00110 (Reference 2.8-4) concluded the baseline conditions have not changed since the DNNP EA that was conducted in 2009 NK054-REP-07730-00029 (Reference 2.8.1) – a conclusion that is confirmed in the 2022 EIS Review Report NK054-REP-07730-00055 (Reference 2.8-10).

The impact of baseline characteristics of the DNNP site and surrounding environment on dispersion of radioactive material are summarized as follows:

- Impact of meteorology and climate, including Temperature Normals, Precipitation Normals, and Wind Speed and Direction – Subsection 2.8.1
- Impact of hydrology and hydrogeology – Subsection 2.8.2
- Impact of land cover and use – Subsection 2.8.3
- Impact of population, including numbers, locations, ages, and critical groups – Subsection 2.8.4
- Impact of accident scenarios and dispersion models – Subsection 2.8.5
- Impact of biological data – Subsection 2.8.6

Table 2.8-1 lists key characteristics and parameters within the Survey Areas of 10 km and 30 km of the Darlington Nuclear site that encompasses both the Darlington Nuclear Generating Station (DNGS) and DNNP sites.

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Table 2.8-1: DNNP and Darlington Nuclear Sites Characteristics and Parameters

Characteristic	Value/Description			
2.8.1 Meteorology and Climate				
2.8.1 Climate	Humid with four distinct seasons, uniform precipitation year-round, delayed spring and autumn, moderate temperatures in winter and summer			
2.8.1.1 Temperature Normals	Local Oshawa/Bowmanville Meteorological Stations) Mean Highest	July 4-y monthly average	21.5 °C	
	Local (Oshawa/Bowmanville Meteorological Stations) Mean Lowest	January 4-y monthly average	-4.1 °C	
	Regional (Toronto Meteorological Station) Mean Highest	July 4-y monthly average	21.5 °C	
	Regional (Toronto Meteorological Station) Mean Lowest	January 4-y monthly average	-4.1 °C	
	Mean Daily Maximum	August 2016	23.0 °C	
	Mean Daily Minimum	January 2019	-6.4 °C	
2.8.1.2 Precipitation Normals	Average annual	866 mm (of which <11% snowfall)		
	Total monthly average	From 50.5 mm in February to 98.7 mm in September		
2.8.1.3 Wind Speed and Direction	Predominant (Average wind frequency at 10m height)		ENE (wind from WSW)	
	Average Speed		2.4 m/s (Calm winds of <2 m/s were reported 37% of time)	
		Direction Wind Blowing From	Darlington Nuclear Wind Frequency (%)	
		N	7.22	
		NNE	3.09	
		NE	3.65	
		ENE	8.48	
		E	8.25	
		ESE	4.60	
		SE	3.43	
		SSE	2.25	
		S	2.33	
		SSW	2.35	
		SW	6.65	
		WSW	9.18	
		W	9.98	
		WNW	8.34	
		NW	9.82	
NNW		10.38		
Total	100			

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Table 2.8-1: DNNP and Darlington Nuclear Sites Characteristics and Parameters

Characteristic	Value/Description	
2.8.2 Impact of Hydrology and Hydrogeology		
2.8.2.1 Impact of Hydrology	Lake current	Easterly – near shore Speed in all direction 9 to 18 cm/s
	Lake water temperature	Surface – Freezing to 20 °C Ambient (Winter) – 0.5 °C in January to 7.7°C in November
	Surface Drainage	South of railway – slopes toward Lake Ontario Northeast of railway – slopes toward the east
	Stormwater	Collected in natural channels or swales and constructed outfalls and conveyed to the lake; or ponds
2.8.2.2 Existing Hydrogeological Conditions	Groundwater aquifers	South of railway – north to south Northeast of railway – toward the east Flows are impacted by subsurface structures of BWRX-300 facility.
	Urban areas water supply	Municipal water supply for Lake Ontario
	Rural areas water supply	Surface water intake (lakes) or ground water wells
2.8.3 Impact of Land Cover and Use		
Terrain Type – Water	Lake Ontario – South of the site from the E to the WSW sectors	
Terrain Type – Ploughed land	Within 3 km – Open grassland, farmland, residential homes, parking lots, and industrial land with low-elevation or low-density buildings to the north of the site from the W to the ENE sectors	
Cities	All are farther than 3 km: W and WNW – Oshawa, Whitby, NW – Courtice, and NE – Bowmanville	
Rural Areas	With tall trees, North of the site – NW to NNE, and ENE sectors	
Ecological Features	Meadow (24%), thicket (14%), woodland (5%), and swamp (5%)	
Vegetation communities	Bluff communities	West and east – cover <1% of the Darlington Nuclear site, shrubs with 10% tree cover
	Beach communities	Cover <1% of the Darlington Nuclear site, exposed to the lake with patchy vegetation cover
	Forested areas	Cover about 3% of the Darlington Nuclear site, with 60% tree cover with variable substrate types and conditions
	Cultural communities (resulting from cultural or anthropogenic disturbances)	Cover much of the site, include meadows (24%), thickets (14%), woodlands (5%)

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Table 2.8-1: DNNP and Darlington Nuclear Sites Characteristics and Parameters

Characteristic	Value/Description	
	March areas and swamps	3.7% and 5.4%, respectively of the Darlington Nuclear site
Land use in Durham Region and Clarington Municipality	Variety of landscape and lakeshore communities of small rural towns, as well as villages, hamlets, and farm holdings in the northern portion	
	Residential, industrial, and commercial areas	Generally located in Courtice (6.4 km NW of the site), and Bowmanville (4 km NE of the site)
	Agriculture	Predominant land use in Clarington
2.8.4 Impact of Population (Based on Site-specific Survey (2018) and Pathway Analyses (2016))		
Numbers (2016 census)	Within 30 km	<ul style="list-style-type: none"> - Approximately 500,000 within 30 km radius (88% WSW to NNW, 12% E to NE, and 0.0% [Lake Ontario] SW to E of the site) - 90% of population reside in urban areas
	Within 10 km	Approximately 100,000 residents
	0 to 2.0 km	Only 20 residents
By age (2016 census)	Durham Region	Children (aged under 15) (18%), Young persons (aged 15-29) (19%), Adults (aged 30-64) (49%), Older adults (aged 65+) (14)
By Gender (2018 survey)	Ontario	Largest age group is 20 to 24 for males; 55 to 59 for females
	Durham Region	Largest age group is 50 to 59 for males; 50 to 54 for females
Public Dose Assessment	Critical Groups (site-specific surveys) (NOTE: Annual site-specific survey reports dose for the top three critical groups, as well as specifically for the dairy farm potential critical group)	1. Rural Residents 2. Oshawa/Courtice Residents 3. Bowmanville Residents 4. Local Farms 5. Local Dairy Farms 6. West-East Beach Residents 7. Darlington Provincial Park Campers 8. Sport Fisher 9. Industrial/Commercial Workers
	Site-specific survey (2018) and pathway analyses (2016)	Done about every 5 years Within each critical group, 3 age classes are used – 0-5 years (Infant), 5 to 15 years (child), 16 to 70 years (adult) Group and age classes with highest dose are reported as the site dose for the given year
2.8.5 Impact of Accident Scenarios and Dispersion Models		

Table 2.8-1: DNNP and Darlington Nuclear Sites Characteristics and Parameters

Characteristic	Value/Description
	Refer to Chapter 15, Section 15.5 for DBAs and DECAs with and without core melt; as well as events related to irradiated fuel pool and fuel handling
	<p>2.8.6 Impact of Biological Data</p> <p>The baseline terrestrial flora, fauna, and food chain data as well as baseline aquatic biota and food chain data were updated in 2020 in NK054-CORR-00531-10533 (Reference 2.8-9) and did not change the conclusion of the 2009 EIS of NK054-REP-07730-00029 (Reference 2.8.1) as evidenced in the 2022 EIS documented in NK054-REP-07730-00055 (Reference 2.8-10)</p>

2.8.1 Impact of Meteorology and Climate

Meteorological characteristics are relevant to the dispersion of material in water, air, and soil as they directly impact the characteristics of the plume, including distance, direction, deposition, and ground concentrations. Relevant meteorological characteristics include temperature, precipitation as well as wind speed and direction.

The Darlington Nuclear site is in Southern Ontario on the north shore of Lake Ontario (refer to Subsection 2.1.1 for additional information). The Darlington Nuclear site displays a humid continental climate with four distinct seasons. In general, Southern Ontario climate is highly modified by the influence of the Great Lakes which results in uniform precipitation amounts year-round, delayed spring and autumn, and moderated temperatures in winter and summer, as described in D-REP-07701-00001 (Reference 2.8-2).

Refer to Section 2.6 for additional DNNP site information relevant to local and regional meteorological characteristics, hazards from meteorological events, and extreme values of meteorological parameters.

2.8.1.1 Temperature Normals

The most recent Canadian Climate Normals available span the 1981-2010 period. The meteorological stations at Oshawa and Bowmanville represent the local climate conditions at the Darlington Nuclear site, while the meteorological station at Toronto's Pearson Airport represents the regional conditions. The highest mean temperatures, both regionally and locally, occurred in July, and the lowest mean temperatures occurred in January, as shown in Table 2.8-2. Similar to the local and regional conditions, the highest (21.5 °C) and the lowest (-4.1 °C) 4-year average monthly temperatures at the Darlington Nuclear site occurred in July and January, respectively. The mean daily maximum temperature (23.0 °C) was recorded in August 2016, and the mean daily minimum temperature (-6.4 °C) was recorded in January 2019, as reported in D-REP-07701-00001 (Reference 2.8-2).

Table 2.8-2: Temperature Normals Near the Darlington Nuclear Site (Reference 2.8-2)

Month	Daily Mean (°C)				Mean Daily Maximum (°C)				Mean Daily Minimum (°C)			
	Regional Study Area	Local Study Area		Site Study Area	Regional Study Area	Local Study Area		Site Study Area	Regional Study Area	Local Study Area		Site Study Area
	TOR ¹	OSH ²	BOW ³	DN ⁴	TOR ¹	OSH ²	BOW ³	DN ⁴	TOR ¹	OSH ²	BOW ³	DN ⁴
January	-5.5	-4.8	-5.6	-4.1	-1.5	-1.1	-1.4	-1.5	-9.4	-8.5	-9.9	-6.4
February	-4.5	-3.6	-4.4	-2	-0.4	0.1	0	-0.5	-8.7	-7.3	-8.8	-4.1
March	0.1	0.4	-0.2	-0.1	4.6	4.2	4.3	2	-4.5	-3.5	-4.6	-1.1
April	7.1	6.6	6.4	5.4	12.2	10.8	11.3	8.3	1.9	2.5	1.5	3.2
May	13.1	12.3	12.4	12.6	18.8	16.9	18	13.8	7.4	7.7	6.8	11.6
June	18.6	17.6	17.5	17.8	24.2	22.3	23.1	18.3	13	12.9	11.8	17.4
July	21.5	20.6	20	21.5	27.1	25.1	25.8	22.1	15.8	15.9	14.3	20.7
August	20.6	20	19.2	21.3	26	24.3	24.8	23	15.1	15.6	13.5	19.5
September	16.2	15.9	15	18	21.6	20.2	20.4	18.8	10.8	11.7	9.5	16.8
October	9.5	9.5	8.7	11.2	14.3	13.3	13.7	13.2	4.6	5.6	3.6	9.1
November	3.7	4.2	3.4	3.3	7.6	7.4	7.2	6.3	-0.2	1	-0.4	1.1
December	-2.2	-1.2	-2.2	-1.7	1.4	2.1	1.6	-0.1	-5.8	-4.4	-6	-5.2
Year	8.2	8.1	7.5	8.6	13	12.1	12.4	10.3	3.3	4.1	2.6	6.9

1. Toronto Lester B. Pearson International Airport, 1981-2010 Climate Normals

2. Oshawa Water Pollution Control Plant (WPCP), 1981-2010 Climate Normals

3. Bowmanville Mostert Station, 1981-2010 Climate Normals

4. Darlington Nuclear, 2016-2019 (2017 data from Darlington Nuclear site on-site meteorological tower, while 2016, 2018, 2019 data from Pickering Nuclear on-site meteorological tower).

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2.8.1.2 Precipitation Normals

The Bowmanville climate station is the closest to the Darlington Nuclear site. The 1981-2010 climate normal precipitation data, listed in Table 2.8-3, from the Bowmanville Mostert Station are used to characterize precipitation patterns for the Darlington Nuclear site. During this period the Bowmanville station reported an average annual precipitation of approximately 866 mm; with snowfall representing less than 11% of the total precipitation measured. Total monthly precipitation averages range from approximately 50.5 mm in February to approximately 98.7 mm in September, per D-REP-07701-00001 (Reference 2.8-2).

Table 2.8-3: Precipitation at Bowmanville Mostert Station (1981-2010)

Month	Monthly Averages			Daily Extremes		
	Precipitation (mm)	Rain (mm)	Snow (cm)	Precipitation (mm)	Rain (mm)	Snow (cm)
January	63.1	32.2	31	46.2	46.2	29
February	50.5	32.8	17.7	42.2	42.2	19.4
March	55	41	14.1	47.6	47.6	20.8
April	70.6	68	2.6	43.4	43.4	10.2
May	75.9	75.9	0	36.4	36.4	0
June	83.8	83.8	0	50.6	50.6	0
July	63.2	63.2	0	51.1	51.1	0
August	78.1	78.1	0	81.2	81.2	0
September	98.7	98.7	0	84	84	0
October	70.8	70.6	0.1	48.6	48.6	12.2
November	88.6	83.1	5.6	71.4	71.4	15.5
December	68.1	46.1	22	41.1	41.1	24
Annual Total	866.4	773.5	93.1	-	-	-

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2.8.1.3 Wind Speed and Direction

As discussed in the annual EMP report N-REP-03443-10027 (Reference 2.8-3), the wind speed, direction, and frequency are measured continuously at meteorological towers at the Darlington Nuclear site. As shown in Table 2.8-4 for the year 2021, the landward sector at the Darlington Nuclear site the wind predominantly blew toward was the ENE sector (wind from WSW), based on the average annual wind frequencies at a 10 m height. Over all sectors, the wind predominantly blew from the north and west sectors. The dominant wind direction was NNW (10.38% of the time), followed by W (9.98% of the time) and NW (9.82% of the time).

Table 2.8-4: Darlington Nuclear – 2021 Annual Average Wind Frequency by Direction (at 10 m height)

Direction Wind Blowing From	Darlington Nuclear Wind Frequency (%)
N	7.22
NNE	3.09
NE	3.65
ENE	8.48
E	8.25
ESE	4.60
SE	3.43
SSE	2.25
S	2.33
SSW	2.35
SW	6.65
WSW	9.18
W	9.98
WNW	8.34
NW	9.82
NNW	10.38
Total	100

Notes:

- (1) Shaded fields indicate landward wind sectors.
- (2) Bolded values indicate landward wind sectors with the highest wind frequency.

As reported in the 2020 ERA for the Darlington Nuclear site D-REP-07701-00001 (Reference 2.8-2), wind speeds were measured from 2013-2019 at the Darlington Nuclear on-site meteorological towers at a height of 10 m. The average wind speed was approximately 2.4 m/s. Calm winds of less than 2 m/s were reported approximately 37% of the time. The prevailing winds for these years were measured to be from the north-west sector – the north direction (9.6% of the time) followed by the west direction (8.9% of the time). The wind rose for the 2013-2019 data is provided in Figure 2-8 of D-REP-07701-00001 (Reference 2.8-2).

2.8.2 Impact of Hydrology and Hydrogeology

Hydrological and hydrogeological characteristics are relevant to the dispersion of material in water. These characteristics influence the flow and concentration of radioactive and conventional contaminants, as well as impact the populations that are affected. Relevant characteristics include aquifer type, groundwater flow, stormwater runoff, municipal water supply sources, lake currents and temperature, and major lake water intake and discharge structures.

Refer to Section 2.5 for further information on the implication of hydrological and hydrogeological conditions, including abnormal phenomena at the DNNP site on the design and safe operation of the BWRX-300 facility.

2.8.2.1 Impact of Hydrology

There is very little current net flow along the northern shore of Lake Ontario. However, the current in the nearshore region is overall easterly and is influenced by brief patterns of strong winds exerting stress at the water surface. Lake current speeds for all directions for the 2012-2016 period typically ranged from about 9 to 18 cm/s and were typically slower during spring and early summer, (May through June) than during late summer, fall and winter (August through April), as described in the 2020 ERA D-REP-07701-00001 (Reference 2.8-2).

Lake-wide surface temperatures typically range from freezing in the winter to approximately 20 °C in the summer. Ice formation in the winter is typically limited to the nearshore areas at the eastern end of the lake within the Kingston Basin. Average ambient water temperatures in the winter have varied from 0.5 °C in January to 7.7 °C in November. The water temperatures recorded from December 2011 to March 2012 and from December 2011 to April 2012 in the Darlington Nuclear study area had an average temperature of 3.8 °C and 4.4 °C, respectively, per the 2020 ERA D-REP-07701-00001 (Reference 2.8-2).

The intake pumphouse/forebay of the BWRX-300 facility provides the transition of water flowing from the intake tunnel up to the Circulating Water System pumps (refer to Subsection 2.5.2) via an onshore vertical shaft. The intake offshore tunnel transitions into a porous veneer intake. Similarly, the submerged discharge tunnel connects to a discharge shaft that is located near the shoreline bluff, to convey returned heated water to the diffusers. Refer to Chapter 9B, Subsection 9B.3.5, for design information on the BWRX-300 pumphouse/forebay, intake and discharge shafts and tunnels, lakebed intake structure and discharge diffusers.

The surface drainage at the Darlington Nuclear site is divided by the Canadian National Railway line which runs east to west across the site (refer to Section 2.1, and Figure 2.1.1.2). The area south of the railway tracks generally slopes toward Lake Ontario while the area north of the railway tracks and east of Holt Road slopes toward the east. In the developed parts of Darlington Nuclear site including the DNGS areas, stormwater is collected in natural channels/swales and constructed outfalls and conveyed to Lake Ontario. Currently, a stormwater pond is located to the south of the Engineering Support Services Building and another pond is associated with the Darlington Waste Management Facility (DWMF). Another stormwater pond is located north of the lagoons which collect runoff from adjacent parking lots and from the railroad tracks (refer to the 2020 ERA D-REP-07701-00001 (Reference 2.8-2)). These features could change as the DNNP site is further developed, and the BWRX-300 design progresses.

To support the Site Preparation Licence renewal application in 2020, OPG obtained hydrological data, surface water data, and sediment quality data in the site, as well as in the local, and regional study areas, as provided in the 2009 DNNP EIS NK054-REP-07730-00029 (Reference 2.8-1).

The 2022 EIS in NK054-REP-07730-00055 (Reference 2.8-10) reports that the BWRX-300 deployment will have no residual adverse effects on site drainage and identified minor changes

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in DNNP flows and the number of days per year that an area of land is wet can be mitigated using best industry practices.

2.8.2.2 Existing Hydrogeological Conditions

The information on existing groundwater conditions discussed in the 2020 ERA D-REP-07701-00001 (Reference 2.8-2) and the 2009 DNNP Supporting Environment Studies NK054-REP-01210-0001 (Reference 2.8-5) is detailed in Subsection 2.5.5.

Inside the protected area at DNGS, groundwater flow is further influenced by anthropogenic subsurface features such as foundations, drain systems and sumps, and the vacuum building.

For the protected area at the DNNP, the Power Block footprint is smaller than the DNGS footprint. Also, the Reactor Building (RB) is embedded in the soil and extends to bedrock, impacting connection between groundwater flows at the north and south of the structure, per the 2020 ERA D-REP-07701-00001 (Reference 2.8-2). Such anthropogenic DNNP structures would influence the hydrostratigraphic layers and the neighboring groundwater flows. (Refer to Chapter 1, Subsection 1.5.2, and Table 1.5.2 for dimensions of the RB and other buildings in the Power Block).

Recharge of precipitation is expected to be low at the Darlington Nuclear site in areas where till is encountered at surface. Within these areas most precipitation runs off to surface water ditches or yard drainage features, as described in the 2020 ERA D-REP-07701-00001 (Reference 2.8-2). (Refer to Subsection 2.5.3 for additional information on potential sources of flooding).

Since the Site Preparation Licence renewal application in 2020 included in NK054-CORR-00531-10533 (Reference 2.8-9), OPG examined groundwater flow characteristics at the Darlington Nuclear site as part of annual groundwater monitoring (refer to Subsection 2.5.5.3). Furthermore, additional geotechnical investigations are completed for the DNNP's onshore Power Block area, with the results documented in the 2022 NK054-REP-10180-00001 DNNP Geotechnical and Seismic Hazard Investigation Plan – Phase 1 (Reference 2.8-11).

Groundwater on the Darlington Nuclear site is not used as drinking water and is not considered to be potable.

Annual groundwater quality monitoring (described in Subsection 2.5.5.3) is carried out across the site study area. Recent monitoring results, such as the levels of tritium, Volatile Organic Components, Benzene, Toluene, Ethylbenzene and Xylene, Petroleum Hydrocarbons, sodium, chloride, and metals in groundwater, are used to establish the groundwater quality baseline. Based on the annual groundwater monitoring results for the period of 2019 to 2021, groundwater quality remains consistent with that documented in the licence to prepare site application, per the 2020 Site Preparation Licence Renewal Activity Report NK054-REP-01210-00110 (Reference 2.8-4). The tritium concentrations at the sampled perimeter groundwater locations remained low in 2021. This is aligned with a trend observed indicating the tritium levels over time have remained nearly steady or decreased, which indicates stable or improved environmental performance. The groundwater quality results were compared to the Ministry of Environment, Conservation and Parks' Provincial Water Quality Objectives, based on the assumption that groundwater pumped during construction or in the long term will be discharged to the natural environment. Some groundwater samples exhibited elevated concentrations of total metals, dissolved metals, phenols, and toluene above the selected Provincial Water Quality Objectives. Several samples exhibited pH outside the acceptable Provincial Water Quality Objectives range of 6.5 to 8.5. However, given that the water is not used for drinking and is not considered potable, the conclusions of the original Site Evaluation, reported in the 2020 renewal of licence to prepare site application NK054-REP-01210-00110 (Reference 2.8-4), are valid.

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Urban areas such as Bowmanville to the east and Courtice to the west of the Darlington Nuclear site rely on municipal water supply from a Lake-Ontario-based source. The more rural areas of Durham are supplied by individual water supply systems from either surface water intakes (lakes) or ground water wells. There are rural and farm residents in rural areas in all landward wind sectors around the site at distances of about 2 km to 5 km. Residents in these areas obtain at least a portion of their water supply from wells, and use it for drinking, bathing, and irrigation. However, there are no potable groundwater supply wells within or downgradient of potential source areas on-site. As water on the Darlington Nuclear site is not used for human consumption, the only on-site pathway for human exposure to groundwater would be from ingestion of water from Lake Ontario after dilution of the groundwater in the lake. Off-site drinking water wells are influenced by atmospheric tritium, but this makes a negligible contribution to dose. Concentrations of potential chemical stressors in off-site drinking water wells are not influenced by the Darlington Nuclear site, refer to the 2020 ERA D-REP-07701-00001 (Reference 2.8-2).

2.8.3 Impact of Land Cover and Use

Land cover and use characteristics are relevant to the dispersion of material in water, air and soil as these characteristics define the terrain cover and impact deposition. Relevant characteristics include terrain type, vegetation type, vegetation height, building height, and locations.

The terrain cover surrounding the Darlington Nuclear site is broadly characterized for air dispersion modelling (refer to Subsection 2.8-5) in the Derived Release Limits and Environmental Action Levels for DNGS NK38-REP-03482-10001 (Reference 2.8-6). The major terrain types are as follows:

- Water: Lake Ontario to the south of the site from the E to the WSW sectors
- Ploughed Land: At the site boundary to a distance of 3 km, open grassland, farmland, residential homes, parking lots, and industrial land with low-elevation or low-density buildings to the north of the site from the W to the ENE sectors

At distances further than 3 km from the site boundary, inspection of aerial photographs shows cities with larger buildings, including Oshawa and Whitby to the W and WNW of the site, and Bowmanville to the NE of the site. Rural areas with tall trees, including Ganaraska Forest, are located north of the site from the NW to the NNE sectors and ENE sectors.

The dominant ecological feature of the Darlington Nuclear site is meadow (24%), followed by thicket (14%), woodland (5%), and swamp (5%). In general, the Darlington Nuclear site has four main areas, per NK054-REP-01210-0001 (Reference 2.8-5):

1. In the northwest there are sports fields, a large settling pond (Coot's Pond), and Bobolink Hill comprised of cultural meadow and cultural thicket
2. In the northeast there are agricultural fields, cultural thicket, and deciduous forest as well as three constructed wetland ponds (Treefrog, Dragonfly and Polliwog ponds)
3. In the southeast there are mostly cultural meadows
4. In the south centre and southeast is the DNGS

There are various terrain types and vegetation communities on or immediately surrounding the Darlington Nuclear site, including bluffs, beach, forest, cultural woodland, cultural meadow, cultural thickets, marshland, swamp, and urban areas. The dominant vegetation cover surrounding the Darlington Nuclear site relates to agricultural use, including row crops and pastureland, as detailed in D-REP-07701-00001 (Reference 2.8-2).

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Bluff communities are present west and east of the Darlington Nuclear site and cover a very small portion (<1%) of the Darlington Nuclear site. Bluff communities are characterized by variable vegetation cover that can range from patchy and barren to herbaceous cover. Generally, bluffs have no more than 10% tree cover because of erosion which results in steep, sometimes near vertical faces that are more than 2 meters in height. The bluff community on the west side of the Darlington Nuclear site is dominated by shrubs, mostly willows with Red-Osier Dogwood and Nannyberry. The bluff community on the east side of the Darlington Nuclear site is characterized by open or sparsely vegetated land due to ongoing erosional disturbance. The most abundant vegetation on these bluffs is Colt's Foot, refer to D-REP-07701-00001 (Reference 2.8-2).

The beach community covers a very small fraction (<1%) of the Darlington Nuclear site and much of the area is relatively exposed to the lake. The beach community is characterized by patchy vegetation cover that varies from sparse cover to areas with treed cover equal to or less than 60%, as described in D-REP-07701-00001 (Reference 2.8-2).

Forested areas cover about 0.16 km² (about 3%) at the Darlington Nuclear site. The forest community is characterized by a high level of tree cover (more than 60%) as well as variable substrate types and conditions and is classified as a coniferous, deciduous, or mixed forest type, as detailed in D-REP-07701-00001 (Reference 2.8-2).

Much of the Darlington Nuclear site vegetation communities are characterized as cultural communities such as cultural meadows, thickets, and woodlands (including plantations) that generally resulted from or are maintained by cultural or anthropogenic disturbances. Cultural woodlands, meadows, and thickets arise following anthropogenic disturbance. Cultural woodlands cover approximately 5% of the Darlington Nuclear site. They are characterized by a relatively open canopy (less than 60% cover). Cultural meadows cover approximately 24% of the Darlington Nuclear site. There are many types of cultural thickets that cover approximately 14% of the Darlington Nuclear site. They are formed during early successional stages following anthropogenic disturbance. Shrubs generally comprise the bulk of the vegetation cover and include a high proportion of non-native species, refer to D-REP-07701-00001 (Reference 2.8-2) for additional information.

Marsh areas cover over approximately 0.2 km² on the Darlington Nuclear site, or 3.7% of the total area. Swamp areas are the most dominant of the Wetland Community Classes at the Darlington Nuclear site, covering approximately 0.25 km², or 5.4% of the total Darlington Nuclear site. Swamps are characterized by the presence of wetland trees and shrubs and a low proportion of tree and shrub cover, as reported in D-REP-07701-00001 (Reference 2.8-2).

Durham Region is characterized by a variety of landscapes and communities including major lakeshore urban communities in the southern portion, and small rural towns, villages, hamlets and farm holdings in the northern portion of the region. Urban land uses are generally parallel the shoreline of Lake Ontario in the communities of Pickering, Ajax, Whitby, Oshawa and Clarington, while rural land uses are found in the communities of Brock, Scugog and Uxbridge in the northern portion of the region, all are described in D-REP-07701-00001 (Reference 2.8-2).

Urban land uses in the Municipality of Clarington, including residential, commercial, and industrial, are generally located in Courtice, located approximately 6.4 km northwest of the Darlington Nuclear site, and Bowmanville, located approximately 4 km northeast of the site. Agriculture is a predominant land use in the Municipality of Clarington and is less predominant in the City of Oshawa west of the site, per D-REP-07701-00001 (Reference 2.8-2). (Refer to Subsection 2.1.1 for recent and forecast land use data for the Municipality of Clarington and the City of Oshawa.)

2.8.4 Impact of Population

Population characteristics are relevant to the determination of the potential effects of the dispersion of material in water, air, and soil as the dispersion of radioactive and conventional contaminants affects the population surrounding the Darling Nuclear site. Relevant characteristics include population numbers, locations, ages, and critical groups.

The census data for the region used in the most recent Review of the Darlington Nuclear Site-Specific Survey, reported in NK38-REP-03443-10004 (Reference 2.8-7), are for 2016.

A population of approximately 500,000 resides within a 30 km radius of the Darlington Nuclear site, based on 2016 census data shown in Table 2.8-5. The bulk of this population (approximately 88% or 478,634 individuals) resides west of the Darlington Nuclear site, in the west-south-west to north-north-west sectors, while approximately 12% (64,575 individuals) reside east of the Darlington Nuclear site in the north to east north-east sectors. Areas south and east of the Darlington Nuclear site (south-west to east) are occupied by Lake Ontario. Only 20 residents reside within a 0 to 2 km radius of the centre of Darlington Nuclear site and approximately 99,953 individuals reside within 10 km of the Darlington Nuclear site, as documented in D-REP-07701-00001 (Reference 2.8-2).

The majority of residents in the Durham Region live in urban areas. Over 90% of the population in Pickering, Ajax, Oshawa, and Whitby resides in urban areas, whereas, the townships of Brock, Scugog and Uxbridge represent the greatest percentage of the rural population in Durham. Urban/rural population trends for Durham indicate this trend will continue into 2031, per D-REP-07701-00001 (Reference 2.8-2).

Children under the age of 15 comprised 18.0% of Durham's population in 2016, while young persons (aged 15-29), adults (aged 30-64) and older adults (aged 65+) comprised 19.2%, 49.4% and 14.4%, respectively. Ontario Population Estimates for 2018 indicate the 20 to 24 age group is the largest age group for males and 55 to 59 for females in Ontario, while in Durham Region the largest age group was 50 to 59 for males and 50 to 54 for females, refer to D-REP-07701-00001 (Reference 2.8-2).

In public dose assessments, "critical groups" are used to estimate the mean realistic impacts of emissions on the most affected individuals. The site-specific surveys identify the potential critical groups for Darlington Nuclear site. Approximately every five years the site-specific surveys and pathway analyses are reviewed to ensure the public dose accurately represents the public living near Darlington Nuclear site. Site-specific surveys were most recently reviewed in 2018 and pathway analyses were last updated in 2016. The EMP design reviews were conducted in 2018, and minor changes are implemented in 2019 which primarily affect which potential critical groups are used for reporting purposes, as documented in N-REP-03443-10027 (Reference 2.8-3).

An individual with the average characteristics of the critical group is known as the "Representative Person" as described in CSA N288.1-14 (Reference 2.8-8). Dose estimates are calculated for a number of potential critical groups for Darlington Nuclear site, and for three age classes within each potential critical group. The three age classes are 0-5 years (infant), 6-15 years (child), and 16-70 years (adult). The dose estimates to these three age groups are sufficient to characterize doses to the public. For practical implementation in dose calculations, the dose coefficients, and characteristics for a one-year-old infant, a 10-year-old child, and an adult are used to represent the three age classes. The group and age class with the highest dose is reported as the site public dose for the given in year, as described in N-REP-03443-10027 (Reference 2.8-3). (Refer to Subsection 2.9.1.2 for information on radiological dose to the public).

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Nine potential critical groups are identified for the Darlington Nuclear site. The list of potential critical groups around Darlington Nuclear site includes the following, per NK38-REP-03443-10004 (Reference 2.8-7):

1. Rural Residents
2. Oshawa/Courtice Residents
3. Bowmanville Residents
4. Local Farms
5. Local Dairy Farms
6. West-East Beach Residents
7. Darlington Provincial Park Campers
8. Sport Fisher
9. Industrial/Commercial Workers

The annual public dose is calculated for specific three potential critical groups only, which have yielded the highest dose estimates in recent years. These are the Farms, the West/East Beach Residents, and the Rural Residents, as described in N-REP-03443-10027 (Reference 2.8-3). Additionally, the annual public dose is also calculated for the local dairy farm potential critical group as the dairy farm group is exposed to the most media types and pathways.

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Table 2.8-5: Population Distribution Surrounding Darlington Nuclear Site Based on 2016 Census Data (Reference 2.8-2)

Direction	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
0-2 km	20	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	20
2-4 km	10	3,516	0	0	0	0	0	0	0	0	0	0	0	10	69	50	3,655
4-6 km	1,612	6,803	5,037	315	0	0	0	0	0	0	0	0	0	1,611	1,646	589	17,613
6-8 km	569	14,691	5,809	314	0	0	0	0	0	0	0	0	5	13,936	10,172	247	45,743
8-10 km	751	1,507	196	1,217	0	0	0	0	0	0	0	0	7,389	15,749	5,729	384	32,922
10-12 km	897	221	462	5,004	0	0	0	0	0	0	0	0	15,568	29,781	7,768	251	59,952
12-14 km	390	129	398	3,375	0	0	0	0	0	0	0	0	7,115	27,662	15,599	412	55,080
14-16 km	436	734	943	875	0	0	0	0	0	0	0	0	9,013	21,052	7,294	214	40,561
16-22 km	850	873	691	1,287	0	0	0	0	0	0	0	732	50,773	60,986	4,655	1,394	122,241
22-30 km	1,224	1,562	981	876	0	0	0	0	0	0	0	7,998	141,667	6,853	2,705	1,556	165,422
Total	6,759	30,036	14,517	13,263	0	0	0	0	0	0	0	8,730	231,530	177,640	55,637	5,097	543,209

2.8.5 Impact of Accident Scenarios and Dispersion Models

Accident scenarios and associated dispersion models are described in Chapter 15, Section 15.5, for Design Basis Accidents (DBAs), Design Extension Conditions (DECs) with and without core melt, as well as for irradiated fuel pool and fuel handling events for BWRX-300 site-specific application.

2.8.6 Impact of Biological Data

The biological characteristics of the site were documented in the 2009 DNNP EIS, NK054-REP-07730-00029 (Reference 2.8-1), to support the original application of the Site Preparation Licence. The report includes both baseline of terrestrial flora, fauna and food chain data, as well as baseline aquatic biota and habitat, and food chain data. The biological characterization underwent a baseline update for the 2020 Site Preparation Licence renewal, which is provided in NK054-CORR-00531-10533 (Reference 2.8-9). The 2020 updated baseline conditions will not change the conclusion with respect to residual adverse effects of the on the environment nor the conclusions of the original Site Evaluation. The same conclusion is confirmed the recent 2022 EIS documented in NK054-REP-07730-00055 (Reference 2.8-10).

2.8.7 References

- 2.8-1 NK054-REP-07730-00029 R000, 2009, "Environmental Impact Statement New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.
- 2.8-2 D-REP-07701-00001 R001, 2020 "Environmental Risk Assessment for the Darlington Nuclear Site," Ontario Power Generation.
- 2.8-3 N-REP-03443-10027 R000, "Results of Environmental Monitoring Programs," Ontario Power Generation.
- 2.8-4 NK054-REP-01210-00110 R001, 2020, "DNNP – Site Preparation Licence Renewal Activity Report – Environment," Ontario Power Generation.
- 2.8-5 NK054-REP-01210-0001 R000, 2020, "Darlington New Nuclear Project Supporting Environment Studies – Environment," Ontario Power Generation.
- 2.8-6 NK38-REP-03482-10001 R002, "Derived Release Limits and Environmental Action Levels for Darlington Nuclear Generating Station," Ontario Power Generation.
- 2.8-7 NK38-REP-03443-10004 R001, 2021, "Review of the Darlington Nuclear Site-Specific Survey," Ontario Power Generation.
- 2.8-8 CSA N288.1-14, "Guidelines for calculating derived release limits for radioactive material in airborne and liquid effluents for normal operation of nuclear facilities," CSA Group.
- 2.8-9 NK054-CORR-00531-10533, 2020, "Application for Renewal of OPG's Darlington New Nuclear Project (DNNP) Nuclear Power Reactor Site Preparation License (PRSL)," Ontario Power Generation.
- 2.8-10 NK054-REP-07730-00055 R000, 2022, "Darlington New Nuclear Project Environmental Impact Statement Review Report For Small Modular Reactor BWRX-300," Ontario Power Generation.
- 2.8-11 NK054-REP-10180-00001 R000, (GOLDER 2022), "Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project", Volume 2 of 2 "Geotechnical Interpretation of Design Parameters," Ontario Power Generation.

2.9 Radiological Conditions Due to External Sources

Section 2.9 details information on:

- Radiological Conditions in the Environment – Subsection 2.9.1, including
 - Radiological Baseline Conditions – Subsection 2.9.1.1
 - Radiological Dose to Public Due to Activities on DNGS Site – Subsection 2.9.1.2
- Radiation Monitoring Systems – Subsection 2.9.2, including
 - Environmental Monitoring Program – Subsection 2.9.2.1
 - TLD Monitoring – Subsection 2.9.2.2
 - Gamma Monitoring – Subsection 2.9.2.3
 - Effluent Monitoring – Subsection 2.9.2.4

Table 2.9-1 lists key characteristics and parameters for the radiological conditions due to sources external to the DNNP site.

Table 2.9-1: DNNP Site Radiological Conditions in 2021

Characteristic	Value/Description		
2.9.1 Radiological Conditions in the Environment			
Sources of Baseline radiation and Radioactivity	<ul style="list-style-type: none">Natural backgroundNuclear testing, nuclear facilitiesDNGS, Tritium Removal Facility, DWMF		
Radiological Emissions	Small fraction of the Derived Release Limit (DRL) <ul style="list-style-type: none">2016 to 2019 <0.01 – 0.41% of the DRLsIn 2021 <0.01 – 0.53% of the DRLs		
2.9.1.1 Radiological Baseline Conditions			
NOTE: The unit Bq/kg-C means becquerels per each kilogram of Carbon			
Air Samples – Concentrations	tritium	Range: 0.2 to 1.8 Bq/m ³	Average: 0.87 Bq/m ³
	C-14	Range: 206 to 248 Bq/kg-C	Average: 230 Bq/kg-C
	Ar-41, Xe-133, Xe-135, and Ir-192		Estimated to be below detection
Terrestrial Samples – Concentration	Average tritium	In fruits	17.8 Bq/L
		In vegetables	17.5 Bq/L
		In milk	4.3 Bq/L
		In animal feed	8.6 Bq/L
	Average C-14	In fruits	230 Bq/kg-C
In vegetables		248 Bq/kg-C	
In milk		229 Bq/kg-C	
In animal feed		236 Bq/kg-C	
Soil Sampling in 2017 (every 5 years)	<ul style="list-style-type: none">Cs-137, background values (from 1.7 to 9.0 Bq/kg) are present as results of historic weapon testing and around DNGS (5.1 to 7.2 Bq/kg)		

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Table 2.9-1: DNNP Site Radiological Conditions in 2021

Characteristic	Value/Description	
		<ul style="list-style-type: none"> Co-60 and Cs-134, due to emission from DNGS and other nuclear sites, neither detected.
Aquatic Samples – Concentration	tritium	<ul style="list-style-type: none"> All nearby water Supply plants – Average is below provincial standard of 7,000 Bq/L Bowmanville, Newcastle, and Oshawa water supply plants, range from 4.6 to 6.6 Bq/L Well Water – Average 12.0 Bq/L Lake Water – Average 9.6 Bq/L Fish – Average <3.4 Bq/L
	C-14	<ul style="list-style-type: none"> Fish – Average 243 Bq/kg-C
	C-137	<ul style="list-style-type: none"> Fish – Average 0.2 Bq/kg Sand Beach – (< 0.1) to 0.2 Bq/kg
	Co-60 and Cs-124	<ul style="list-style-type: none"> Fish – Not detected Sand Beach – Not detected
	Gross beta activities	All nearby water Supply plants – Average 1 Bq/L, which is below Health Canada Guideline for drinking water
<p>NOTES:</p> <ol style="list-style-type: none"> In 2021 ground water monitoring program, tritium concentrations at the sampled Darlington Nuclear site perimeter groundwater locations remained low. In general, tritium trends over time show levels have remained nearly steady or decreased, indicating stable or improved environmental performance Where unexpected tritium concentrations are identified, investigations are completed to determine the root cause and to implement corrective measures. Ongoing results confirm that tritium in groundwater is mainly localized within the station protected area and the site perimeter tritium concentrations remain low 		
2.9.1.2 Radiological Dose to the Public		
<p>Public dose for the Darlington Nuclear site was 0.6 µSv/year (represented by the adult farm resident critical group); which is</p> <ul style="list-style-type: none"> <0.1% of the regulatory limit of 1,000 µSv/year for a member of the public <0.1% of the background radiation around Darlington Nuclear site 		
2.9.2 Radiation Monitoring Systems		
2.9.2.1 Environmental Monitoring Program		
2.9.2.1.1 Atmospheric Sampling	tritium	Active samplers at six site boundary locations. Samples are collected and analysed monthly
	C-14	Monitored at four boundary locations and analysed each quarter
	Noble gases	8 detectors that monitor gamma radiation dose rate continuously

Table 2.9-1: DNNP Site Radiological Conditions in 2021

Characteristic	Value/Description	
2.9.2.1.2 <i>Aquatic Sampling</i>	Drinking water	Samples taken every 8-12-hour shift. Weakly composites are analysed weekly for tritium and monthly for gross beta activates
	Well water	Collected from four wells and analysed monthly for tritium
	Lake water	Sampled from two beaches and analysed monthly for tritium
	Fish	At DNGS – Muscle-tissue eight replicated target fish species are collected for tritium, C-14, Co-60, Cs-134, Cs-137, and Potassium-40 (K-40) measurements
	Sand	Samples collected from three beaches and analysed annually using gamma spectrometry to detect Cs-137
	Groundwater	81 monitoring locations are sampled each year for tritium.
2.9.2.1.3 <i>Terrestrial Sampling (tested for tritium and C-14)</i>	Fruits and Vegetables	Sampled three times from each of five locations representing the growing season
	Milk	Samples collected monthly from three dairy farms around the site
	Animal feed	Samples collected form four dairy farms with two replicates per visit. Dry feed and wet feed are collected separately
	Eggs	Sampled quarterly with three samples replicated per visit. Poultry samples collected annually with eight samples replicated per visit
2.9.2.2 Thermoluminescent Dosimeter (TLD) Monitoring		
Located around the site and off-site. TLD cards are analysed annually when they are changed. They are located around the DWMF fence line		
2.9.2.3 Gamma Monitoring System		
The automated fixed monitors provide real-time gamma dose rate measurements		
2.9.2.4 Effluent Monitoring Program		
Establishes surveillance and monitoring of effluents, refer to Chapter 20, Subsection 20.11.3.		

2.9.1 Radiological Conditions in the Environment

To characterize the potential effects of the BWRX-300 operation on the surrounding environment, the baseline conditions must first be identified, described and delineated. Baseline radiation and radioactivity in the area of the DNNP site includes:

- Natural background
- Background from anthropogenic sources (fallout from nuclear testing and releases from other nuclear sites)
- Releases from activities on the Darlington Nuclear site, including operation of the existing DNGS, Tritium Removal Facility, and DWMF

Radiological emissions from the Darlington Nuclear site, including the DWMF, represented a small fraction of the DRLs. The four-year period 2016 – 2019 emissions ranged from 0.01 to

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0.41% of the DRLs, as reported in the 2020 ERA for the Darlington Nuclear site D-REP-07701-00001 (Reference 2.9-1). The 2021 emissions were from 0.01 to 0.53% of the DRLs, as noted in the annual report on the results of the EMP N-REP-03443-10027 (Reference 2.9-2).

The radiological baseline conditions were established in the 2009 DNNP Environmental Impact Statement (EIS) – DNNP Environmental Assessment (EA) NK054-REP-07730-00029 (Reference 2.9-3). Updates to the radiological baseline conditions since the 2009 EIS-EA was conducted are discussed in detail in documentation including:

- The annual EMP report N-REP-03443-10027 (Reference 2.9-2)
- The 2020 ERA for the Darlington Nuclear site D-REP-07701-00001 (Reference 2.9-1)
- The 2020 DNNP – Site Preparation Licence Renewal Activity Report – Environment NK054-REP-01210-00110 (Reference 2.9-4)
- The 2020 DNNP Supporting Environment Studies – Environment NK054-REP-01210-0001 (Reference 2.9-5)
- The 2022 DNNP EIS NK054-REP-07730-00055 (Reference 2.9-16)

The 2020 Site Preparation Licence Renewal Activity Report NK054-REP-01210-00110 (Reference 2.9-4) concludes the radiological baseline conditions have not changed since the 2009 EIS-EA, per NK054-REP-07730-00029 (Reference 2.9-3). The same conclusion is reached in the 2022 DNNP EIS NK054-REP-07730-00055 (Reference 2.9-16). Details of these conditions are summarized in the following Subsections 2.9.1.1 and 2.9.1.2.

2.9.1.1 Radiological Baseline Conditions

The radiological baseline conditions in the area surrounding the Darlington Nuclear site are discussed in detail in the annual EMP report N-REP-03443-10027 (Reference 2.9-2), which demonstrates that all levels of radionuclides monitored around the Darlington Nuclear site remained stable since 2009 NK054-REP-07730-00029 (Reference 2.9-3). A Mann-Kendall trend analysis at the 95% confidence level did not indicate any statistically significant trends over the past 10 years for tritium in any medium sampled. For C-14, a Mann-Kendall trend analysis at the 95% confidence level over the past 10 years of data either indicated a statistically significant downward trend (C-14 in air at the Darlington Nuclear site boundary, C-14 in milk at dairy farms) or did not indicate any statistically significant trends (C-14 in fruit and vegetables, and C-14 in fish). A similar analysis was not conducted for noble gas parameters, as measurements taken at the Darlington Nuclear site boundary had average dose rates that were typically below detection limits.

Summaries are presented in the following paragraphs of the results of the annual results of the EMP report N-REP-03443-10027 (Reference 2.9-2), where sampling locations are available – as shown Figure 2.9-1.

Air Samples

Samples of air are collected to monitor the environment around the Darlington Nuclear site.

1. The 2021 tritium in air annual average concentrations measured at Darlington Nuclear site boundary locations ranged from 0.2 to 1.8 Bq/m³, with an average concentration of 0.87 Bq/m³. The 2021 annual average C-14 in air concentrations measured at Darlington Nuclear site boundary locations ranged from 206 to 248 Bq/kg-C, with an average concentration of 230 Bq/kg-C.

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2. The annual boundary average noble gas dose rate is estimated from the monthly data from each detector. The Darlington Nuclear site boundary average dose rates for Ar-41, Xe-133, Xe-135, and Ir-192 are typically below the detection limits.

Terrestrial Samples

Terrestrial baseline sampling is done in fruits and vegetables, milk, animal feed, eggs and poultry, and soil around the Darlington Nuclear site.

Fruits and Vegetables

Fruits and vegetables, the 2021 average concentration for tritium near the Darlington Nuclear site was 17.8 Bq/L in fruits and 17.5 Bq/L in vegetables. The 2021 average concentration of C-14 was 230 Bq/kg-C in fruits and 248 Bq/kg-C in vegetables. A Mann-Kendall trend analysis of average fruit and vegetable activity at the 95% confidence level did not indicate any statistically significant trend over the past 10 years for tritiated water tritium and C-14.

Milk

The 2021 average concentration of tritium was 4.3 Bq/L based on three dairy farms around the Darlington Nuclear site. The 2021 average concentration of C-14 in milk from dairy farm locations in the vicinity of the Darlington Nuclear site was 229 Bq/kg-C. A Mann-Kendall trend analysis of average milk activity at the 95% confidence level did not indicate any statistically significant trend over the past 10 years for tritium and C-14.

Animal Feed

The average tritium concentration was 8.6 Bq/L for wet feed (forage). No dry feed samples were available in 2021. The average C-14 concentration in animal feed was 236 Bq/kg-C for wet feed (forage). No trend analysis was performed on animal feed since, beginning in 2013, wet feed and dry feed have been sampled separately, resulting in changes to sampling frequency and replicates.

Eggs and Poultry

The concentration of tritium in eggs was 4.4 Bq/L and tritium in poultry was 10.3 Bq/L. Concentration of C-14 in eggs was 230 Bq/kg-C and in poultry was 229 Bq/kg-C. No trend analysis was performed as less than 10 years of data have been collected from sampling locations thus far.

Soil

Soil is sampled every five years to identify possible radionuclide accumulation over time. The last soil sampling took place in 2017. Background values of Cs-137 are present in the soil as a result of historic weapons testing fallout. Co-60 and Cs-134, if detected, would be a result of emissions from the DNGS or other nuclear stations. In 2017, Cs-137 concentrations in background soil samples taken at provincial background locations ranged from 1.7 to 9.0 Bq/kg. All measured Cs-137 concentrations at locations around the Darlington Nuclear site in 2017 were within the range of values seen at the background locations, ranging from 5.1 to 7.2 Bq/kg. There is no indication of a buildup of activity in soil. Neither Cs-134 nor Co-60 were detected in any soil samples in 2017. Therefore, the Cs-137 measured in these soil samples is from historic weapons testing fallout and not from OPG Operations, as documented in the annual EMP report N-REP-03443-10017 (Reference 2.9-6).

Aquatic Samples

Aquatic baseline sampling is done at nearby water supply plants, in well water, lake water, fish, and beach sand. As a result of the location of the Darlington Nuclear site, there are no depositional sediment locations near enough that are appropriate for sampling due to the high wave energy environment.

Water Supply Plants

The impact of tritium emissions from OPG stations on the nearby water supply plants varies depending upon their distance from the station, lake current direction, location and depth of the water supply plant intake pipe as well as general dispersion conditions. Annual average tritium levels at all nearby water supply plants are well below the Ontario Drinking Water Quality Standard of 7,000 Bq/L. Annual average tritium concentrations measured at the Bowmanville, Newcastle, and Oshawa water supply plants in 2021 ranged from 4.6 to 6.6 Bq/L. Mann-Kendall trend analysis at the 95% confidence level does not indicate any statistically significant trend for tritium at any water supply plant near Darlington Nuclear site. Annual average gross beta activity levels at water supply plants were 0.11 Bq/L. This is well below the gross beta activity screening level of 1 Bq/L, which is a drinking water level recommended by Health Canada in the Guidelines for Canadian Drinking Water Quality: Guideline Technical Document.

Well Water

The 2021 annual average tritium concentration observed in well water samples collected from the Darlington Nuclear site area was 12.0 Bq/L. Based on the past 10 years of data, a Mann-Kendall trend analysis at the 95% confidence level does not indicate any statistically significant trend for tritium in well water.

Lake Water

The 2021 annual average tritium concentration observed in lake water samples collected from two beaches near the Darlington Nuclear site was 9.6 Bq/L. Based on the past 10 years of data, a Mann-Kendall trend analysis at the 95% confidence level indicates no statistically significant trend for Darlington Nuclear site tritium in lake water.

Fish

The 2021 tritium levels in the Darlington Nuclear site diffuser fish samples averaged <3.4 Bq/L, while the annual average C-14 level in same samples was 243 Bq/kg-C. Based on the past 10 years of data, a Mann-Kendall trend analysis at the 95% confidence level does not indicate any statistically significant trend for tritium or C-14 in Darlington Nuclear site fish. Cs-134 and Co-60, which are indicative of reactor operation, were not detected in any fish samples at Darlington Nuclear site in 2021. This is similar to past years. The average Cs-137 value for fish was 0.2 Bq/kg. The presence of Cs-137 in fish is primarily due to nuclear weapons testing and not reactor operation.

Beach Sand

The average concentration of Cs-137 measured at beaches near the Darlington Nuclear site ranged from below detection (< 0.1) to 0.2 Bq/kg in 2021. Similar to previous years, there was no Co-60 or Cs-134 detected in any of the samples.

Groundwater

In 2021, Darlington Nuclear site completed its annual groundwater monitoring program to evaluate groundwater quality and flow across the site and to detect any emergent issues. Tritium

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concentrations at the sampled perimeter groundwater locations remained low. In general, tritium trends over time show that levels have remained nearly steady or decreased, indicating stable or improved environmental performance. There have been isolated cases within the DNGS protected area where tritium concentrations have shown increases, as reported in REP-07701-00001 (Reference 2.9-1). Where unexpected tritium concentrations are identified, investigations are completed to determine the root cause and to implement corrective measures. Ongoing results confirm that tritium in groundwater is mainly localized within the station protected area and the site perimeter tritium concentrations remain low.

2.9.1.2 Radiological Dose to the Public Due to Activities on DNGS Site

The radiological public dose resulting from the operation of existing facilities on the Darlington Nuclear site is calculated annually and the results are published and made available to the public in the annual report summarizing the results of the EMP, per N-REP-03443-10027 (Reference 2.9-2). The dose calculations consider all significant pathways of exposure. Such calculations use the environmental pathway and dosimetric models and parameters that are provided in CSA N288.1-14 (Reference 2.9-7). The data used in the calculations consist of measurements of radionuclides released from the facility in environmental media obtained from the results of the yearly EMP report and consider background contributions where such data are available. For pathways or radionuclides where measured environmental data are not available, the dose is modelled from measured radionuclide emissions data reported in N-REP-03443-10027 (Reference 2.9-2).

Site public dose remains a small fraction of both the annual regulatory dose limit and annual natural background radiation in the area. The results of the annual EMP report N-REP-03443-10027 (Reference 2.9-2) conclude that the 2021 public dose for the Darlington Nuclear site was 0.6 $\mu\text{Sv}/\text{year}$ (represented by the adult farm resident critical group). The Darlington Nuclear site dose is <0.1% of the regulatory limit of 1,000 $\mu\text{Sv}/\text{year}$ for a member of the public, and <0.1% of the background radiation around Darlington Nuclear site. As can be seen in the 2016-2021 EMP reports, the 2016 to 2021 public dose estimates for the critical groups are at most approximately 0.08% of the regulatory public dose limit of 1,000 $\mu\text{Sv}/\text{year}$, and at most approximately 0.06% of the dose from background radiation (1.4 mSv/year) in the vicinity of Darlington Nuclear site.

The public dose is also reported in the Darlington Nuclear site ERA, which is routinely updated in accordance with REGDOC-3.1.1 (Reference 2.9-8). A CSA N288.6-12 (Reference 2.9-9) compliant ERA was produced for the Darlington Nuclear site in 2020 D-REP-07701-00001 (Reference 2.9-1) and included a human health risk assessment and ecological risk assessment for both radiological and non-radiological parameters and physical stressors. The ERA concluded that the Darlington Nuclear site is operating in a manner that is protective of human and ecological receptors residing in the surrounding area. No discernable health effects are anticipated due to the exposure of potential critical groups to the radiological effluent from the Darlington Nuclear site. Demonstration that the critical groups are protected implies that other receptor groups near the Darlington Nuclear site are also protected.

2.9.2 Radiation Monitoring Systems

OPG's radiation monitoring systems, which are currently used for DNGS, comprise on-site, site boundary, and off-site monitoring systems. Detailed information about environmental sampling locations, sampling frequency, the number of samples taken, the media sampled, the sampling method, and the radionuclides monitored can be found in CSA N288.4 on Environmental Monitoring Programs at Nuclear Facilities and Uranium Mines and Mills (Reference 2.9-10). Summaries of four specific aspects of the radiation monitoring systems are presented as follow:

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1. Environmental monitoring systems, including the environmental off-site and site boundary monitoring as well as samples taking and analysis – Subsection 2.9.2.1
2. The off-site and site boundary TLD sites – Subsection 2.9.2.2
3. The Automated Near Boundary Gamma Monitoring System, located around the Darlington Nuclear site boundary – Subsection 2.9.2.3
4. The site Effluent Monitoring Program – Subsection 2.9.2.4

2.9.2.1 Environmental Monitoring Program

The environmental monitoring systems and sampling programs detailed in the annual EMP report N-REP-03443-10027 (Reference 2.9-2) include off-site and site boundary monitoring and are summarized here. Samples taken are analysed at certified laboratories or laboratories with documented comprehensive quality assurance and quality control programs, in accordance with clause 8.3.2 of CSA N288.4 (Reference 2.9-10). The Canadian Association for Laboratory Accreditation certified OPG Health Physics Laboratory, and external contractors, perform the sample collection and analysis for Darlington Nuclear site and provincial EMPs, as per N-PROC-OP-0025 R012 (Reference 2.9-11). Sampling locations are shown in Figure C1 in Appendix C of N-REP-03443-10027 (Reference 2.9-2), which is replicated in Figure 2.9-1.

2.9.2.1.1 Atmospheric Sampling

Concentrations in air are sampled to monitor the environment around the Darlington Nuclear site. Tritium, C-14, and noble gases are measured and reported in N-REP-03443-10027 (Reference 2.9-2).

1. The active tritium in air sampler collects water vapor by passing air continuously at a steady rate through two molecular sieve canisters in series. The active samplers are located at six site boundary EMP monitoring locations around the Darlington Nuclear site. These samples are collected and analysed monthly.
2. C-14 in air is sampled using passive sampling technology. The passive C-14 sampler works by absorption of CO₂ in air into soda lime pellets exposed for a period of an annual quarter. Samples are analysed after each quarter. C-14 in air is monitored at four boundary locations for the Darlington Nuclear site.
3. External gamma radiation doses from noble gases and Ir-192 are measured using sodium iodide (NaI) spectrometers set up around the Darlington Nuclear site. There are a total of eight detectors around the Darlington Nuclear site that monitor the dose rate continuously.

2.9.2.1.2 Aquatic Sampling

Samples of drinking water sources (municipal and well water), lake water, beach sand and fish are collected to monitor the aquatic environment around the Darlington Nuclear site. Tritium, gross beta, C-14, and gamma activity are measured and reported in N-REP-03443-10027 (Reference 2.9-2).

1. Samples of drinking water are taken during each 8-12-hour shift at water supply plants that supply water to Durham Region the Bowmanville water supply plant, the Newcastle water supply plant, and the Oshawa water supply plant. Weekly composites of these samples are analysed for tritium, and monthly composites are analysed for gross beta activity.
2. Monthly well water samples are collected from four wells around the Darlington Nuclear site area. Samples are analysed monthly for tritium.

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3. Lake water for recreational use is sampled from two beaches in the vicinity of the Darlington Nuclear site on a monthly basis and analysed for tritium. It is used to assess the water immersion dose exposure pathway from swimming in lake water.
4. At the Darlington Nuclear site, fish sampling takes place over the cooling water discharge diffuser. The target fish species to be collected at Darlington Nuclear site and at background locations is White Sucker, with Brown Bullhead as the backup species. Eight replicate fish samples are collected and analysed at each location. A sample consists of the fish muscle tissue only, and excludes the head, skin, fins, and as many bones as possible. Tritium, C-14, Co-60, Cs-134, Cs-137, and Potassium-40 (K-40) measurements are performed on each fish sample.
5. Sand from three beaches around the Darlington Nuclear site is collected annually to represent a potential pathway for external dose. Eight replicates are collected per sampling location. Gamma spectrometry is performed on these samples. Beach sand samples were collected at a background location to determine the Cs-137 concentrations in sand due to atmospheric weapons test fallout.
6. Groundwater monitoring occurs of each year, with 81 groundwater monitoring locations at Darlington Nuclear site sampled in 2021 for tritium, the key parameter of concern, refer to NK38-REP-10140-10031 (Reference 2.9-12). Annual water level measurements are also conducted.

2.9.2.1.3 Terrestrial Sampling

Samples of soil, fruits, vegetables, animal feed, milk, eggs, and poultry are collected to support the public dose calculation for the Darlington Nuclear site. Terrestrial biotas receive exposure from both airborne and waterborne emissions. Tritium and C-14 are measured, per N-REP-03443-10027 (Reference 2.9-2).

1. Fruits and vegetables are sampled three times from each location for a representation of the entire growing season. Each sample is analysed for C-14 and tritium. A total of five locations for fruit and vegetable were sampled around the Darlington Nuclear site.
2. Milk sampling is used to estimate the portion of dose received from milk ingestion for the dairy farm potential critical group. Milk samples are collected on a monthly basis from dairy farms around the Darlington Nuclear site and analysed for tritium and C-14. Samples are collected from three dairy farms around the Darlington Nuclear site.
3. Locally grown animal feed is collected from four dairy farms around the Darlington Nuclear site, twice a year, with two replicates collected per visit. Since 2013, dry feed (grains, hay, etc.) and wet feed (forage) are collected separately. Animal feed is analysed for tritium and C-14.
4. Eggs are sampled on a quarterly basis and three sample replicates are collected per visit. Poultry is collected annually with eight sample replicates collected per visit. Both eggs and poultry are analysed for tritium and C-14. One farm location around the Darlington Nuclear site is sampled for eggs.

2.9.2.2 Thermoluminescent Dosimeter Monitoring

TLDs are located around the Darlington Nuclear site perimeter as well as at off-site locations. The TLDs contain field cards that passively monitor the airborne dose over the course of a year. Cards are read and analysed annually when they are changed. The net readings for the four elements from the field card readings are input to an algorithm that converts the readings into air kerma (short for Kinetic Energy Released per unit mass of Air, which is a measure of energy in

joules (J) deposited in a unit mass (kg) of air; thus, in J/kg), ambient dose equivalent and directional dose equivalent, as described in N-PROG-RA-0001 (Reference 2.9-13).

Also, TLDs are located around the DWMF fence line. The DWMF perimeter dose rates are measured and reported quarterly.

2.9.2.3 Gamma Monitoring System

The Automated Near Boundary Gamma Monitoring System, located around the Darlington Nuclear site boundary, is a fixed radiological detection and monitoring system designed to provide real-time gamma dose rate measurements, as reported in N-PROG-RA-0001 (Reference 2.9-13). Refer to Chapter 19, Section 19.3 for additional relevant information.

2.9.2.4 Effluent Monitoring Program

The Darlington Nuclear Site Effluent Monitoring Program is governed by OPG's N-STD-OP-0031 Monitoring of Nuclear and Hazardous Substances in Effluents (Reference 2.9-14). This standard establishes minimum requirements to establish an appropriate surveillance and monitoring program for nuclear and hazardous substances in airborne and waterborne effluents from operating OPG Nuclear facilities, including the DNGS, in accordance with CSA N288.5-11 (Reference 2.9-15). The effluent monitoring program is further discussed in Chapter 20, Subsection 20.11.3.

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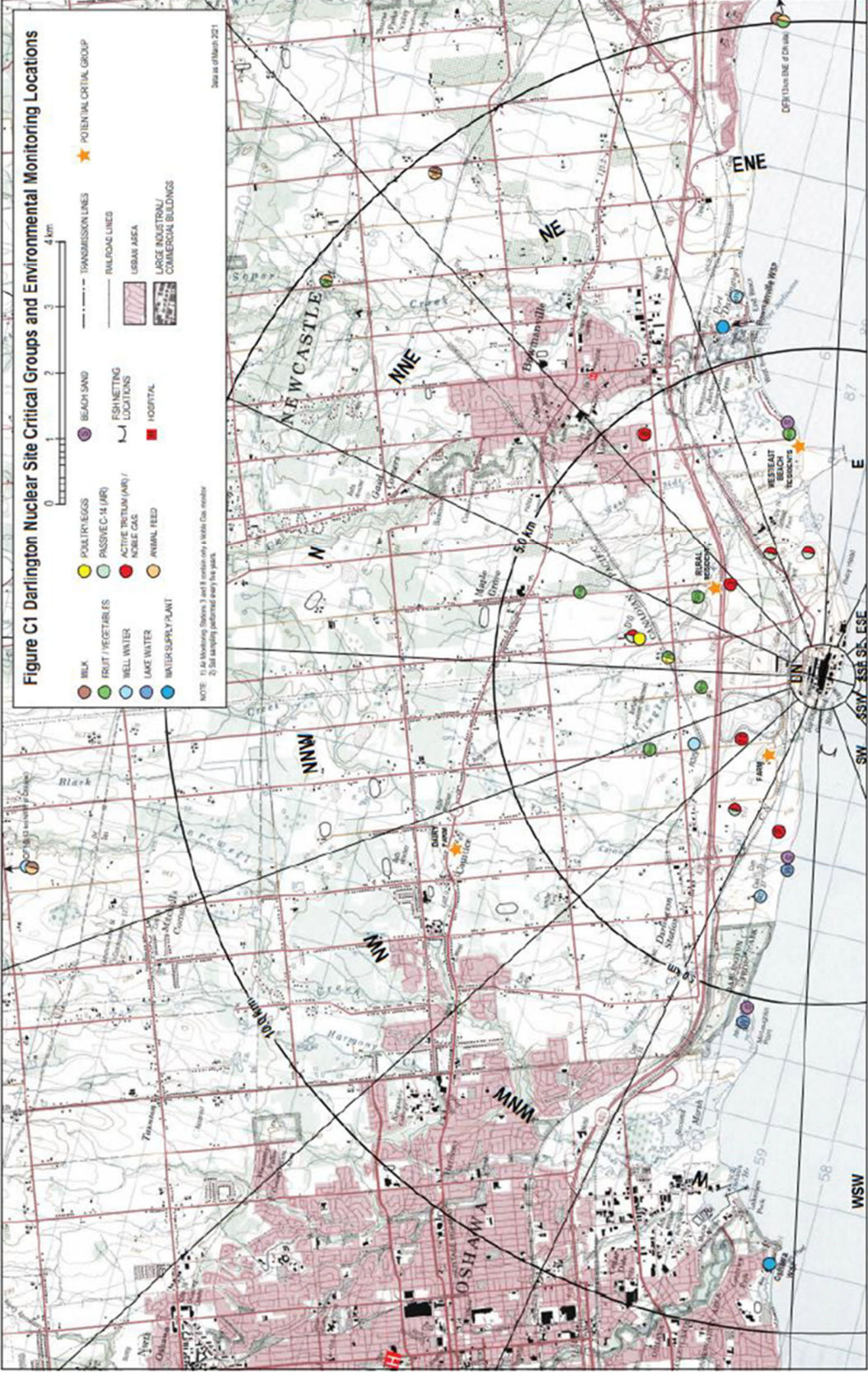


Figure 2.9-1: Darlington Nuclear Site Critical Groups and Environmental Monitoring Locations (Reference 2.9-2)

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2.9.3 References

- 2.9-1 D-REP-07701-00001 R001, 2020, "Environmental Risk Assessment for the Darlington Nuclear Site," Ontario Power Generation.
- 2.9-2 N-REP-03443-10027 R000, "Results of Environmental Monitoring Programs," Ontario Power Generation.
- 2.9-3 NK054-REP-07730-00029 R000, 2009, "Environmental Impact Statement New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.
- 2.9-4 NK054-REP-01210-00110 R001, 2020, "DNNP – Site Preparation Licence Renewal Activity Report – Environment," Ontario Power Generation.
- 2.9-5 NK054-REP-01210-0001 R000, 2020, "Darlington New Nuclear Project Supporting Environment Studies – Environment," Ontario Power Generation.
- 2.9-6 N-REP-03443-10017 R000, "Results of Environmental Monitoring Programs," Ontario Power Generation.
- 2.9-7 CSA N288.1-14 "Guidelines for calculating derived release limits for radioactive material in airborne and liquid effluents for normal operation of nuclear facilities," CSA Group.
- 2.9-8 CNSC Regulatory Document REGDOC-3.1.1, "Reporting Requirements for Nuclear Power Plants."
- 2.9-9 CSA N288.6-12, "Environmental Risk Assessments At Class I Nuclear Facilities And Uranium Mines And Mills," CSA Group.
- 2.9-10 CSA N288.4, "Environmental Monitoring Programs At Nuclear Facilities And Uranium Mines And Mills," CSA Group.
- 2.9-11 N-PROC-OP-0025 R012, "Management of the Environmental Monitoring Programs," Ontario Power Generation.
- 2.9-12 NK38-REP-10140-10031 R001, 2021, "Darlington Nuclear Groundwater Monitoring Program Results," Ontario Power Generation.
- 2.9-13 N-PROG-RA-0001 R019, "Consolidated Nuclear Emergency Plan," Ontario Power Generation.
- 2.9-14 N-STD-OP-0031 R009, "Monitoring of Nuclear and Hazardous Substances in Effluents," Ontario Power Generation.
- 2.9-15 CSA N288.5-11, "Effluent Monitoring Programs At Class I Nuclear Facilities And Uranium Mines And Mills," CSA Group.
- 2.9-16 NK054-REP-07730-00055 R000 2022, "Darlington New Nuclear Project Environmental Impact Statement Review Report For Small Modular Reactor BWRX-300," Ontario Power Generation.

2.10 Site-Related Issues in Emergency Preparedness and Response and Accident Management

The information presented in Section 2.10 includes:

- General Consideration – Subsection 2.10.1
- Feasibility of Emergency Preparedness and Response – Subsection 2.10.2
- Evacuation Time Estimates and Route – Subsection 2.10.3
- Support Networks in the Vicinity of the Site – Subsection 2.10.4
- Administrative Measures with External Organizations – Subsection 2.10.5

In Table 2.10-1, a summary description is included of site-related emergency preparedness and response feasibility, relevant evacuation time estimates; supporting agencies and services; communication systems; provincial and on-site plans; and other nuclear organization.

Table 2.10-1: Summary of DNNP Site Relevant Characteristics and Parameters

Characteristic	Value/Description		
2.10.2 Feasibility of Emergency Preparedness and Response			
Accessibility	<ul style="list-style-type: none">• Studies considered number of personnel on site, regional population change, infrastructure updates, geography, and weather patterns.• Main entrance: Holt Road South via Energy Drive, or Highway 401, or Park Road via Highway 401 to Energy Road.		
DNNP Traffic Management Plan	Developed to guide site transportation demands during various phases of project, including construction		
BWRX-300 Design	<ul style="list-style-type: none">• Incorporates reliable and passive safety functions with redundancy and diversity that satisfy safety goal requirements• Informed by DSA and Probabilistic Safety Assessment (PSA) results to develop optimized accident management strategies and measures.		
2.10.3 Evacuation Time Estimates and Route			
Estimates	<ul style="list-style-type: none">• Provides off-site emergency planners with projections on how long it may take for various emergency planning sectors and the Detailed Planning Zone (DPZ) to evacuate.• Considered various scenarios as time of day, day of week, road restrictions, special event assemblies and weather conditions.		
Routes	<ul style="list-style-type: none">• On-site process and travel route for site evacuations are documented in site-specific instructions, including DNNP site during various phases of the project.• Measures to evacuate publicly accessible areas on the Darlington Nuclear site.		
Infrastructure	Impacted local businesses and transportation networks		
2.10.4 Support Networks in the Vicinity of the Site			
Agencies, Businesses, Services, Plans	<ul style="list-style-type: none">• Ambulances and Hospital• Municipal services• Potassium Iodide Program	<ul style="list-style-type: none">• Police force• Alerting systems• PNERP• Consolidated Nuclear Emergency Plan (CNEP)	<ul style="list-style-type: none">• On-site and off-site communication systems• Information to media

Table 2.10-1: Summary of DNNP Site Relevant Characteristics and Parameters

Characteristic	Value/Description
Off-site Alerting System	Managed by Durham Region and the Province of Ontario
Designated and Host Municipalities	<ul style="list-style-type: none"> Administered the Potassium Iodide Program Provide centres for Emergency Workers, Evacuation, and Reception (with personnel and resources support provided from OPG)
2.10.5 Administrative Measures with External Organizations	
The Province of Ontario, Provincial Nuclear Emergency Response Plan (PNERP)	<ul style="list-style-type: none"> Provides the off-site planning basis for nuclear emergencies with the goal of ensuring public safety in the event of a nuclear emergency Establishes the principles, concepts, organization, responsibilities, policy, functions, and interrelationships which govern all off-site nuclear emergency planning, preparation, and response in Ontario
Other Nuclear Partners	<ul style="list-style-type: none"> Nuclear partners in Canada are expected to respond, if necessary CANada Deuterium Uranium (CANDU) Owners Group for support and technical assistance Institute of Nuclear Power Operations (INPO) for necessary support from the industry

2.10.1 General Consideration

In accordance with Subsection 4.10.2 of REGDOC-1.1.2 (Reference 2.10-17), OPG, as the licensee for the BWRX-300 facility, has established an effective DNNP Nuclear Emergency Preparedness Plan NK054-PLAN-01210-00002 (Reference 2.10-1) which is governed by OPG CNEP N-PROG-RA-0001 (Reference 2.10-2). These two plans cover aspects such as:

- Feasibility of emergency preparedness and response
- Local infrastructure for evacuation adequacy
- Availability of support networks in the vicinity of the site
- Availability of transport, communication and infrastructure external to site
- Need for administrative measures
- Roles of response organization other than OPG

Elaboration on these aspects and associated detailed information are included in the following Subsections 2.10.2 to 2.10.5.

2.10.2 Feasibility of Emergency Preparedness and Response

The BWRX-300 facility accessibility for OPG personnel, contractors, and response crews, as well as for the transport of any equipment necessary in an emergency, is critical for the purposes of emergency preparedness and response at the DNNP site. Such accessibility is considered by OPG in the design of the BWRX-300 facility for the construction, commissioning, operation, and decommissioning phases. In this regard, events at both the DNNP site and the existing DNGS site are considered since an event at one site may affect personnel and the emergency response at the other site. Emergency response is, therefore, considered for the entire Darlington Nuclear site. Protocols throughout the project phases are included in the DNNP Nuclear Emergency Preparedness Plan NK054-PLAN-01210-00002 (Reference 2.10-1).

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To ensure accessibility for both off-site emergency responders and on-site personnel to and from the DNNP site, OPG conducted studies that considered estimated number of all personnel in the Darlington Nuclear site, regional population changes, infrastructure updates, geography, and weather patterns. The results of these studies are formalized into plans and reports to assist with emergency planning; primarily in, DNGS Development of Evacuation Time Estimates, per NK38-REP-03490-10133 (Reference 2.10-3), Summary Report: Site Evaluation Studies for Nuclear Installations at Darlington External Human Induced Events NK054-REP-01210-00010 (Reference 2.10-4), Darlington New Nuclear Project Traffic Management Plan NK054-PLAN-08965.4-00001 (Reference 2.10-5) and Updated Traffic Management Plan NK054-REP-07730-0969014 (Reference 2.10-20). In addition, detailed analysis of the DNNP site accessibility is noted in Site Evaluation for OPG New Nuclear at Darlington – Nuclear Safety Considerations NK054-REP-01210-00008 (Reference 2.10-6).

The main entrance to the DNNP site is per the existing entrance to the entire Darlington Nuclear site via Holt Road South in Bowmanville, Ontario. Holt Road South is accessible via Energy Drive eastbound on Highway 401 and has a direct off-exit of Highway 401 westbound. An alternate access point from westbound Highway 401 to Energy Drive is Park Road. Park Road traverses the western part of the Darlington Nuclear site, crossing 2nd Line, which then connects to Holt Road. Energy Drive west of Park Road is named Megawatt Drive. Additional detailed information on transportation networks on the Darlington Nuclear site and in the surrounding area is provided in Subsection 2.1.5.

For the purpose of Subsection 2.10.2, a generic site map displaying the Darlington Nuclear site in relation to major roadways is shown in Figure 2.10.2-1, where the area allotted to DNNP is shaded in yellow east of the DNGS area, and the DNGS exclusion zone of 914 m is shown, per D-PLAN-00120-0001 (Reference 2.10-7).

The DNNP Traffic Management Plan (Reference 2.10-5) was initiated by OPG to guide, in a safe manner, site transportation demands during various phases of the BWRX-300 facility including construction. This Traffic Management Plan assesses the impact of traffic within the vicinity of the DNNP site, in the area noted in Figure 2.10.2-2.

Chapter 15, Subsection 15.6.1 states that the specific objectives of the PSA and severe accident analysis (SAA) are to demonstrate that the BWRX-300 facility is designed with features that incorporate highly reliable and available passive safety functions with significant redundancy and diversity to comply with the safety goal requirements in REGDOC-2.5.2 (Reference 2.10-9).

Further, as described in Chapter 15, Subsection 15.1.5, DEC's are identified to aid in designing and implementing safety features (complementary design features) to mitigate the consequences of DEC's. The Severe Accident Management (SAM) program is informed by the insights of the Deterministic Safety Analysis (DSA) and results of the PSA for the development, implementation, training and optimization of accident management strategies and measures, as identified in Chapter 15 Subsection 15.6.1.

Additionally, Chapter 13, Subsection 13.4.3 discusses the programmatic approach to develop emergency operating procedures and severe accident management guidelines (SAMG) in accordance with REGDOC-2.3.2 (Reference 2.10-21).

2.10.3 Evacuation Time Estimates and Route

OPG made available to off-site planning authorities a revised Darlington Site Evacuation Time Estimate, per NK38-REP-03490-10133 (Reference 2.10.3) using the 2016 National Census Data with per decade population projections out to 2088, as well as current and forecasted infrastructure.

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The estimate provides off-site emergency planners with projections on how long it may take for current emergency planning sectors and the DPZ to evacuate if required. Variables such as time of day, day of week, road restrictions, special event assemblies and weather were assessed as to how those factors may impact the evacuation duration. In the first quarter of 2023, OPG will issue an updated Darlington Site Evacuation Time Estimate based on the 2021 national census data and will subsequently be shared with stakeholders.

On-site, the process and travel route for site evacuations are described in D-INS-0349-10030 (Reference 2.10-10). The current revision of such OPG's instructions considers the DNNP site during various phases of the project. The main exit routes are via:

1. Park Road to Energy Drive to Highway 401 westbound
2. Old Holt Road, continuing onto Holt Road northbound of Highway 401 east and westbound

During an evacuation from the Darlington Nuclear site, Energy Drive will be closed, as necessary, by local police between Park Road to Holt Road to control traffic volume and delays. Additionally, procedures exist for OPG to evacuate publicly accessible areas on the Darlington Nuclear site, per INS-03490-10015 (Reference 2.10-11), including the Darlington Waterfront Trail and the Hydro Soccer Fields (refer to Subsections 2.1.7 and 2.1.8).

Local infrastructure within the vicinity of the DNNP site is described in Section 2.3, which includes local businesses, and transportation networks that are impacted by an emergency on-site in their current and future expanded state.

2.10.4 Support Networks in the Vicinity of the Site

Collaboration of OPG with local government agencies and businesses is essential to the DNNP emergency response capabilities. Shared roads, emergency services, communication networks, and transportation networks are utilized to assist with site response, evacuation, and relocation services, as required.

During construction, prior to turnover to Operations, the fire protection controls and response are primarily the responsibility of the prime contractor or constructor, per CSA N293-12 (Reference 2.10-18). Once handover to Operations occurs, OPG's own fire protection program, with its necessary updates for the BWRX-300 facility, will be in place and be compliant with CSA N293S1:21 (Reference 2.10-19).

Arrangements also exist for local ambulance service and hospital support for casualties from the Darlington Nuclear site. Toronto Hospital Corporation, Western Division, has been provincially designated and funded as the radiation trauma centre for Ontario. This includes the capability to deal with contaminated casualties, trauma, and acute radiation syndrome. Lakeridge Health—Bowmanville Hospital is the primary local hospital designated to receive contaminated casualties from DNGS. DNNP is expected to be included in this agreement, encompassed under the Darlington Nuclear site. Agreements are also in place to provide support to the site from the local police force in the event of an on-site security event (Reference 2.10-2). Subsection 2.1.4 and Subsection 2.1.6 provide, respectively, additional details on Municipal Services local to the area as well as on public transit.

To communicate with off-site emergency responders during an event, OPG currently uses Durham NEXGEN P-25 Radio system – part of the Durham Emergency Communication. As the DNNP progresses, and prior to Operations, these systems will be assessed for future use.

As noted in Chapter 9A, Subsection 9A.9.1.3, the off-site communication system is designed to satisfy emergency plan requirements for accident conditions, including notification of personnel and implementation of evacuation procedures. This capability includes communications support

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to both on-site and off-site emergency response facilities; at least one on-site and one off-site communications system, each with a backup power source. The on-site communications involve immediate notification process and secondary communication methods to alert all on-site personnel in all vital areas during the full spectrum of accident or incident conditions under maximum potential noise levels. This capability also includes communications support for firefighting, including support of alternative and dedicated shutdown capabilities.

As noted in Subsection 2.10.2, the SAMG is informed by the insights of the DSA and results of the PSA for the development, implementation, training and optimization of accident management strategies and measures. This includes review of the existing Beyond Design Basis Accident telecommunications equipment designated for DNGS, which also are considered for DNNP and rely on external infrastructure to function.

Durham Region and the Province of Ontario manage alerting systems to let the public know when a nuclear emergency occurs. Durham Region's public alerting system includes loud sirens, located within the Automatic Action Zone of the Darlington Nuclear site and an automated landline telephone calling system. The automated telephone system sends a recorded message to landline phones in the DPZ area around the nuclear station. The Province of Ontario manages the Alert Ready system. These alerts broadcast through television, radio, and cellphones. The off-site public alerting systems are currently applicable to DNGS but expected to be utilized for DNNP. Prior to fuel-in commissioning, this will be identified as part of the revised PNERP (Reference 2.10-12).

The DNGS station has an established Potassium Iodide Program, which satisfies the requirements of the PNERP (Reference 2.10-12) and REGDOC-2.10.1 (Reference 2.10-13), both are encompassed by the CNEP (Reference 2.10-2). The program is supported by designated municipalities to ensure continued availability of Potassium Iodide to residents of the DPZ and Ingestion Planning Zone, and information is available to the general public, including on-line, as per N-GUID-03491-10011 (Reference 2.10-14). Similar to the public alerting systems, this program is currently applicable to DNGS, but expected to be utilized for DNNP. Prior to fuel-in commissioning, this will be identified as part of the revised PNERP.

The PNERP (reference 2.10-12) outlines the requirements for designated municipalities and host municipalities to include provisions for Emergency Worker Centres, Evacuation Centres, and Reception Centres in the unlikely event of an evacuation, as noted in D-INS-0349-10030 (Reference 2.10-10). OPG supports these Off-site Centres by providing personnel and resources for personal monitoring and decontamination. The current facilities applicable to the DNGS are listed in Appendix C3.4 of CNEP (Reference 2.10-12). It is to be determined whether such facilities are required for DNNP which, if so, will be reflected in a future revision of the PNERP. Additionally, OPG has two Mobile Monitoring and Decontamination Units that are poised and ready for deployment when designated by the Provincial Emergency Operations Centre (PEOC). OPG deploys on-site and off-site radiation survey teams to the area, if required.

The Joint (Emergency) Information Centres intending to disseminate Information to the media are also set up between OPG, the Province of Ontario, and local municipalities. Refer to the CNEP (Reference 2.10-2). OPG's Nuclear Crisis Communication Standard (Reference 2.10-15) provides corporate direction for assisting with site emergencies. This standard outlines how information is passed between the incident station, emergency response facilities, Corporate Media Desk, and the public domain.

There are no known issues at this time that would hinder the implementation of DNNP emergency response actions. OPG is currently working with the Province of Ontario to develop timelines for PNERP revisions to incorporate a separate implementing plan for the DNNP site or as part of the DNGS site implementing plan.

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Descriptions of the development of the DNNP emergency response plan, and the emergency response facilities are detailed in Chapter 19, Sections 19.1 and 19.2, respectively.

2.10.5 Administrative Measures with External Organizations

In the Province of Ontario, Canada, the PNERP (Reference 2.10-12) provides the off-site planning basis for nuclear emergencies with the goal of ensuring public safety in the event of a nuclear emergency. The PNERP Master Plan (Reference 2.10.12) establishes the principles, concepts, organization, responsibilities, policy, functions, and interrelationships which govern all off-site nuclear emergency planning, preparation, and response in Ontario. Each nuclear facility identified in the PNERP has its own implementing plan which is site-specific in nature and deals with local characteristics, planning and operational particulars. OPG has a memorandum of understanding in place with the Province of Ontario to revise the PNERP prior to fuel-in commissioning to include DNNP and issue a revised Darlington implementing plan or a separate implementing plan for DNNP (Reference 2.10-1).

OPG continues to collaborate with the Province of Ontario and other external organizations responsible for off-site nuclear emergency planning to ensure the implementation of their respective emergency plans and related protective actions accommodate the lifecycle of BWRX-300 facility built on the DNNP site.

Other nuclear partners within Canada are requested to respond where necessary, for any assistance in a nuclear event at DNGS and DNNP, as per the existing mutual aid response memoranda of understanding.

OPG also has arrangements for support and technical assistance with the CANDU Owners Group members and INPO, a consortium of nuclear utilities, to obtain any necessary support available from the industry during an emergency. INPO operates a 24-hour emergency assistance line and an Emergency Response Centre in Atlanta, Georgia, USA, to provide support to member utilities.

Further information on external administrative assistance is provided in the Emergency Planning and Preparedness Technical Support Document: New Nuclear – Darlington Environmental Assessment NK054-REP-07730-00021 (Reference 2.10-8), and the DNNP Nuclear Emergency Preparedness Plan NK054-PLAN-01210-00002 (Reference 2.10-1).

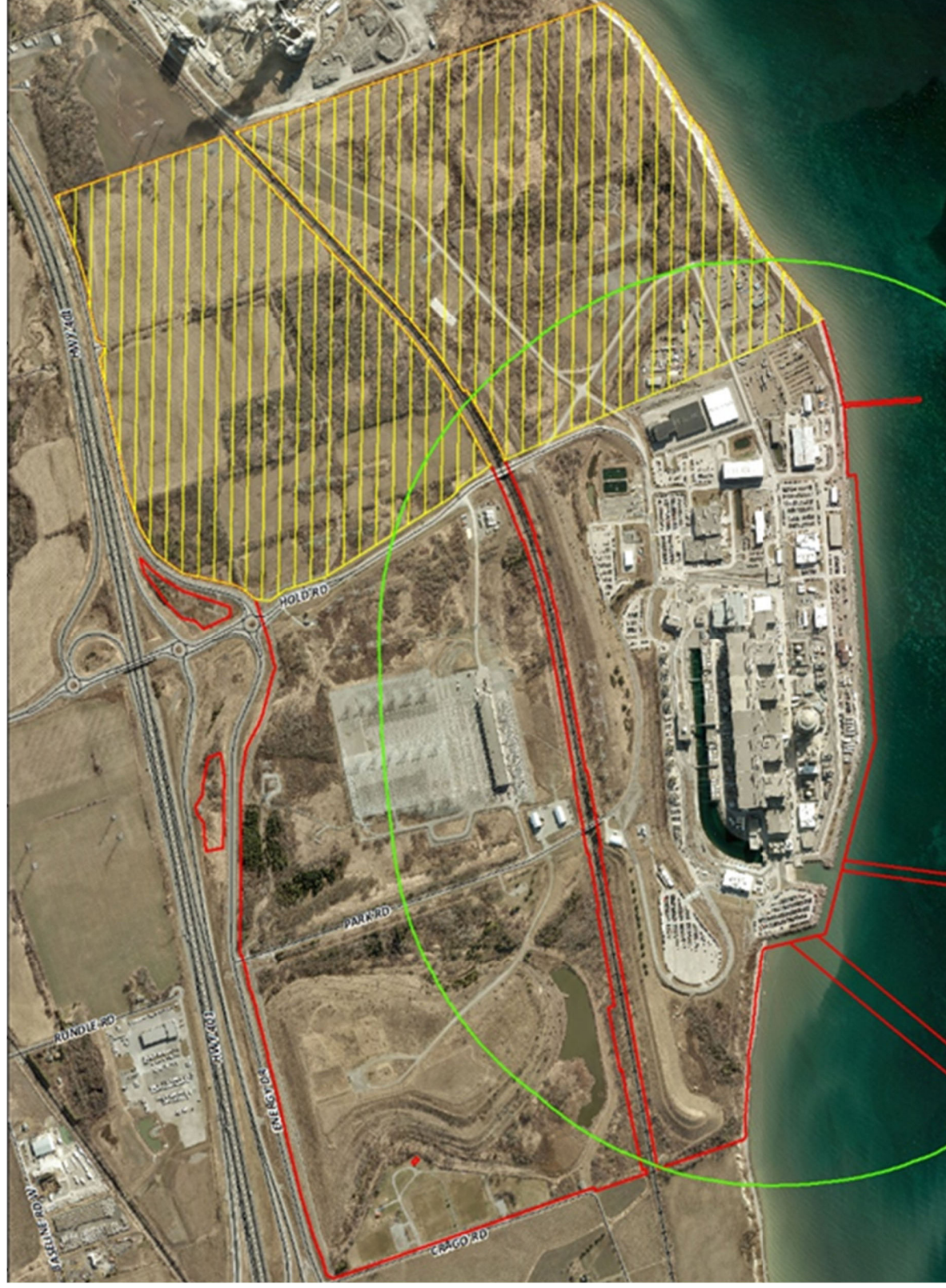


Figure 2.10.2-1: Darlington Nuclear Site Showing DNGS and DNNP Areas

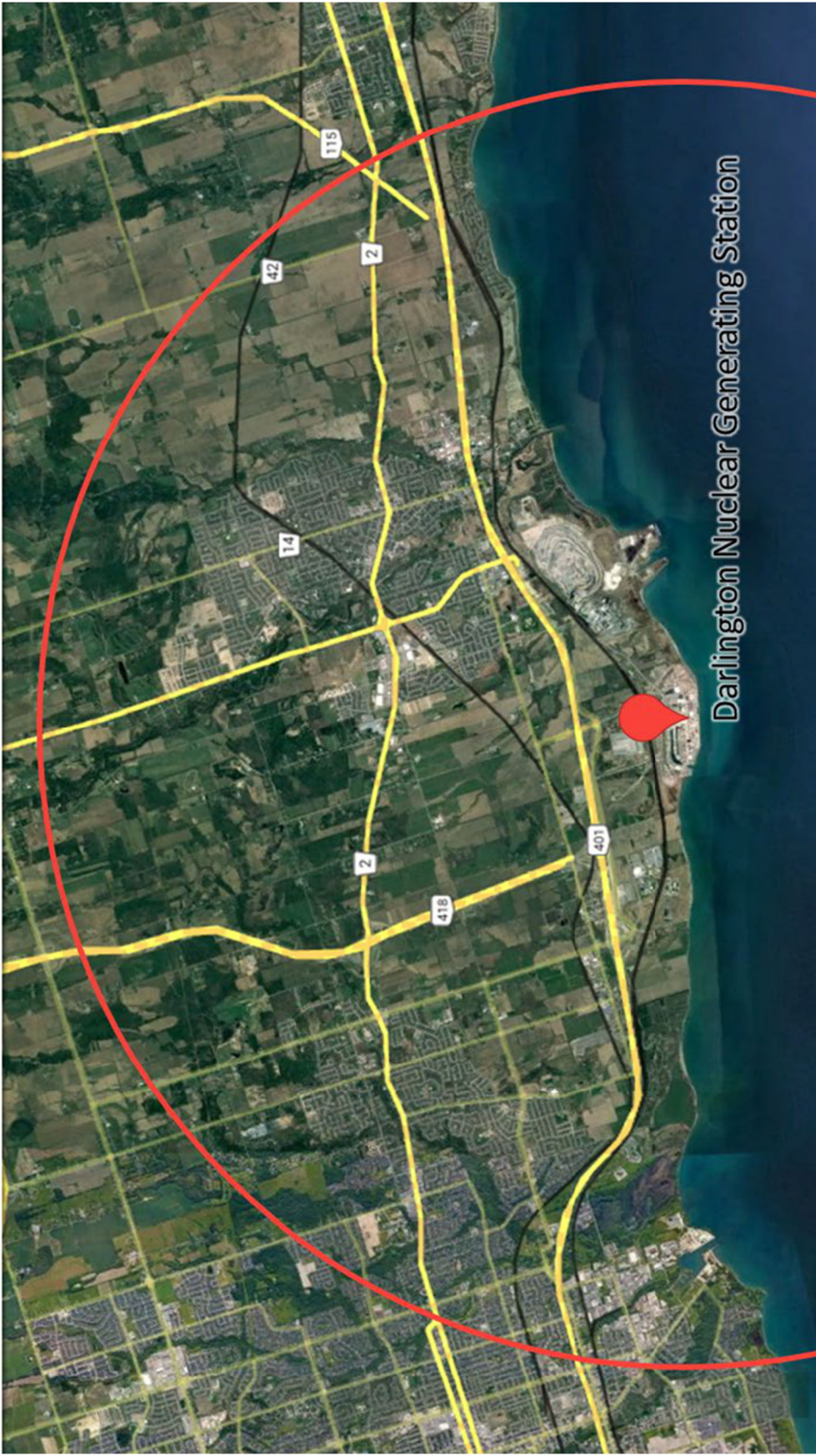


Figure 2.10.2-2: Area of Consideration for Traffic Management Plan

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2.10.6 References

- 2.10-1 NK054-PLAN-01210-00002, "DNNP Nuclear Emergency Preparedness Plan," Ontario Power Generation.
- 2.10-2 N-PROG-RA-0001, "Consolidated Nuclear Emergency Plan," Ontario Power Generation.
- 2.10-3 NK38-REP-03490-10133, "Darlington NGS Development of Evacuation Time Estimates," Ontario Power Generation.
- 2.10-4 NK054-REP-01210-00010, "Summary Report: Site Evaluation Studies for Nuclear Installations at Darlington External Human Induced Events," Ontario Power Generation.
- 2.10-5 NK054-PLAN-08965.4-00001, "Darlington New Nuclear Project Traffic Management Plan (TMP)," Ontario Power Generation.
- 2.10-6 NK054-REP-01210-00008, "Site Evacuation for OPG new Nuclear at Darlington – New Nuclear Safety Considerations," Ontario Power Generation.
- 2.10-7 D-PLAN-00120-0001, "Darlington Nuclear Generating Station Campus Plan," Ontario Power Generation.
- 2.10-8 NK054-REP-07730-00021, "Emergency Planning and Preparedness Technical Support Document: New Nuclear – Darlington Environmental Assessment," Ontario Power Generation.
- 2.10-9 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities; Nuclear Power Plants."
- 2.10-10 D-INS-03490-10030, "Evacuation Relocation," Ontario Power Generation.
- 2.10-11 D-INS-03490-10015, "Security First Line Manager," Ontario Power Generation.
- 2.10-12 Ontario Provincial Nuclear Emergency Response Plan (PNERP) – Master Plan, 2017.
- 2.10-13 CNSC Regulatory Document REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response."
- 2.10-14 N-GUID-03491-10011, "Potassium Iodide (KI) Pill Administration Guide," Ontario Power Generation.
- 2.10-15 N-STD-AS-0010, "Nuclear Crisis Communication Standard," Ontario Power Generation.
- 2.10-16 OPG-PROC-0028, "Crisis Management and Communications Centre Procedure," Ontario Power Generation.
- 2.10-17 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 2.10-18 CSA N293-12, "Fire Protection for Nuclear Power Plants," CSA Group.
- 2.10-19 CSA N293S1:21, "Supplement No. 1 to N293-12, Fire protection for nuclear power plants (application to small modular reactors)," CSA Group.
- 2.10-20 NK054-REP-07730-0969014, "Updated Darlington New Nuclear Project Traffic Management Plan (TMP)," Ontario Power Generation.
- 2.10-21 CNSC Regulatory Document REGDOC-2.3.2, "Operating Performance – Accident Management."

2.11 Monitoring of Site-Related Parameters

Section 2.11 provides a description of the strategy for monitoring site-related parameters relevant to the DNNP site, with emphasis on the site parameters that need to be monitored for the hazards identified in Section 2.2 which affect the DNNP through the lifecycle of the BWRX-300 facility. The information in Section 2.11 satisfies the requirements of Subsection 4.5.2 of REGDOC-1.1.2 (Reference 2.11-15) and the guidance of Subsection 7.4.2 of REGDOC-2.5.2 (Reference 2.11-16).

The information in Section 2.11 covers:

- Volcanic Phenomena Monitoring – Subsection 2.11.1
- Surface Faulting Monitoring – Subsection 2.11.2
- Seismic and Geotechnical Monitoring – Subsection 2.11.3
- Meteorological Monitoring – Subsection 2.11.4
- Hydrological monitoring – Subsection 2.11.5
- Radiation Monitoring – Subsection 2.11.6
- Environmental Monitoring – Subsection 2.11.7
- Biological Organisms and Human Induced Hazards Monitoring – Subsection 2.11.8
- Long Term Monitoring Program – Subsection 2.11.9

Table 2.11-1 summarizes key DNNP characteristics and the approach for monitoring key site parameters.

Table 2.11-1: DNNP Site Characteristics and Parameters Monitoring Approach

Characteristic	Monitoring Approach
2.11.1 Volcanoes Monitoring	Hazard Screened out – No site-specific parameter to be monitored
2.11.2 Surface Faulting Monitoring	Hazard Screened out – No site-specific parameter to be monitored. Any changes will be evaluated within the long-term monitoring program.
2.11.3 Seismic and Geotechnical Monitoring	<ul style="list-style-type: none"> • Southern Ontario Seismic Network stations on Darlington Nuclear site • Current site-specific information is used during construction, with monitoring of excavation and blasting effects. • The Foundation Interface Analysis (FIA) work in (Reference 2.11.19) is fed by the site-specific parameters reported in (Reference 2.11-20) and will be updated by monitored specific geotechnical and seismic parameters during operation. • In-service monitoring approach of and instrumentation for BWRX-300 structures include testing and surveillance programs for below-grade structures and foundations over their design lives <p>Field instrumentation system with recordings is benchmarked against design estimates of settlement and vertical and horizontal movement around the deeply embedded RB and the foundations of the Control Building (CB), TB, and RWB</p>

Table 2.11-1: DNNP Site Characteristics and Parameters Monitoring Approach

Characteristic	Monitoring Approach	
2.11.4 Meteorological Monitoring	<ul style="list-style-type: none"> On-site meteorological tower Environment Canada maintained stations, and notification on severe weather conditions	
2.11.5 Hydrological Monitoring	<ul style="list-style-type: none"> Precipitation, groundwater flow and groundwater hydrology Lake Ontario water levels Lake current real-time monitoring system	
2.11.6 Radiation Monitoring (refer to Section 2.9)	<ul style="list-style-type: none"> Environmental off-site and site boundary monitoring and sampling Off-site and site boundary TLD sites Automated Gamma monitoring system Effluent Monitoring Program	
2.11.7 Environmental Monitoring	Environmental Monitoring Program, detailed in Chapter 20, Subsection 20.11.2	
2.11.8 Biological Organisms and Human Induced Hazards Monitoring	Waterborne, and Airborne Hazards and Biological Organisms	Monitored and controlled in a manner to enable the continued safe operation of the BWRX-300
	Human Induced Hazards—General	Screened out based on Design Mitigation – No Site-specific parameter to be monitored
	Air Transportation activities	Hazard Screened out – No site-specific parameter to be monitored
	Chemical Explosions	Screened out based on Design Mitigation – No Site-specific parameter to be monitored
	Activities at nearby industrial and other facilities	St. Marys Cement plant seismic monitoring station
2.11.9 Long Term Monitoring Program	To be determined potential impacts of climate changes on BWRX-300 operation via long-term monitoring, review, and updates	

2.11.1 Volcanic Phenomena Monitoring

There are no volcanic structures or active volcanoes in the vicinity of the DNNP site. Therefore, the volcanic hazard is not a potential hazard to the DNNP site, and no site-specific parameter to be monitored for this hazard as it is screened out, as per the 2020 DNNP application to renew the Site Preparation Licence NK054-CORR-00531-10533 (Reference 2.11-2).

2.11.2 Surface Faulting Monitoring

There are no active surface faults or tectonic plates in the vicinity of the DNNP site. Therefore, there is no site-specific parameter to be monitored for surface faulting hazard at the DNNP site as this is screened out, as described in the 2020 NK054-CORR-00531-10533 (Reference 2.11-2). Any changes in this hazard are to be evaluated as part of the long-term monitoring program.

2.11.3 Seismic and Geotechnical Monitoring

Site-related parameters are monitored to account for effects from seismic or geotechnical hazards, including earthquakes. Characterization of the seismicity of the region surrounding the

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site, using the Southern Ontario Seismic Network stations on Darlington Nuclear site, forms an essential part of the assessment of the seismic hazard.

Considering the proximity of the DNNP and DNGS sites, the updated hazard curve characterizing the seismic conditions for DNGS in the 2021 Darlington Risk Assessment NK38-REP-03611-10041 (Reference 2.11-1) is deemed applicable to the DNNP site and, thus, is to be utilized during the design and construction stages of the BWRX-300 facility.

The DNNP site-specific geotechnical considerations are discussed in Section 9.3 of the 2009 DNNP Site Evaluation of geotechnical aspects NK054-REP-01210-00011 (Reference 2.11-3). During the construction of the BWRX-300 facility, the effects of any excavation or blasting is to be monitored for their impact on the existing DNGS Power Blocks.

All permanent cut/fill slopes within the areas for DNNP site are to be instrumented and monitored regularly during and after completion of construction and during operation of the BWRX-300 facility (Reference 2.11-3). The information in NEDO-33914-A (Reference 2.11-4) identifies the BWRX-300 advanced civil construction and design approach.

The activities during construction and commissioning are to be monitored to identify the surfaces of civil structures that are exposed to soil, backfill or engineered fill, rock, and groundwater. The monitoring results are evaluated to determine susceptibility of the civil structures surfaces material to deterioration, and the ability to perform the intended design function under the anticipated conditions. An FIA is described in Section 4 of NEDO-33914-A (Reference 2.11-4). The FIA is further advanced specifically for the DNNP BWRX-300 in the 2023 NK054-REP-03500.8-00003 DNNP FIA report (Reference 2.11-19) by running analytical models which employed site-specific parameters that are reported in the 2022 geotechnical investigation and laboratory tests (Reference 2.11-20). The 2023 DNNP FIA report (Reference 2.11-19) analysed the subsurface soil and rock interface with the structures of the Power Block buildings including the deeply embedded RB, and new loads arising during the operational life of the BWRX-300, such as loads from ground motions, pressures, and from potential subsurface deformations that originate from subgrade instabilities and potential liquefaction (Reference 2.11-22). (Additional information on FIA as related to the DNNP and BWRX-300 is provided in Subsection 2.7.3.2, Subsection 2.7.3.3, and Subsection 2.7.5.1).

The in-service monitoring approach, presented in Section 3.3 of NEDO-33914-A (Reference 2.11-4) for the BWRX-300 also covers post-construction testing and in-service surveillance programs for below-grade structural members and foundation. Some of such activities include periodic examination of inaccessible areas, monitoring of groundwater chemistry, and monitoring of settlements and differential displacements. The purpose of the in-service monitoring programs is to monitor the condition of BWRX-300 structures over their design lives to ensure the credited safety functions as well as the overall structural integrity are maintained. The overall integrity of all civil structures, regardless of safety classification, is critical for plant personnel to safely maintain plant facilities during service and through decommissioning.

Additionally, DNNP will have a field instrumentation system related to the BWRX-300 deeply embedded RB. As described in NEDO-33914-A (Reference 2.11-4), field instrumentation that is beyond the current regulatory guidelines, is deployed to monitor the magnitude and distribution of pore pressure and amount of deformation during excavation, construction, loading and continuing through the BWRX-300 plant operation. The instrumentation provides recordings that are frequently benchmarked against design estimates. Short-term and long-term settlement monitoring plans are developed that can detect both vertical and horizontal movements in and around the structures, as well as differential distortion across the foundation footprint and differential settlements between the foundations of the CB, Turbine Building (TB), RWB and RB.

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Free field and in-service seismic instrumentation are further discussed in Chapter 3, Subsection 3.3.1.5 as follows:

- Location and description of instrumentation – Subsection 3.3.1.5.1
- Design and installation – Subsection 3.3.1.5.2
- Maintenance and testing – Subsection 3.3.1.5.3
- Arrangement for control room operator notification – Subsection 3.3.1.5.4
- Comparison of measured and predicted responses – Subsection 3.3.1.5.5

2.11.4 Meteorological Monitoring

With respect to meteorological factors, data such as temperature, wind speed, and wind direction are required for monitoring the direction of dispersion of any potential containment release from the DNNP site to the surrounding environment. The meteorological data are used to calculate DRLs and dose to the public through off-site radiological environmental monitoring. In the event of an accidental release off-site, the meteorological factors provide data to support the CNEP N-PROG-RA-0001 (Reference 2.11-9).

The meteorological tower at the Darlington Nuclear site described in the 2009 NK054-REP-01210-00013 (Reference 2.11-5) is located just north of the site, just southeast of the intersection of Highway 401 and Holt Road (main access to the site). The tower has no significant obstructions from nearby buildings. Meteorological data available from the site consist of wind speed and direction at two heights (10 m and 50 m) and temperature at one height (10 m). Humidity, air pressure, and precipitation are currently not logged on-site by the meteorological tower. However, the information is readily available from Environment Canada stations as listed in Section 2.2.1 of the 2012 NK054-REP-01210-00016 (Reference 2.11-6). The data collected from the Darlington Nuclear site, per NK054-REP-01210-00013 (Reference 2.11-5) are used and adapted for to the DNNP site characteristics and the BWRX-300 design. The development of a DNNP on-site meteorological program progresses, tracked by CNSC commitment D-C-8, Meteorological Monitoring Station.

Additionally, notifications from Environment Canada for existing OPG facilities are received on severe weather which allow OPG to enter the severe weather emergency preparedness procedure N-PROC-RA-0095 (Reference 2.11-18).

2.11.5 Hydrological Monitoring

The assessment of the potential flood hazards at DNNP is described in the 2022 NK054-REP-02730-00001, Flood Hazard Assessment (Reference 2.1-21)

The BWRX-300 does include precipitation as a site-related parameter for monitoring and is assessed against the flooding hazard as part of the safety analysis as the detailed design progresses, as described in the 2020 Application to renew DNNP Site Preparation Licence NK054-CORR-00531-10533 (Reference 2.11-2). As noted in Subsection 2.11.4, precipitation is monitored through local Environment Canada weather stations.

Groundwater flow and groundwater hydrology were assessed as a part of the 2020 NK054-CORR-00531-10533 (Reference 2.11-2), and conditions monitoring with respect to hydrology, boreholes and wells were fitted with equipment for sampling and level monitoring purposes. Sections 3.5 and 3.6 of Volume 2 of the 2022 DNNP Geotechnical Investigation (Power Block) NK054-REP-01210-00175 (Reference 2.11-21) updated the information and database on groundwater flow and hydrostratigraphic units. Annual groundwater monitoring has occurred across the DNNP site study area since the original 2009 Site Evaluation NK054-REP-01210-

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00011 (Reference 2.11.3). Additional information is presented on groundwater conditions, flow, and hydro-stratigraphy in Subsection 2.5.5 and Subsection 2.7.3.2.4. Further information on the groundwater monitoring program is provided in Chapter 20, Subsection 20.11.4.

Levels in Lake Ontario are monitored by various organizations, including the Canadian Hydrographic Service, National Oceanic and Atmospheric Administration and Environment Canada as described in Section 8.2 of the 2009 NK054-REP-01210-00012 (Reference 2.11-13). The water level of Lake Ontario is controlled by the International Joint Committee— a joint group between Canada and the USA. Additional information is presented in Subsection 2.5.2.1 on how Lake Ontario water level is monitored and regulated.

The current in Lake Ontario is also monitored using the Lake Current Monitoring System as described in the 2019 NK38-OM-61100 (Reference 2.11-10) which resides in the lake approximately 1.6 km offshore of the Bowmanville Water Supply Plant, east of Darlington Nuclear site. The Lake Current Monitoring System real-time current profile measurement system is used in the event of a radiological liquid emission from Operations that takes place on the DNNP site. The Lake Current Monitoring System consists of an Acoustic Doppler Current Profiler and a Remote System Manager base station. The data acquired from Lake Current Monitoring System is also applicable to the DNNP given it is part of the Darlington Nuclear site.

2.11.6 Radiation Monitoring

Radiation Monitoring is comprised of on-site, site boundary, and off-site monitoring systems and programs. Information on radiation monitoring is available in the following subsections:

1. The environmental off-site and site boundary monitoring systems and sampling programs (Environmental Monitoring Program) – Subsection 2.9.2.1
2. The TLDs that are located around the Darlington Nuclear site perimeter as well as at off-site locations – Subsection 2.9.2.2
3. The Automated Near Boundary Gamma Monitoring System, located around the Darlington Nuclear site boundary – Subsection 2.9.2.3
4. Site Effluent Monitoring Program – Subsection 2.9.2.4

2.11.7 Environmental Monitoring

The Darlington Nuclear Environmental Monitoring Program identifies the contaminants and physical stressors to be monitored and conducts monitoring in the environment surrounding the site. The Environmental Monitoring Program is discussed in detail in Chapter 20, Subsection 20.11.2.

2.11.8 Biological Organisms and Human Induced Hazards Monitoring

2.11.8.1 Biological Organisms

Biological hazards specific to the DNNP site are similar to those of the 2019 DNGS NK38-REP-03611-10043 (Reference 2.11-7), given the two sites proximity.

Examples of such hazards are waterborne (e.g., fish, algae, zebra-mussel, or biofouling), large animals (e.g., herds of deer) or flying birds/insects (e.g., flocks of geese). These biological hazards are monitored and controlled in a manner enabling the safe operation of the plant.

Biofouling control typically involves appropriate biomonitoring and application of appropriate biocides/antimicrobials specific to the circuits and sensitivity of the system components. The control of the biofilms is a standard operational procedure at facilities supplied by water from Lake Ontario, and accordingly this form of biofouling is manageable for the BWRX-300 using available

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technology, as described in the 2009 DNNP Site Evaluation on nuclear safety considerations NK054-REP-01210-00008 (Reference 2.11-8).

Additional information on the impact of biological and animal hazards on the safe operation of BWRX-300 facility is provided in Subsection 2.2.7.1, and on potential biofouling hazard and its impact on cooling lake water supply is presented in Subsection 2.5.2.2.

2.11.8.2 Human Induced Hazards

With respect to non-malevolent human induced hazards, all events were screened out, per the 2019 Hazards Screening Analysis NK38-REP-03611-10043 R003 (Reference 2.11-7) from the need to perform a PSA. As discussed in the following subsections, human induced hazards are screened out qualitatively or quantitatively based on the design and robustness of the BWRX-300 facility. No specific parameters are to be monitored for external human induced hazards.

2.11.8.2.1 Air Transportation Activities

As discussed in Subsection 2.2.3.1, hazards from air transportation accidents are screened out. No site-specific parameter is expected to be monitored for aircraft/flight impacts for the DNNP site. Refer to Subsection 2.2.3.1 for additional information.

2.11.8.2.2 Chemical Explosions

The DNNP site has various shipping lanes, which carry bulk marine shipments and the Canadian National Railway which runs within the exclusion zone of the site. The probability of accidents posing significant threat to the site is low, per the 2019 NK38-REP-03611-10043 (Reference 2.11-7). Transport vehicles carrying toxic and hazardous materials (mainly gaseous) pose a threat to worker safety which is recognized in the Site Evaluation. No site-specific parameter is expected to be monitored for chemical explosions for impacts on the DNNP site. For additional information on hazards resulting from transportation accidents refer to Subsections 2.2.3.2, 2.2.3.3, and 2.2.3.4, and from stationary non-nuclear accident refer to Subsection 2.2.4.

2.11.8.2.3 Activities at Nearby Industrial and Other Facilities

The St. Marys Cement plant is located on the east side of DNNP site, about 700 meters from the proposed BWRX-300 location. This cement plant performs blasting at the quarry that leads to shock waves in the ground that could travel up to the BWRX-300 structures. Such shock waves are monitored using vibration monitors at a seismic monitoring station on the St. Marys property boundary. The St. Marys Cement plant is also committed to comply with the agreement established with OPG, which states that the cement plant should not carry out blasts that may exceed the maximum allowable horizontal, vertical, longitudinal, and radial velocities of 3 mm/s, per the 2019 NK38-REP-03611-10043 (Reference 2.11-7). As part of the DNGS seismic hazard curve provided in the 2021 NK38-REP-03611-10041 (Reference 2.11-1) to be used also for the DNNP site, underground shock wave effects are to be addressed through the PSA. Refer to Subsection 2.2.6 for additional information.

2.11.9 Long Term Monitoring Program

The work conducted in the 2023 report on Climate Change Impact NK054-PLAN-07007-00001 (Reference 2.11-20) confirmed the low impact of climate change stipulated in Subsection 2.5.4. Such work included climate modelling and reviewed published articles to evaluate the anticipated impact of climate change on the DNNP site and surrounding area.

Long term monitoring (periodic review/update) of applicable site-specific hazards is an inherent feature of the PSA process. As per REGDOC-2.4.2 (Reference 2.11-14), the PSA models for nuclear stations are updated every 5 years, or sooner if the facility undergoes major changes and are managed by the 2021 Preparation, Maintenance and Application of Probabilistic Safety

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Assessment N-STD-RA-033 (Reference 2.11-11). As part of this process, site-related parameters that feed into the hazard screening are revisited for new modelling methods or for any changes in the site parameters. The screening criteria for the PSA are updated every 5 years as per the 2018 OPG's Probabilistic Safety Assessment Guide N-GUID-03611-10001 (Reference 2.11-12). For cases in which data are regularly monitored at the site (e.g., wind speed or other meteorological data), and cases for which data are collected from external sources (e.g., air traffic in the vicinity of the site), the new data are assessed as part of the hazard screening for the DNGS site. A similar long-term approach is applied for the DNNP site to assess all site-related parameters for any changes.

Long term monitoring of climate change data is to be performed in accordance with REGDOC-1.1.1 (Reference 2.11-17) which requires the Site Evaluation and Site Characterization be revisited at each licensing phase to confirm it remains valid with changing environmental conditions. REGDOC-1.1.2 (Reference 2.11-15) reinforces this requirement for the Licence to Construct application and requires site characteristics be confirmed for the construction phase. REGDOC-2.5.2 (Reference 2.11-16) also requires the design of a nuclear power plant to consider all site characteristics that may affect the safety of the plant and monitoring of site-related parameters be in place throughout the lifecycle of the plant. Hazards that are applicable to the DNNP site and affected by climate change are to be monitored. Parameters associated with these climate change hazards (e.g., meteorological, lake temperature) are to be obtained from a variety of sources, including but not limited to, site-located instrumentation and local weather data. The frequency at which a climate change hazard is to be measured and analysed will depend on the nature of the hazard and its impact on the DNNP facility (e.g., nuclear safety impact, commercial impact). Climate change hazards will undergo risk assessment and where suitable will be subject to risk treatment (e.g., adaptive action or a risk monitoring plan). Where a risk monitoring plan is in place the trigger point for an adaptive action will be specified with consideration for the duration required to implement the action. The 2023 NK054-PLAN-07007-00001 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts (Reference 2.11-20) provides additional information on lifecycle considerations including long term monitoring.

2.11.10 References

- 2.11-1 NK38-REP-03611-10041 R003, 2021, "Update of the OPG Darlington Site Probabilistic Seismic Hazard Assessment for the Darlington Risk Assessment (DARA)," Ontario Power Generation.
- 2.11-2 NK054-CORR-00531-10533, 2020, "Application for Renewal of OP's Darlington New Nuclear Project (DNNP) Nuclear Power Reactor Site Preparation License (PRSL)," Ontario Power Generation.
- 2.11-3 NK054-REP-01210-00011 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington— Part 6: Evaluation of Geotechnical Aspects," Ontario Power Generation.
- 2.11-4 NEDO-33914-A, Revision 2, 2022, "BWRX-300 Advanced Civil Construction and Design Approach" GE-Hitachi Nuclear Energy Americas, LLC.
- 2.11-5 NK054-REP-01210-00013 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington— Part 4: Evaluation of Meteorological Events," Ontario Power Generation.
- 2.11-6 NK054-REP-01210-00016 R002, 2012. "Site Evaluation of the OPG New Nuclear at Darlington— Part 2: Dispersion of Radioactive Materials in Air and Water," Ontario Power Generation.

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- 2.11-7 NK38-REP-03611-10043 R003, 2019, "Hazards Screening Analysis – Darlington," Ontario Power Generation.
- 2.11-8 NK054-REP-01210-00008 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington – Nuclear Safety Considerations," Ontario Power Generation.
- 2.11-9 N-PROG-RA-0001 R019, "Consolidated Nuclear Emergency Plan," Ontario Power Generation.
- 2.11-10 NK38-OM-61100 R013, 2019, "Environmental Monitoring – Air and Water," Ontario Power Generation.
- 2.11-11 N-STD-RA-033 R006, 2021, "Preparation, Maintenance and Application of Probabilistic Safety Assessment," Ontario Power Generation.
- 2.11-12 N-GUID-03611-10001 Volume 8, 2018, "OPG Probabilistic Safety Assessment (PSA) Guide – External Hazard Screening," Ontario Power Generation.
- 2.11-13 NK054-REP-01210-00012 R001, 2009, "Site Evaluation of the OPG New Nuclear at Darlington— Part 5: Flood Hazard Assessment," Ontario Power Generation.
- 2.11-14 CNSC Regulatory Document REGDOC-2.4.2, "Safety Analysis Probabilistic Safety Assessment (PSA) for Reactor Facilities."
- 2.11-15 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 2.11-16 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 2.11-17 CNSC Regulatory Document REGDOC-1.1.1, "Licence Application Guide: Site Evaluation and Site Preparation for New Reactor Facilities."
- 2.11-18 N-PROC-RA-0095, "Severe Weather Emergency Preparedness," Ontario Power Generation.
- 2.11-19 NK054-REP-03500.8-00003, 2023, "Darlington New Nuclear Project Foundation Interface Analysis (FIA) Report," Ontario Power Generation
- 2.11-20 NK054-PLAN-07007-00001, 2023, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts," Ontario Power Generation
- 2.11-21 NK054-REP-01210-00175 R000, 2022, "Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project", Volume 2 of 2 "Geotechnical Interpretation of Design Parameters," Ontario Power Generation.
- 2.11-22 NK054-REP-03500.8-00002 R000, 2022, "Darlington New Nuclear Project— Seismically-Induced Soil Liquefaction Assessment," Ontario Power Generation

2.12 Ongoing Work Plans

2.12.1 Introduction

Section 2.12 details information on plans to complete ongoing DNNP specific works involving geotechnical investigations, laboratory tests, analyses, and assessments to validate and update existing DNNP parameters or generate new site-specific characterizations and parameters to supplement and update existing database. Each disposition plan provides:

- Background information on the ongoing work
- The schedule and workflow by which the ongoing work is to be completed
- Risks associated with the ongoing work
- Chapter 2 sections impacted by the ongoing work
- Progress of work, including deliverables

Details of each work is provided as follows:

Subsection 2.12.2 – Foundation Interface Analysis (FIA)

Subsection 2.12.3— Site Geotechnical and Seismic Hazard Investigation Plan, which includes

- Geotechnical investigations (Power Block) and laboratory tests
- Offshore geotechnical investigation
- Site-specific Probabilistic Seismic Hazard Assessment (PSHA)
- Seismically-induced liquefaction assessment

Subsection 2.12.4— Flood Hazard Assessment

Subsection 2.12.5 – Climate Change Impact

Subsection 2.12.6 – 3-second Wind Gust Validation

Subsection 2.12.7- Winter PMP Validation

Subsection 2.12.8 – PMP Validation

The results of each completed work are incorporated into the impacted sections in Chapter 2. A summary description of each work along with the deliverables are provided in Table 2.12-1.

Table 2.12-1: DNNP Projects Closure Plans and Associated Updates

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.2	Foundation Interface Analysis (FIA) <u>Status:</u> Complete	<p>The FIA results will support the evaluation of the construction plan, the stability of the excavation, ground improvements and the design of excavation support systems. Also, the results of ground pressure demands on the below-grade exterior walls of the RB will be used to validate ground pressure design loads. The FIA will be performed with three dimensional models representing the site conditions at all project stages, including design, construction, and operation.</p> <p>Specific tasks are as follows:</p> <ul style="list-style-type: none"> • Evaluation of the subgrade materials and the materials surrounding the deeply embedded BRWX-300 RB • Confirmation that the Radwaste Building, Turbine Building, and CB foundations are to be supported by the engineered fill, intermediate glaciolacustrine, and lower till soils • Confirmation of the stability of sand and rock excavation for the stability of the deeply embedded RB shaft evaluation for excavation and construction <p>The resulting report will discuss:</p> <ol style="list-style-type: none"> 1. Effects of excavation, dewatering (based on hydrogeology report) and construction on subgrade material properties 2. Evaluations of potential for unstable rock mass or unstable blocks and wedges including the joints and sizes of the potential blocks or wedges 3. Results of the FIA of the site characterization, excavation, construction, loading, operation stages 4. Inputs and results of sensitivity FIA or additional stability analysis <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> 1. NK054-REP-03500.8-00003, 2023, "Darlington New Nuclear Project Foundation Interaction Analysis (FIA) Report," Ontario Power Generation

Table 2.12-1: DNNP Projects Closure Plans and Associated Updates

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.3	<p>Site Geotechnical and Seismic Hazard Investigation Plan</p> <p><u>Status:</u> Complete</p>	<p>The main deliverables of OPG's Site Geotechnical and Seismic Hazard Investigation are as follows:</p> <ol style="list-style-type: none"> 1. Perform Geophysical Survey and Mapping of Subsurface Strata 2. Detailed Site Investigation and Geotechnical Lab Tests 3. Excavation and Stockpile / Earth Removal 4. Geological Hazard Scenarios 5. Liquefaction Potential Assessment 6. DNNP Probabilistic Seismic Hazard Analysis 7. DNNP Specific Seismic Hazard <p>The results of this work will be used for the confirmation of BWRX-300 bounding parameters</p> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> 1. (NK054-REP-01210-00175) Golder Associates Ltd. (Golder), 2022, Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project, Revision 2, Volumes 1 and 2, July 29 2. (NK054-REP-10180-00001) Golder Associates Ltd. (Golder), 2023, Offshore Geotechnical Investigation Darlington New Nuclear Project, Revision 0. 3. (NK054-REP-03500.8-00001) Kinectrics Inc., K-620423/RP/0001 R01, "Darlington New Nuclear Project-- Site-Specific Probabilistic Seismic Hazard Assessment," 2022 4. (NK054-REP-03500.8-00002) Kinectrics Inc., K-620423/RP/0002 R00, "Darlington New Nuclear Project-- Seismically-Induced Soil Liquefaction Assessment," 2022

Table 2.12-1: DNNP Projects Closure Plans and Associated Updates

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.4	Flood Hazard Assessment <u>Status:</u> Complete	<p>This Hydrological Analysis is expected to follow a similar format to the original flood assessment covering:</p> <ul style="list-style-type: none"> • Identification of Flooding Hazards • Description of DNNP Site Layout • Assessment of Flooding Hazards • Flood Protection • Modification of the Flood Hazard over time • Monitoring and Warning for Plant Protection • Conclusions and Recommendations <p>The results of this work are used for the confirmation of BWRX-300 bounding parameters</p> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> 1. NK054-REP-02730-00001 R000, 2022, "Flood Hazard Assessment", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-001 R01).

Table 2.12-1: DNNP Projects Closure Plans and Associated Updates

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.5	Climate Change Impact <u>Status:</u> “CNCS Deliverable 1: DNNP Strategy for Addressing Climate Change Impacts.” Complete	<p>Conditions from climate change which impact flooding have been incorporated into Chapter 2 based on the 2022 NK054-REP-02730-00001 “Flood Hazard Assessment” (Reference 2.12-4).</p> <p>OPG has issued the 2023 NK054-PLAN-07007-00001 R000 “Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts” (Reference 2.12-5). This strategy has two primary phases: Climate Change Risk Assessment and Climate Change Risk Treatment. Following work will be performed on an as-required basis to integrate climate change assessments into the current nuclear safety framework. This will include lifecycle considerations such as long-term monitoring and periodic reassessment of hazards associated with climate change DNNP commitment D-C-7 in accordance with the strategy outlined in NK054-PLAN-07007-00001 (Reference 2.12-5). D-C-7 will be completed prior to start of construction as per NK054-REP-01210-00078 (Reference 2.12-2).</p> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> 1. (NK054-PLAN-07007-00001 R000), 2023, “Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts”, Ontario Power Generation 2. NK054-REP-07007-1049426 R001, 2023, “Darlington New Nuclear Project – Hazard Bounding Analysis,” Ontario Power Generation 3. NK054-REP-07007-1028871 R000, 2022, “Darlington New Nuclear Project— Gradual Climate Change and Natural Hazard Identification,” Ontario Power Generation
2.12.6	3-second Wind Gust Calculation <u>Status:</u> Complete	<p>While maximum wind speed is an instantaneous wind speed, the 3-second gust value is a sustained wind speed. Maximum wind speed is shown in Subsection 2.6.5.</p> <p>Key two aspects of this work are:</p> <ul style="list-style-type: none"> • Calculation of the site characteristic for 3-second wind gust speed is in progress • Value will confirm BWRX-300 bounding approach <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> 1. NK054-REP-02730-00003 R000, 2022, “Wind Gust Analysis”, Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-003 R01)

Table 2.12-1: DNNP Projects Closure Plans and Associated Updates

Sub-section	Disposition Plan/Status	Description/Deliverables
2.12.7	Winter PMP Validation <u>Status:</u> Complete	<p>Work started to finalize appropriate consideration for snow load with a Winter Probable Maximum Precipitation (PMP) event. DNNP considers this a review level condition.</p> <p>Finalization of the coincident snow load and winter PMP is complete.</p> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> 1. NK054-REP-02730-00004 R000, 2022, "Winter PMP Validation", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-004 R01)
2.12.8	PMP Validation <u>Status:</u> Complete	<p>Confirmation of rainfall and PMP</p> <p><u>Deliverables:</u></p> <ol style="list-style-type: none"> 1. NK054-REP-02730-00002 R000, 2022, "PMP Validation", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-002 R01)

2.12.2 Foundation Interface Analysis

2.12.2.1 Background

OPG has undertaken a site-specific, non-linear FIA, to ensure the stability of structures, supporting media, soil, and rock per NUREG-800 SRP 2.5.4 guidance. The FIA results support the evaluation of the construction plan, the stability of the excavation, ground improvements and the design of excavation support systems. Also, the results of ground pressure demands on the below-grade exterior walls of the RB are used to validate ground pressure design loads. The FIA is performed with three dimensional models representing the site conditions at all project stages, including design, construction, and operation.

The schematic workplan for the FIA modelling is shown in Figure 2.12.2-1.

All relevant available reports describing ground conditions and structural details are reviewed including but not limited to: Geotechnical Investigation Factual and Interpretation Reports, NEDO 33914 Licensing Topical Report [1], and relevant nuclear standards/guidelines. The factual data are summarized and classified for each geological unit and the input parameters required for FIA numerical modelling are calculated or extracted from the laboratory and in-situ test results. The structural information such as the shoring design, construction staging, and the structure details are reviewed and summarized in our FIA interaction modelling activity.

All relevant available reports describing ground conditions and structural details are reviewed, including but not limited to:

- Geotechnical Investigation Factual and Interpretation Reports, NEDO 33914 Licensing Topical Report (Reference 2.12-1)
- Relevant nuclear standards/guidelines.

This information is used to develop the Finite Element Analysis method and 3D framework in Plaxis 3D, allowing full FIA interaction modelling.

The Technical Report is prepared based on the FIA modelling, includes the results of the FIA of the deeply embedded BWRX-300 RB and the surrounding Power Block foundations at the DNNP site. The report discusses:

1. Effects of excavation, dewatering (based on hydrogeology report) and construction on subgrade material properties
2. Evaluations of potential for unstable rock mass or unstable blocks and wedges including the joints and sizes of the potential blocks or wedges
3. Results of the FIA of the site characterization, excavation, construction, loading, operation stages
4. Inputs and results of sensitivity FIA or additional stability analysis

2.12.2.2 Project Schedule and Logic

The report concludes the results of the FIA for the deeply embedded BWRX-300 RB and the surrounding Power Block foundations at the DNNP site. The schematic workplan for the FIA modelling is shown in Figure 2.12.2-1.

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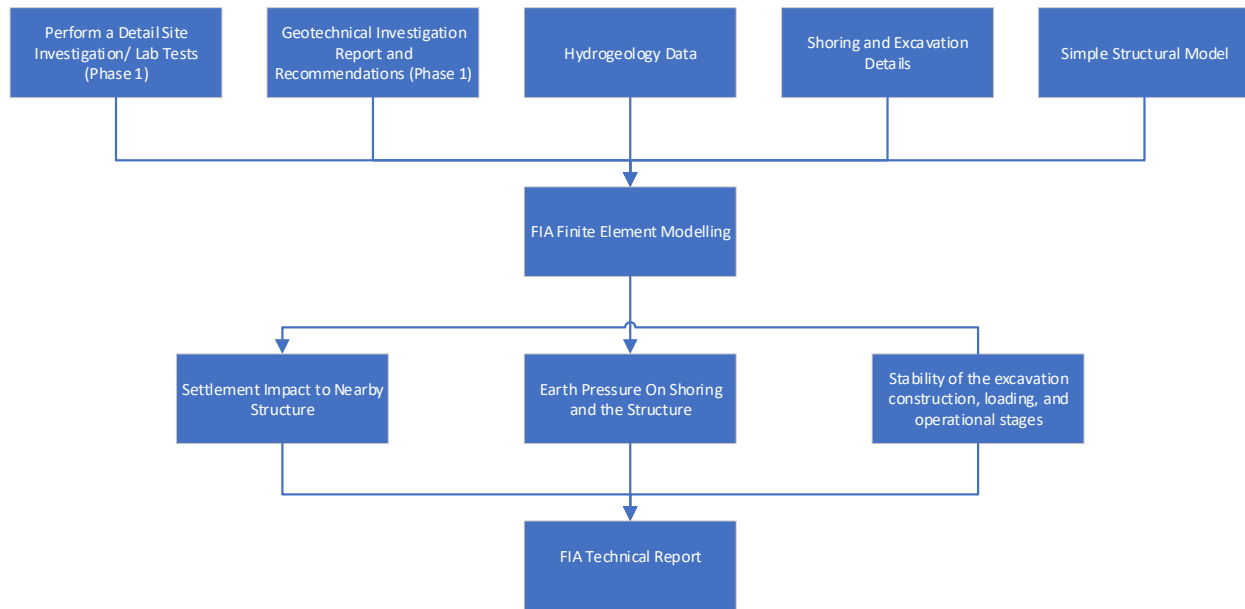


Figure 2.12.2-1: FIA Modelling Workflow and Deliverables

2.12.2.3 Risks

Project timeline is dependent on DNNP confirmatory geotechnical investigation results (Laboratory Test Results and In-Site Test Results) (refer to Subsection 2.12.3). Any delays to the geotechnical investigation may cause a delay to the FIA final deliverable (Technical Report)

2.12.2.4 Impacted Chapter 2 Sections

Section 2.7 – Geology, Seismology, and Geotechnical Engineering.

2.12.2.5 Progress of Work

1. Review completed of recent reports by Golder Associates Ltd. (refer to Subsection 2.12.3) that includes site-specific results of geotechnical investigations and laboratory tests
2. Information received on shoring and excavation details from AECON
3. A simple structural model is tested and verified
4. FIA Finite Element modelling is developed
5. Technical memoranda developed, circulated for review and comments, on the following topics:
 - a. Bearing Capacity Evaluations of the BWRX-300 RB and the Surrounding Power Block Foundations at the DNNP Site
 - b. Settlement Evaluations of the BWRX-300 RB and the Surrounding Power Block Foundations at the DNNP Site
 - c. Excavation and Construction Stages of the BWRX-300 RB Shaft
 - d. FIA Numerical Modelling
6. Additional key parameters are sought and confirmed for use as input to the FIA model

7. Final report is complete

Work is complete and closed. The results are incorporated in Section 2.7.

Deliverables:

The following report was submitted by the outsourced contactor, and was reviewed and accepted by to OPG:

1. NK054-REP-03500.8-00003, 2023, "Darlington New Nuclear Project Foundation Interface Analysis (FIA) Report," Ontario Power Generation

2.12.3 Site Geotechnical and Seismic Hazard Investigation Plan

2.12.3.1 Background

Geotechnical Program

OPG has undertaken a detailed site geotechnical program which provides information on the soil physical, mechanical, and dynamic properties of overburden and rock material. The program assesses whether there are karstic features in the local bedrock at the site. The program is linked to the existing CNSC commitment D-P-9 Site Geotechnical and Seismic Hazard Investigation (Reference 2.12-2). The schematic workplan for OPG's Geotechnical Program is shown in Figure 2.12.3-1.

The geotechnical and seismic hazard investigation program, undertaken by OPG, has primary goals to gather sufficient geological data for the proposed DNNP site, identify potential geotechnical and seismic related hazards, and perform the necessary safety evaluations, analyses, and assessments. Investigation methods used included compilation, review and evaluation of existing/historical documents, detailed geophysical and geotechnical site exploration, and extensive in-situ and laboratory testing. Each of these methods are applicable to all stages of the Site Evaluation process, but to varying extents. The main deliverables of OPG's Site Geotechnical and Seismic Hazard Investigation are as follows:

- Perform Geophysical Survey and Mapping of Subsurface Strata
- Detailed Site Investigation and Geotechnical Lab Tests
- Excavation and Stockpile / Earth Removal
- Geological Hazard Scenarios
- Liquefaction Potential Assessment
- DNNP Probabilistic Seismic Hazard Analysis
- DNNP Specific Seismic Hazard

The results of the OPG's Geotechnical and Seismic Hazard Investigation feed into Section 2.7 Geology, Seismology, and Geotechnical Engineering.

2.12.3.2 Project Schedule and Logic

OPG's Geotechnical Program for Phase 1 is demonstrated in the Project Logic of Figure 2.12.3-1. DNNP's Geotechnical and Seismic Investigations are linked to the existing DNNP CNSC commitment D-P-9 Site Geotechnical and Seismic Hazards Investigations (Reference 13-3).

2.12.3.3 Risks

Delays in completing this program may impact completing OPG work on FIA discussed in Subsection 2.12.2.

2.12.3.4 Impacted Chapter 2 Sections

Subsection 2.7.3 Geotechnical Characteristics

2.12.3.5 Progress of Work

1. Completed geophysical investigation and mapping of subsurface strata
2. Completed detailed site investigation and laboratory tests
3. Drafted report on the geophysical investigation and laboratory tests as well as recommendations
4. Excavation and earth removal studies continue
5. Site-specific characteristics and site response analysis is progressing
6. DNNP PSHA is progressing
7. Liquefaction potential is being assessed and is progressing

Work is complete and closed. The results are incorporated in Section 2.7.

Deliverables

The reports were submitted by the outsourced contactor, and were reviewed and accepted by OPG:

1. NK054-REP-01210-00175 R01, (Golder 2022) "Phase I Geotechnical Investigation (Power Block) Darlington New Nuclear Project," Volumes 1 and 2, Ontario Power Generation.
2. (NK054-REP-10180-00001) Golder Associates Ltd. (Golder), 2023, Offshore Geotechnical Investigation Darlington New Nuclear Project, Revision 0.
3. NK054-REP-03500.8-00001 R00, 2022, Kinectrics Inc., K-620423/RP/0001 R01, "Darlington New Nuclear Project— Site-Specific Probabilistic Seismic Hazard Assessment," Ontario Power Generation.
4. (NK054-REP-03500.8-00002) Kinectrics Inc., K-620423/RP/0002 R00, "Darlington New Nuclear Project— Seismically-Induced Soil Liquefaction Assessment," 2022

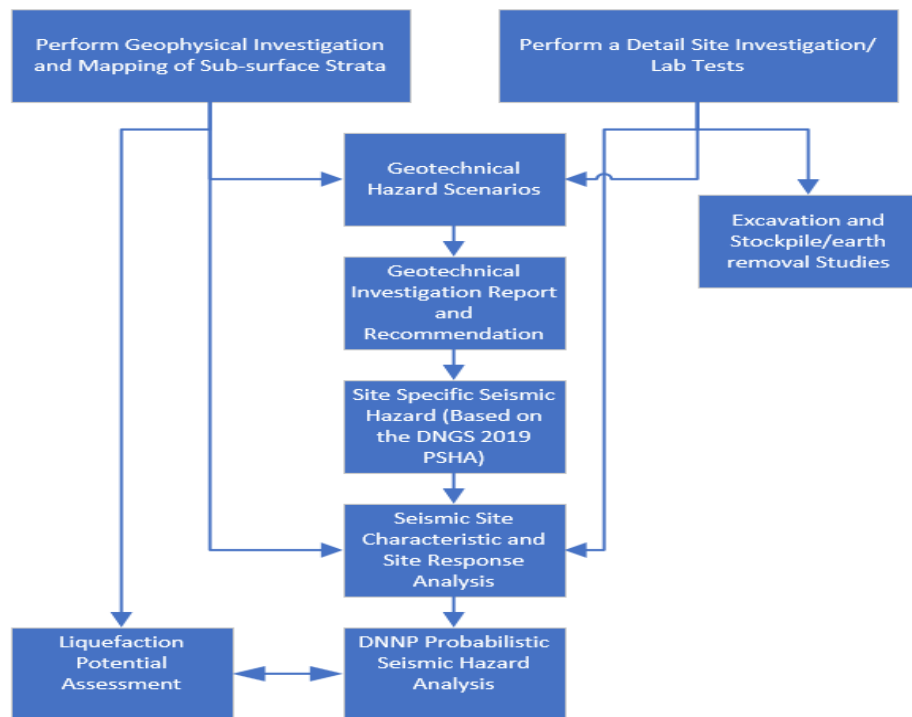


Figure 2.12.3-1: Workflow for the Geotechnical Program

2.12.4 Flood Hazard Assessment

2.12.4.1 Background

A Flood Hazard Assessment is required for Section 2.5 Hydrology.

A previous DNNP Flood Hazard Assessment was completed (Reference 2.12-4) as part of the original Site Evaluation in 2009 included in the EIS and Licence to Prepare Site process, which reflects a site build for up to 4800 mWe of either an EPR, AP-1000, ACR or EC-6 reactor type.

The construction of a 300 mWe BWRX-300 Small Modular Reactor at the DNNP site, led to different site layout, plant grade, and topography to that previously evaluated in Reference 2.12.4. This requires an update to the Flood Hazard Assessment.

OPG has contracted an outsource to complete the Hydrological Analysis which followed a similar format to the original flood assessment covering:

- Review of existing work and data
- Completion of a gap analysis to determine if additional modelling and analysis is required
- Completion of required modelling and analysis
- Organization of information, identification of flood hazards and mitigations, meeting the requirements outlined in REG-DOC1.1.1 and IAEA Nos. NS-R-3, SSG-18, and other regulatory documents

- Identification of Flooding Hazards
- Description of DNNP Site Layout
- Assessment of Flooding Hazards
- Flood Protection
- Modification of the Flood Hazard over time
- Monitoring and Warning for Plant Protection
- Conclusions and Recommendations

2.12.4.2 Project Schedule and Logic

The following deliverables close this ongoing work:

- Draft Flood Hazard Assessment report
- Final Flood Hazard Assessment report

2.12.4.3 Risks

None.

2.12.4.4 Impacted Chapter 2 Sections

Section 2.5 Hydrology.

2.12.4.5 Progress of Work

1. Work is completed and a final report is delivered and accepted by OPG
2. OPG issued, in December 2022, the report as NK054-REP-02730-00001, "Flood Hazard Assessment," Ontario Power Generation.
3. The report has the following contents
 1. Introduction
 2. General Site Description and Characteristics
 3. Existing Site Conditions – Potential Flood Hazards
 4. Post-Development Site Layout
 5. Assessment of Flood Hazards
 6. Mitigation Measures
 7. Modification of the Flood Hazard with Time
 8. Monitoring and Warning for Plant Protection
 9. Conclusions and Recommendations
 10. References

Work is complete and the results are incorporated in impacted sections of Chapter 2

Deliverables:

1. NK054-REP-02730-00001 R000, 2022, "Flood Hazard Assessment", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-001 R01).

2.12.5 Climate Change Impact

2.12.5.1 Background

The potential effects of climate change on external natural hazards such as flooding and temperature as well as life cycle considerations including long-term monitoring programs (refer to Subsection 2.5.4, Subsection 2.6.4, Subsection 2.6.12, and Subsection 2.11.9) are linked to the existing commitment D-C-7, Contingency Plan for Flooding and Other Extreme Weather Hazards (Reference 2.12-2). To address this commitment, OPG has developed NK054-PLAN-07007-00001 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts (Reference 2.12-5), which describes the plan for fulfilling the requirements of commitment D-C-7, and consequently ensuring the DNNP facility is resilient to climate change hazards. Additional information on long term monitoring of climate change hazards is provided in Section 2.11.9.

The DNNP Strategy for addressing Climate Change Impact consists of the following three phases:

1. Phase 1 – Climate Change Risk Assessment

The purpose of this phase is to perform a climate change risk assessment for the DNNP facility to identify climate change hazards, bounding values/ranges, and vulnerable structures, systems, and components. There are two main activities in this phase, the Hazards Identification and Bounding Analysis. Hazard Identification will identify climate change related hazards that can affect DNNP site (e.g., hydrological, meteorological, etc.). Bounding Analysis report will then determine bounding values/ranges for the hazards that pose nuclear safety, commercial, or operational impacts. The values from the bounding analysis will feed into the Plant Envelope Assessment to determine which systems may be vulnerable to climate change hazards.

2. Phase 2 – Climate Change Risk Treatment

The phase analyses the design margins of vulnerable structures, systems and components and develops risk treatments as required. These risk treatments can include adaptation of the design or implementation of risk monitoring plans. The completion of Phase 2 will provide the necessary information that will comply with addressing the effects of climate change on-site.

3. Phase 3 – As Required Work

Work will be performed on an as required basis to integrate climate change assessments into the current nuclear safety framework.

The results of this work are used to confirm low impact of climate change. Where structures, systems, and components are potentially vulnerable to climate change hazards, appropriate risk treatments are developed to ensure climate change resilience is implemented within the design.

To ensure alignment with the regulator, OPG will submit three deliverables to the CNSC. The first being the 2023 NK054-PLAN-07007-00001 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts (Reference 2.12-5) which provides the CNSC a description of the proposed methodology for the close-out of commitment D-C-7. The second deliverable will be a summary report of Phase 1, which outlines the results from the Hazard Identification, Bounding Analysis, and Plant Envelope Assessment. The Phase 1 report will be submitted to the CNSC to progress closure of D-C-7. Lastly, the third deliverable will be a summary report of Phase 2, which will summarize the risk assessment of vulnerable structures, systems, and components and their risk treatment plans. The Phase 2 report will be submitted to the CNSC for closure of DNNP commitment D-C-7. CNSC feedback will be obtained on strategy and deliverables for D-C-7 prior to licence to the start of construction.

2.12.5.2 Project Logic

Phase 1 Climate Change Risk Assessment and Phase 2 Risk Treatment for Vulnerable Systems are to be completed in 2023. This work will be tracked according to the 2023 NK054-PLAN-07007-00001 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts to align with the closing of existing commitment D-C-7 prior to the start of construction.

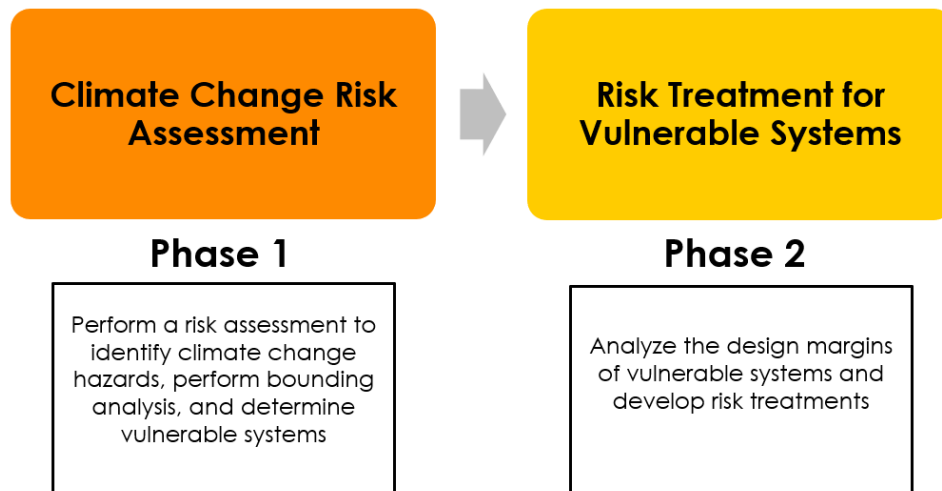


Figure 2.12.5-1: Risk Roadmap for OPG Strategy on Addressing Climate Change Impacts

2.12.5.3 Risks

None.

2.12.5.4 Impacted Chapter 2 Sections

Subsection 2.6.2 Temperature

Subsection 2.6.4 Rainfall

Subsection 2.11.9 Long Term Monitoring Program

2.12.5.5 Progress of Work

OPG issued, in January 2023, the plan as NK054-PLAN-07007-0001, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts," Ontario Power Generation

The plan has the following contents

1. Introduction
2. Objective
3. Regulatory and Governance Drivers
4. Strategy Overview
5. Lifecycle Considerations
6. Strategy Partners
7. Definitions and Acronyms
8. References

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Work is complete and results are incorporated in Subsection 2.5.4, Subsection 2.6.4, Subsection 2.6.12, and Subsection 2.11.9

Phase 1 and 2 of the 2023 NK054-PLAN-07007-0001 Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts are to be completed and tracked to the existing commitment D-C-7.

Deliverables:

1. (NK054-PLAN-07007-00001 R000), 2023, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts", Ontario Power Generation.
2. NK054-REP-07007-1049426 R001, 2023, "Darlington New Nuclear Project – Hazard Bounding Analysis," Ontario Power Generation
3. NK054-REP-07007-1028871 R000, 2022, "Darlington New Nuclear Project— Gradual Climate Change and Natural Hazard Identification," Ontario Power Generation

2.12.6 3-Second Wind Gust Speed

2.12.6.1 Background

Chapter 2, Subsection 2.6.5 requires description of the site characteristic for 3-second wind gust speed. While maximum wind speed is an instantaneous wind speed, the 3-second gust value is a sustained wind speed. Maximum wind speed is shown in Subsection 2.6.5.

Calculation of the site characteristic for 3-second gust wind is in progress and will be added in a future revision.

2.12.6.2 Project Logic

Completion of calculations is undergoing and will be updated in the subsequent revision of PSAR Chapter 2.

2.12.6.3 Assumptions

None.

2.12.6.4 Risks

None.

2.12.6.5 Impacted Chapter 2 Sections

Subsection 2.6.5 – Wind Speed

2.12.6.6 Progress of Work

1. Work is completed and a final report is delivered and accepted by OPG
2. OPG issued, in December 2022, the report as NK054-REP-02730-00003, "Wind Gust Analysis," Ontario Power Generation.
3. The report has the following contents
 1. Introduction
 2. Study Site and Data
 3. Wind Rose Diagram
 4. Frequency Analysis

5. Conclusions

6. References

Work is complete and results are incorporated in impacted sections Chapter 2

Deliverables:

1. NK054-REP-02730-00003 R000, 2022, "Wind Gust Analysis", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-003 R01)

2.12.7 Snow Load and Coincident Winter Probable Maximum Precipitation

2.12.7.1 Background

Work is ongoing to finalize appropriate consideration for snow load with a Winter PMP event. DNNP considers this a review level condition.

2.12.7.2 Project Logic

Completion of calculations is undergoing and will be updated in a subsequent revision of PSAR Chapter 2.

Winter PMP Validation - The requirements of N291 for safety related structures other than containment for 100 years snow loading is not mentioned nor the guidance in it to extrapolate the National Building Code of Canada (NBCC) 50-years value if the 100-years site snow values are not available. N291 mention this for the snow component, however, it is silent about associated rain.

For safety related structures, 100 years snow with 100 years associated rain would be required for the design.

It is recommended that OPG follow the General-Electric Hitachi recommendation in the Design Input Request for Non-Seismic External Hazards at DNNP Site document to determine the following site-specific parameters:

- 100-year return period ground snowpack
- Historical maximum snowpack, including the month of occurrence • 100-year return period ground snowfall
- Historical maximum ground snowfall
- 48-hour Winter PMP over a 25.9-square-kilometer (10-square-mile) area at this location during those months with the historically highest snowpacks.

The depth, area, and duration curves of the probable maximum storm event equivalent to the Winter PMP should be identified. (OPG, 2017)

The anticipated resulting roof loading will be situated in the range of 3.0-4.5 kPa.

2.12.7.3 Risks

None.

2.12.7.4 Impacted Chapter 2 Sections

Subsection 2.6.9 – Snow and Ice Load

2.12.7.5 Progress of Work

1. Work is completed and a final report is delivered and accepted by OPG

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2. OPG issued, in December 2022, the report as NK054-REP-02730-00004, "Winter PMP Validation," Ontario Power Generation.
3. The report has the following contents
 1. Introduction
 2. Existing Values
 3. Winter PMP Usage
 4. Conclusions
 5. References

Work is complete and results are incorporated in impacted sections Chapter 2

Deliverables:

1. NK054-REP-02730-00004 R000, 2022, "Winter PMP Validation", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-004 R01)

2.12.8 Confirmation of Probable Maximum Precipitation

2.12.8.1 Background

Subsection 2.6.4 describes the rainfall and PMP for the Darlington Nuclear site (which includes the DNNP site). Also, Subsection 2.12.4 describes an ongoing work to update the PMP and Probable Maximum Flood for the DNNP site for BWRX-300 unit 1, with potential three additional units.

This information is being supplemented by PMP Validation work being added to Subsection 2.12.4. The supplementary work is to satisfy the requirements of N291 of 100 years return period for safety related structures (similar to wind and snow), and to ensure information in: the recommendation of 21 mm for storm H (in Table 3-1 of the contractor's preliminary report) meets the NBCC as a minimum (as NBCC value for 15 min is 23mm).

2.12.8.2 Project Schedule and Logic

Confirmation work is ongoing. Subsection 2.6.4 is expected to be updated, as required, in the subsequent revision of the PSAR Chapter 2.

2.12.8.3 Assumptions

None

2.12.8.4 Risks

None

2.12.8.5 Impacted Chapter 2 Sections

Subsection 2.6.4 – Rainfall

2.12.8.6 Progress of Work

1. Work is completed and a final report is delivered and accepted by OPG
2. OPG issued, in December 2022, the report as NK054-REP-02730-00002, "PMP Validation," Ontario Power Generation.
3. The report has the following contents
 1. Introduction

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2. Storms
 - a. PMP Validation
 - b. Plant Parameter Envelop Storms
 - c. National Building Code of Canada Storms
3. Conclusions
4. References

Work is complete and results are incorporated in impacted sections Chapter 2

Deliverables:

1. NK054-REP-02730-00002 R000, 2022, "PMP Validation", Ontario Power Generation (SNC Lavalin ID 690633-0000-4HER-002 R01)

2.12.9 References

- 2.12-1 NEDO-33914-A, Revision 2, 2022, "BWRX-300 Advanced Civil Construction and Design Approach," GE-Hitachi Nuclear Energy Americas, LLC.
- 2.12-2 NK054-REP-01210-00078 R007, "Darlington New Nuclear Project Commitments Report," Ontario Power Generation.
- 2.12-3 NK054-PLAN-01210-00033, Site Geotechnical and Seismic Hazard Investigation Plan," Ontario Power Generation.
- 2.12-4 NK054-REP-01210-00012-R01, "Site Evaluation of the OPG New Nuclear at Darlington - Part 5: Flood Hazard Assessment," Ontario Power Generation.
- 2.12-5 NK054-PLAN-07007-00001 R000, 2023, "Darlington New Nuclear Project Strategy for Addressing Climate Change Impacts", Ontario Power Generation.

2.13 Appendices

Appendix A List of Industrial Facilities within the Survey Area

Appendix B List of Roads within the Survey Area

Appendix C List of Park Spaces and Water Bodies within the Survey Area

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APPENDIX A – List of Industrial Facilities within the Survey Area

Company Name	Location
McAshpalt Industries Ltd. - Oshawa Facility	Bottom of Farewell Street
Gerdau Metals Recycling - Oshawa	Waterloo Crt
TMT Salvage & Metal Recyclers	SE Corner - Nelson St & Waterloo Crt
D. Crupi & Sons Ltd.	NE Corner - Nelson St & Wellington Ave E.
Allmix Concrete Oshawa	NE Corner - Farewell St & Harbour Rd.
Coco Paving Plant	SE Corner - Wilson Rd N & Taunton Rd
Covanta Durham York	Courtice Rd. & Megawatt Dr
Courtice Water Pollution Control Plant (WPCP)	Osbourne Rd.
Miller Compost	Baseline Rd & Hancock Rd.
Hydro One Bowmanville SS	Toward bottom of Holt Rd.
St. Marys Cement Group	Bottom of Bowmanville Ave.
CBM Aggregates	Waverley Rd.
Port Darlington WPCP	E Shore Dr.
Bowmanville Water Supply Plant	E Beach Rd.

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APPENDIX B – List of Roads within the Survey Area

Name of Road / Highway / Station	Direction	Road Type
Highway 401	W-E	Hwy
Highway 418	N-S	Hwy
Highway 407	W-E	Hwy
Baseline Road W	W-E	Arterial
Courtice Road	N-S	Arterial
2nd Line W	Internal	Minor Arterial
Park Rd	N-S	Minor Arterial
Energy Dr	W-E	Arterial
Symons Rd	N-S	Minor Arterial
Crago Rd	N-S	Minor Arterial
Megawatt Dr	W-E	Minor Arterial
Osbourne Rd	N-S	Minor Arterial
Darlington Park Rd	W-E	Minor Arterial
Down Rd	N-S	Minor Arterial
Holt Rd	N-S	Arterial
Martin Rd S	N-S	Minor Arterial
Colonel Sam Dr	W-E	Minor Arterial
Cedar Crest Beach Rd	W-E	Minor Arterial
Cove Rd	W-E	Minor Arterial
W Beach Rd	N-S	Minor Arterial
Main St	W-E	Minor Arterial

Name of Road / Highway / Station	Direction	Road Type
E Beach Rd	W-E	Minor Arterial
Port Darlington Rd	N-S	Minor Arterial
Lake Rd	W-E	Minor Arterial
S Service Rd	N-S	Minor Arterial
Lookout Dr	W-E	Minor Arterial
Bennett Rd	N-S	Arterial
Wilmot Creek Dr	N-S	Minor Arterial
Heatherlea Dr		Residential
Hinkley Tr		Residential
Cliff Dr		Residential
Fir Dr		Residential
Niagara Tr		Residential
Wilmot Tr	W-E	Minor Arterial
Little Brook Rd		Residential
Bluffs Rd		Residential
Heritage Ln		Residential
The Cove Rd		Residential
Steelhead Ln		Residential
Fairway Dr		Residential
Service Rd	W-E	Minor Arterial
Bloor St E	W-E	Arterial
Farewell St	N-S	Minor Arterial
Veterans Rd	W-E	Minor Arterial
Wilson Rd S	N-S	Minor Arterial
Raleigh Ave	W-E	Minor Arterial

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Name of Road / Highway / Station	Direction	Road Type
Wentworth St W	W-E	Arterial
Marwood Dr	W-E	Minor Arterial
Harbour Rd	W-E	Arterial
Drake St	N-S	Minor Arterial
Holland St	N-S	Minor Arterial
Simcoe St S	N-S	Arterial
Nelson St	N-S	Minor Arterial
Ritson Rd S	N-S	Arterial
Dnipro Blvd	W-E	Minor Arterial
Conant St	W-E	Residential
Sylvia St		Residential
Myers St		Residential
Sharon Ave		Residential
Trafalgar Ave		Residential
Waterloo St / Crt		Residential
Tilbury St		Residential
Wellington Ave E		Residential
Kawartha Ave	W-E	Minor Arterial
Southlawn Ave	W-E	Minor Arterial
Cloverdale St	N-S	Minor Arterial
Grassmere Crt		Residential
Ravine Rd	N-S	Minor Arterial
Sandra St W/E		Residential
Wolfe St	W-E	Minor Arterial
Daniel St		Residential
Douglas St	N-S	Minor Arterial

Name of Road / Highway / Station	Direction	Road Type
4th Ave	W-E	Minor Arterial
Annis St	W-E	Minor Arterial
Rowena St	N-S	Minor Arterial
Gifford St		Residential
Phillips St		Residential
Merritt St		Residential
Knights Rd		Residential
Cedar St		Residential
Erie St		Residential
Whiting Ave	N-S	Minor Arterial
Robson St		Residential
Frank St		Residential
Valley Dr	W-E	Minor Arterial
Wecker Dr	W-E	Minor Arterial
Outlet Dr		Residential
Birchcliffe Ave	N-S	Minor Arterial
Kluane Ave	N-S	Minor Arterial
Rondeau Crt		Residential
Madawaska Ave		Residential
Sauble St		Residential
Quetico Ave / Crt		Residential
Georgian Crt		Residential
Fundy St / Crt		Residential
Phillip Murray Ave		Residential
Chaleur Ave		Residential
Sharbot St		Residential
Minden St		Residential
Scugog Ave		Residential

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Name of Road / Highway / Station	Direction	Road Type
Jasper Ave		Residential
Banff Ave		Residential
Geneva Ave		Residential
Thomas St	W-E	Minor Arterial
Tamarack Crt		Residential
Erie St		Residential
Grandview Dr	W-E	Minor Arterial
Downview Cres	W-E	Minor Arterial
Endna Crt		Residential
Welsey Dr	N-S	Minor Arterial
Down Cres		Residential
Norman Cres		Residential
Southdown Dr	N-S	Minor Arterial
Southdale Ave	W-E	Minor Arterial
Southgate Dr	N-S	Minor Arterial
Southridge St	N-S	Minor Arterial
Southport Dr		Residential
Townline Rd S	N-S	Minor Arterial
Gord Vinson Ave	W-E	Minor Arterial
Kilgannon Ave		Residential
Pickard Gate		Residential
Cornish Dr	N-S	Minor Arterial
Staples Ave		Residential
Bingham Gate		Residential
Dudley Crt		Residential
Cousins St		Residential

Name of Road / Highway / Station	Direction	Road Type
Fenning Dr	N-S	Minor Arterial
Stainton St		Residential
Roy Nichols Dr		Residential
Southfield Ave		Residential
Aylesworth Ave		Residential
Montague Ave		Residential
Frank Wheeler Ave		Residential
Eastfield Cres		Residential
Rosswell Dr		Residential
Dewell Cres		Residential
Bathgate Cres		Residential
Kersey Cres		Residential
Prestonvale Rd	N-S	Arterial
Trulls Rd	N-S	Arterial
Cigas Rd	W-E	Minor Arterial
Hancock Rd	N-S	Arterial
McKnight Rd		Residential
Courtice Crt		Residential
Solina Rd	N-S	Arterial
Rundle Rd	N-S	Arterial
Maple Grove Rd	N-S	Arterial
Boswell Dr	N-S	Minor Arterial
Ivory Crt		Residential
Shady Lane Cres		Residential
Bonathan Cres		Residential
Connors Crt		Residential
Rustwood St		Residential
Weldick Cres		Residential
Padfield Dr		Residential
Hammond St		Residential

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Name of Road / Highway / Station	Direction	Road Type
Oxley Crt		Residential
Collier Ln		Residential
Dystra Ln		Residential
Sidney Ln		Residential
Connell Ln		Residential
Farmstead Dr		Residential
Autumn harvest Rd		Residential
McBride Ave		Residential
Buxton Ln		Residential
Buttonschaw St	N-S	Minor Arterial
Woolacott Ln		Residential
McPhail Ave		Residential
Shackleton St		Residential
Kimble Ave		Residential
Remmington St	W-E	Minor Arterial
Butson Cres		Residential
Green Rd	N-S	Arterial
Clarrington Blvd	N-S	Minor Arterial
Prince William Blvd	W-E	Minor Arterial
Pethick St	N-S	Minor Arterial
Aspen Springs Dr	W-E	Minor Arterial
Baxter St	N-S	Minor Arterial
West Side Dr		Residential
Landerville Ln		Residential
Fry Cres		Residential
Vail Meadows Cres		Residential
Glen Ray Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Hartwell Ave		Residential
Candler Crt		Residential
Prestonway Dr		Residential
Bonnycastle Dr		Residential
Luttrell St		Residential
Higgon St		Residential
Brodie Crt		Residential
Martin Rd	N-S	Minor Arterial
Bagnell Cres		Residential
Abernethy Cres		Residential
Penfound Dr		Residential
Alonna St		Residential
Clancy Ln		Residential
Bottrell St		Residential
Squires Gt		Residential
Roser Cres		Residential
Walbridge Crt		Residential
Woolner Ln		Residential
Dodds Sq		Residential
Millburn Dr		Residential
Bannister St		Residential
Spicer Sq	W-E	Minor Arterial
Bowmanville Ave	N-S	Arterial
Kings Hill Ln		Residential
McCrimmon Cres		Residential
Wrenn Blvd		Residential
Rhonda Blvd	N-S	Minor Arterial
Chapel St		Residential
Roenigk Dr	W-E	Minor Arterial

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NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Waverley Rd	N-S	Minor Arterial
Strike Ave		Residential
Little Ave		Residential
Cole Ave		Residential
Trewin Ln		Residential
Lawrence Gt / Cres	N-S	Minor Arterial
Hetherington Dr		Residential
Holgate Cres		Residential
Doreen Cres		Residential
Quinn Dr		Residential
The Bridle Path		Residential
Park Ln Circ		Residential
Hillier St		Residential
Rosalynne Ave		Residential
Spry Ave		Residential
Carruthers Dr		Residential
Loscombe Dr		Residential
John Scott Ave		Residential
Lockhart Gt		Residential
Sandringham Dr	W-E	Minor Arterial
Short Cres		Residential
Avondale Dr	N-S	Minor Arterial
Caleche Ave		Residential
Richard Gay Ave		Residential
Stagemaster Cres		Residential
Fieldcrest Ave	N-S	Minor Arterial
Pingle Dr		Residential
Farmington Dr		Residential
Stonefield St		Residential

Name of Road / Highway / Station	Direction	Road Type
Wilkins Cres		Residential
Brownstone Cres		Residential
Hearthstone Cres		Residential
Weaver St		Residential
Phair Ave		Residential
Stirling Ave		Residential
Kennedy Dr		Residential
Faircomb Cres		Residential
McMann Cres		Residential
Strahallan Dr	W-E	Minor Arterial
Bushford St		Residential
Buyson Cres		Residential
Poolton Cres		Residential
Stuart Rd		Residential
Stephen Ave		Residential
Lyndale Cres		Residential
Claret Rd		Residential
Windham Cres		Residential
Parklawn Dr		Residential
Hillhurst Cres		Residential
Inglis Ave		Residential
Yorkville Dr	W-E	Minor Arterial
Granville Dr	N-S	Minor Arterial
Glenabbey Dr	W-E	Minor Arterial
Beechnut Cres		Residential
Rex Tooley Ln		Residential
Oke Rd		Residential
John Walter Cres		Residential
William Ingles Dr		Residential
Wade Sq		Residential

NEDO-33951 REVISION 2
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Adair St		Residential
Katerson Ln		Residential
Meadowglade Rd	W-E	Minor Arterial
Worden Dr		Residential
Hayman St		Residential
Cameron Ferguson St		Residential
Arnold Johnston St		Residential
Old Kingston Rd	W-E	Minor Arterial
Osgoode Gt		Residential
Robert Adams Dr	N-S	Minor Arterial
Renwick Rd		Residential
White Cliffe Dr		Residential
Halstead Rd		Residential
Hathaway Dr		Residential
Decoe Crt		Residential
Mulholland Crt		Residential
Worthington Dr		Residential
Sagewood Ave		Residential
Thornbury St		Residential
Saddlebrook Crt		Residential
Glen Eagles Dr		Residential
Pears Crt		Residential
Sheenan Crt		Residential
Hampstead Gt		Residential
Cale Ave		Residential
McRoberts Cres		Residential
Ferris Sq		Residential
Huntington Cres		Residential
Shuttleworth Dr		Residential

Name of Road / Highway / Station	Direction	Road Type
Partner Dr		Residential
Beckett Cres		Residential
Auburn Ln		Residential
Hemmingway Dr		Residential
Bruntsfield St		Residential
Newport Ave		Residential
Pebble Beach Dr		Residential
Pinedale Cres		Residential
Summerlea Crt		Residential
Turnberry Cres		Residential
Darlington Blvd	N-S	Minor Arterial
Foxhunt Tr		Residential
Empire Cres		Residential
Kingsview Crt		Residential
Edinburgh Ln		Residential
Kingswood Dr		Residential
Kingsway Gt		Residential
Barron Crt		Residential
Olive Ave	W-E	Arterial
Birkdale Crt		Residential
Sunnybrae Cres		Residential
Cherrydown Dr	W-E	Minor Arterial
Pinehurst Ave		Residential
Sunningdale Ave		Residential
Capilano Cres		Residential
Annandale St		Residential
Augusta Crt		Residential
Glenridge Crt		Residential
Labrador Dr		Residential
McClure Crt		Residential
Athabasca St	N-S	Minor Arterial

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NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Sutton Ave		Residential
Maclaren St		Residential
Erinlea Ave		Residential
Wakefield Cres		Residential
Eastlawn St	N-S	Minor Arterial
Merivale Crt		Residential
Carling Ave		Residential
Winter Ave		Residential
Mackenzie Ave		Residential
Kingsmere Ave		Residential
Belvedere Ave		Residential
Lisgar Ave		Residential
Thorncliffe St		Residential
Ridgecrest Ave		Residential
Gatineau St		Residential
Eton St		Residential
Windermere St		Residential
Cumberland Crt		Residential
Ellesmere Crt		Residential
Springdale Crt		Residential
Keewatin St S	N-S	Minor Arterial
Oriole Crt		Residential
Applegrove Ave		Residential
Oriole St		Residential
Melrose St		Residential
Basswood Ave / Crt		Residential
Viewmount St		Residential
Palm Crt		Residential
Hawthorne Crt		Residential
Lorindale Dr	N-S	Minor Arterial

Name of Road / Highway / Station	Direction	Road Type
Ivy Crt		Residential
Martindale St		Residential
Oakdale Dr		Residential
Queensdale Ave		Residential
Walnut Crt		Residential
Carnation Crt		Residential
Capri Crt		Residential
Florell Dr	N-S	Minor Arterial
Harcourt Dr		Residential
Dianne Dr	N-S	Minor Arterial
Karen Crt		Residential
Brenda Crt		Residential
Susan Crt		Residential
Denise Dr		Residential
Ronlea Ave		Residential
Carolyn Ave		Residential
Cherryhill St		Residential
St Andrews St		Residential
Augusta Ave		Residential
Palace St	W-E	Minor Arterial
Brunswick St / Crt		Residential
Riverside Dr N/S	N-S	Minor Arterial
Hoskin Ave		Residential
Taylor Ave	W-E	Minor Arterial
Poplar St / Crt		Residential
Linden St / Crt		Residential
Elmridge St		Residential
Wicklow Dr		Residential
Chesterton Ave		Residential

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NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Browning St		Residential
Shelley Ave		Residential
Tennyson Ave / Crt		Residential
Milton St		Residential
Emerson Ave / Crt		Residential
Coleridge St		Residential
Whitman Cres		Residential
Dean Ave	W-E	Minor Arterial
Addison Cres		Residential
Carman Crt		Residential
Shakespeare Ave	W-E	Minor Arterial
Byron Crt		Residential
Keates Ave		Residential
Chaucer Ave		Residential
Macaulay St		Residential
Loring St		Residential
Austen Crt		Residential
Guelph St	N-S	Minor Arterial
Baldwin St	N-S	Minor Arterial
Windsor St	N-S	Minor Arterial
Crerar Ave	W-E	Minor Arterial
Gliddon Ave	W-E	Minor Arterial
Devon Ave		Residential
Athol St E	W-E	Minor Arterial
Highland Ave	N-S	Minor Arterial

Name of Road / Highway / Station	Direction	Road Type
Cadillac Ave N / S	N-S	Minor Arterial
Lasalle Ave	N-S	Minor Arterial
Central Park Blvd N/S	N-S	Minor Arterial
Arthur St	W-E	Minor Arterial
Bruce St	W-E	Minor Arterial
Oshawa Blvd N/S	N-S	Minor Arterial
Rowe St		Residential
Eulalie Ave	W-E	Minor Arterial
Festhubert Ave		Residential
Courcellette Ave		Residential
Vimy Ave		Residential
Verdun Rd		Residential
St Eloi Ave		Residential
Chadburn Crt		Residential
Mitchell Ave	W-E	Minor Arterial
Viola St		Residential
Kitchener Ave		Residential
Monsah Ave		Residential
Currie Ave		Residential
Montgomery St		Residential
Christine Cres		Residential
Nevis Ave		Residential
Normandy St		Residential
Lomond St		Residential
Dieppe Ave / Crt		Residential
Sterling Ave		Residential
Hillcrest Dr		Residential
Dunkirk Ave		Residential

NEDO-33951 REVISION 2
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Sedan Cres		Residential
Brest Crt		Residential
Drew St	N-S	Minor Arterial
Huron St	N-S	Minor Arterial
Charles St	N-S	Minor Arterial
Court St		Residential
Mary St N / S	N-S	Arterial
Albert St	N	Minor Arterial
Celina St	S	Minor Arterial
John St W / E	W-E	Minor Arterial
Emma St	W-E	Minor Arterial
Hogarth St		Residential
Wilkinson Ave		Residential
Elm St	W-E	Minor Arterial
Maple St		Residential
Banting Ave	W-E	Minor Arterial
Barrie Ave	W-E	Minor Arterial
McKim St		Residential
Summer St		Residential
Stacey Ave	W-E	Minor Arterial
Tylor Cres		Residential
George St		Residential
Edward Ave		Residential
Graburn Ave		Residential
Beatty Ave		Residential
McNaughton Ave		Residential

Name of Road / Highway / Station	Direction	Road Type
Etna Ave		Residential
Toronto Ave		Residential
Jackson Ave		Residential
Howard St	N-S	Minor Arterial
First Ave	W-E	Minor Arterial
Lviv Blvd		Residential
Third Ave		Residential
Front St	N-S	Minor Arterial
Elena Ave		Residential
Albany St		Residential
Fisher St		Residential
Ray St	N-S	Minor Arterial
Ontario St	N-S	Minor Arterial
Richmond St E	W-E	Minor Arterial
Colborne St E	E	Minor Arterial
Brock St E	W	Minor Arterial
Elgin St E	W-E	Minor Arterial
Dearborn Ave		Residential
Kendal Ave		Residential
Carriage Works Dr	N-S	Minor Arterial
William St E	W-E	Minor Arterial
Divison St	N-S	Minor Arterial
Agnes St		Residential
kenneth Ave	N-S	Minor Arterial

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Name of Road / Highway / Station	Direction	Road Type
Roxborough Ave	N-S	Minor Arterial
Patricia Ave		Residential
Delroy Crt		Residential
Westminister Ave		Residential
Beverly St		Residential
Luke St		Residential
Oakes Ave		Residential
Lasalle Crt		Residential
Rogers St		Residential
Dover St		Residential
Digby Ave		Residential
Surrey Dr		Residential
Coventry Crt		Residential
Landsdowne Dr		Residential
Sussex St		Residential
Claymore Cres		Residential
Cambridge Ave		Residential
Regent Dr	W-E	Minor Arterial
Eastglen Dr		Residential
Easthaven St		Residential
Florian Crt		Residential
Eastgrove Ave		Residential
Eastdale Ave		Residential
Eastbourne Ave		Residential
Ascot Crt		Residential
Arden Dr / Crt		Residential
Acadia Dr		Residential
Eastmount St		Residential
Parklane Ave		Residential
Woodlane Crt		Residential
Baker Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Beaufort Ave / Crt		Residential
Southwood St		Residential
Conifer St		Residential
Cherry St		Residential
Holly Crt		Residential
Cleta Crt		Residential
Briar Crt		Residential
Laurel Crt		Residential
Heather Crt		Residential
Newbury Ave		Residential
Grandview St N	N-S	Minor Arterial
Cardinal Crt		Residential
Bluefinch Crt		Residential
Blue Heron Dr		Residential
Killdeer Dr		Residential
Bluejay Cres		Residential
Norwood Crt		Residential
Fleetwood Dr		Residential
Eldorado Ave		Residential
Belair Cres		Residential
Kingsway College Dr		Residential
Rockcliffe St		Residential
Maracle Rd		Residential
Violet Hall Rd		Residential
Clarence Biesenthal Dr		Residential
Leland Rd		Residential
Wilbert Bresett Rd		Residential
Wagar Crt		Residential
Shankel Rd		Residential
Bradenton Path		Residential

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Name of Road / Highway / Station	Direction	Road Type
Apollo St		Residential
Malibu St		Residential
Wood St		Residential
Rolson St		Residential
Haig St		Residential
French St		Residential
Jarvis St	N-S	Minor Arterial
Kingsdale Ave / dr		Residential
Leslie Ave		Residential
Aberdeen St		Residential
Masson St	N-S	Minor Arterial
Leslie St		Residential
Rosedale Ave	W-E	Minor Arterial
Grove Ave		Residential
Sutherland Ave		Residential
Connaught St		Residential
Hillcroft St	W-E	Minor Arterial
Adeline Ave		Residential
Trick Ave		Residential
Pearson St		Residential
Greta St	W-E	Minor Arterial
Grierson St		Residential
Minto St / Crt		Residential
Hillsdale Ave		Residential
Laracor Ln		Residential
Jasmine Cres		Residential
Lilac Crt		Residential
Tulip Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Darcy St	W-E	Minor Arterial
Juniper St / Crt		Residential
Violet Crt		Residential
Verbana Crt		Residential
Wildflower Crt		Residential
Marigold Ave / Crt		Residential
Robert St	W-E	Minor Arterial
Gardenia Crt		Residential
Orchid Crt		Residential
Lavender Crt		Residential
Marica Ave		Residential
Caledon Crt		Residential
Spirea Crt		Residential
Sycamore Cres		Residential
Iris Crt		Residential
Trillium Crt		Residential
Beatrice St E	W-E	Minor Arterial
Lobelia Crt		Residential
Nonquon Rd		Residential
Pentland St		Residential
Lauder Rd		Residential
Maplewood Dr		Residential
Orange Cres		Residential
Juliana Dr		Residential
Bernhard Cres		Residential
Amstel Cres		Residential
Marken Cres		Residential
Arnhem Dr		Residential
Holcan Ave		Residential
Fernwood Ave		Residential
Rembrandt Crt		Residential

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NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Crestwood Dr		Residential
Everson Crt		Residential
Oakwood Ave		Residential
Brentwood Ave		Residential
Edgewood Ave		Residential
Beechwood St		Residential
Pinewood St		Residential
Dogwood Ave		Residential
Harwood Dr		Residential
Humewood Ave		Residential
Wychwood St		Residential
New Gate Ave		Residential
Clifton Dr		Residential
Rodney Crt		Residential
Lexington St		Residential
Exeter St	N-S	Minor Arterial
Mayfair Ave		Residential
Terrace Dr	N-S	Minor Arterial
Canonberry Crt		Residential
Ashley Crt		Residential
Hackney Crt		Residential
Carnaby Crt		Residential
William Booth Cres		Residential
Lambeth Crt		Residential
Charrington Ave		Residential
Whitehall Crt		Residential
Downing Crt		Residential
Tiffany Circ		Residential
Paddington Cres		Residential
Old Brampton Crt		Residential
Chelsea Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Old Pye Crt		Residential
Torrington Crt		Residential
Trowbridge Crt		Residential
Highgate Ave		Residential
Burnley Crt		Residential
Cardigan Crt		Residential
Compton Cres		Residential
Kensington Cres		Residential
Trowbridge Dr		Residential
Dover St		Residential
Brighton Crt		Residential
Aspen Crt		Residential
Gothic Crt / Dr		Residential
Greenbriar Dr		Residential
Grange Crt		Residential
Camelot Crt / Dr		Residential
Chancery Crt		Residential
Gaylord Dr		Residential
Merlin Crt		Residential
Percival Crt		Residential
Cavendish Crt		Residential
Lancelot Cres		Residential
Gentry Cres		Residential
Glebe Ave		Residential
Galahad Dr		Residential
Gladfern St		Residential
Pascoe Crt		Residential
Avery Crt		Residential
Deauville Crt		Residential
Attersley Dr		Residential
Bayla Crt		Residential
Foxrun Crt		Residential
Cricklewood Dr		Residential

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NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Cobblehill Dr		Residential
Courville Crt		Residential
Bennett Cres		Residential
Mountjoy Crt		Residential
Hayes Ave		Residential
Lavis St / Crt		Residential
Storie Ave		Residential
Dyer Crt		Residential
Crowells St / Crt		Residential
Meadowhill Crt		Residential
Trailridge Cres		Residential
Cresthill Crt		Residential
Strawberry Crt		Residential
Pepperbush Crt		Residential
Elderberry Dr		Residential
Idylwood Crt		Residential
Greenlane Dr / Crt		Residential
Pondtail Crt		Residential
Beaconhill Crt		Residential
Snowberry St / Crt		Residential
Wolfberry Crt		Residential
Buttonbush Crt		Residential
Keswick Crt		Residential
Greystone Crt		Residential
Brasswinds Tr		Residential
Songbird Dr		Residential
Cascade Dr		Residential
Summerwood Hgts		Residential
Silverfox Crt		Residential
Grand Ridge Ave		Residential
Taggart Cres		Residential

Name of Road / Highway / Station	Direction	Road Type
Langley Circ / Gt		Residential
Walter Ave		Residential
Blackthorn St		Residential
Nina Crt		Residential
Cranberry St		Residential
Pinetree Crt		Residential
Thimbleberry Circ		Residential
Palmtree Cres		Residential
Lemans Ave		Residential
Safari Dr		Residential
Century St		Residential
Skylark Ave		Residential
Laguna St		Residential
Corsica Ave		Residential
Astra Ave		Residential
Le Sabre St		Residential
Andover Crt / Dr		Residential
Vega St		Residential
Nova St		Residential
Kilmaurs Ave / Crt		Residential
Dartmoor St		Residential
Hartgrove Ln		Residential
Aldershot Dr		Residential
Faywood Cres		Residential
Margate Dr		Residential
Nottingham Cres		Residential
Langford St		Residential
Shaftsbury St		Residential
Oldman Rd		Residential
Cotsworld Crt		Residential
Dickers Dr		Residential
Traddles Ave		Residential
Wickham St		Residential

NEDO-33951 REVISION 2
NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Micawber St		Residential
Peggotry Circ		Residential
Copperfield Dr		Residential
Steerforth St		Residential
Coyston Crt / Dr		Residential
Beneford Rd		Residential
Jim Brewster Circ		Residential
Drinkle Cres		Residential
Wadebridge Cres		Residential
Autumnwood Tr		Residential
Kettering Dr		Residential
Krawchuk Cres		Residential
Oxbow Cres		Residential
Aldsworth Cres		Residential
Cronk Crt		Residential
Hanmore St / Crt		Residential
Baynes Ave		Residential
Maddock Dr / Crt		Residential
Corbetts Rd		Residential
Grandlea Crt		Residential
Ripley Cres		Residential
Kingsley Crt		Residential
Lindsay Blvd		Residential
Sproule Cres		Residential
Stone Cottage Cres		Residential
Royal Orchard Dr		Residential
Ridge Valley Dr		Residential
Sandcliff Dr		Residential
Rathburn St		Residential
Trail Valley Dr		Residential
Pondview Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Edward Bolton Cres		Residential
Tall Pine Crt		Residential
Glenbourne Dr / St / Crt		Residential
Glaspell Cres		Residential
Gyatt Cres		Residential
Whitelaw Ave		Residential
Stire St		Residential
Meath dr		Residential
Magnolia Ave		Residential
Ashgrove Cres		Residential
Liveoak St		Residential
Ridgemount Blvd		Residential
Macinally Crt		Residential
Benson St		Residential
Mountview Dr / Crt		Residential
Highbrooke Crt		Residential
Summitview Cres		Residential
Forest Hill Crt		Residential
Springbank Dr		Residential
Westridge Dr / Crt		Residential
Roseheath St		Residential
Hinterland Crt		Residential
Swiss Hgts		Residential
Matterhorn St		Residential
Oberland Dr		Residential
Interlake Dr		Residential
William Tell Dr		Residential
Briarwood Dr		Residential
Pinecrest Rd		Residential
Bridle Crt		Residential
Varcoe Rd		Residential

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NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Dale Park Dr		Residential
Dalepark Dr		Residential
Cherry Blossom Cres		Residential
Briar Hill Gate		Residential
Valleycrest Dr		Residential
Centrefield Dr		Residential
Bellevue Crt		Residential
Windsor Valley Pl / Gt		Residential
Black Creek Tr		Residential
Carriage Ln		Residential
Barrington Pl		Residential
Nash Rd	W-E	Minor Arterial
Lawson Rd		Residential
Wabbokish Crt		Residential
Sheco Crt		Residential
Cloverfield St		Residential
Washburn Park		Residential
Spyfield Tr		Residential
Tooley Rd	N-S	Minor Arterial
Rowland Crt		Residential
McLellan Dr		Residential
Oban Crt		Residential
Alderbrook Dr		Residential
Goldpine Ave		Residential
Abbeywood Cres		Residential
Mossgrave Crt		Residential
Devondale St		Residential
George Reynolds Dr		Residential
Mull Cres		Residential

Name of Road / Highway / Station	Direction	Road Type
Birchfield Dr		Residential
Centrefield Dr		Residential
Homefield Sq		Residential
Oakfield Gt		Residential
Hartsfield Dr		Residential
Old Varcoe Rd		Residential
Mahaffy Pl		Residential
Springfield Ln		Residential
McLean Rd		Residential
Longwood Crt		Residential
Broadlands Cres		Residential
Firwood Ave		Residential
Kintyre St		Residential
Dunkin Ave		Residential
Arran Crt		Residential
Leith Crt		Residential
Jura Crt		Residential
Islay Crt		Residential
Mallory St		Residential
Daiseyfield Ave		Residential
Page Pl		Residential
Adelaide Ave		Residential
Niddery St		Residential
Vetzel Crt		Residential
Vivian Dr		Residential
Timberlane Crt		Residential
Sherry Ln		Residential
Prince Rupert Dr		Residential
Lord Duncan Crt		Residential
Firmer St		Residential
Fices Rd		Residential
Richfield Sq		Residential
Westmore St		Residential

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NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Lynwood Ave		Residential
Glenview Rd		Residential
Fourth Ave		Residential
Jane Ave		Residential
Sleeman Sq		Residential
Cecil Found Cres		Residential
Pidduck St		Residential
Meredith Crt		Residential
Skinner Crt		Residential
Pebblestone Rd	W-E	Minor Arterial
Tyler St		Residential
Leith Crt		Residential
Bradley Blvd		Residential
Progress Dr		Residential
Fewster St		Residential
Jolliffe St		Residential
Living Crt		Residential
Moyse Dr		Residential
Moulton Crt		Residential
Simnick Cres		Residential
Harry Gay Dr		Residential
Duval St		Residential
Tabb St		Residential
Elmer Adams Dr		Residential
Holyrod Dr		Residential
Arthur Trewin St		Residential
Gordon Cowling St		Residential
Brookhill Blvd		Residential
Meachin gt		Residential

Name of Road / Highway / Station	Direction	Road Type
Hovey Ln		Residential
Ted Miller Cres		Residential
Daigle Ln		Residential
Purdy Pl		Residential
Quick Tr		Residential
Murray Tabb St		Residential
Harvey Jones Ave		Residential
Summersford Dr		Residential
Gough Ln		Residential
Carl Raby St		Residential
Forsey Ln		Residential
Ross Wright Ave		Residential
Kilpatrick Crt		Residential
Stevens Rd	W-E	Minor Arterial
Uptown Ave		Residential
Old Scugog Rd	N-S	Minor Arterial
Buttery Crt		Residential
Maryleah Crt		Residential
Taunus Crt		Residential
Glenelge Crt		Residential
Craig Crt		Residential
Munday Crt		Residential
Wellington St		Residential
Sturrock Ave		Residential
Rehder Ave		Residential
Edsall Ave		Residential
Frederick Ave		Residential
Luvmere Crt		Residential
Linden Ln		Residential
Barbara St		Residential

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Name of Road / Highway / Station	Direction	Road Type
Jackman Rd		Residential
Don Morris Crt		Residential
Mill Ln		Residential
West Scugog Ln		Residential
Terry Cres		Residential
Willoughby Pl		Residential
Kaukonen Crt		Residential
Crockett Pl		Residential
N Scugog Crt		Residential
Westover Dr		Residential
Piper Cres		Residential
Hockley Ave		Residential
Nicks St		Residential
Childs Crt		Residential
Bons Ave		Residential
Lunney Cres		Residential
Goddall Cres		Residential
Dan Sheehan Ln		Residential
Edwin Carr St	N-S	Minor Arterial
Kenneth Cole Dr		Residential
Carey Ln		Residential
Richard Davies Cres		Residential
Robb Ln		Residential
Sidney Rundle Ave		Residential
Northglen Blvd	W-E	Minor Arterial
Loana Ln		Residential
Jerome Way		Residential
Moses Cres		Residential
Crombie St		Residential

Name of Road / Highway / Station	Direction	Road Type
John Matthew Cres		Residential
Jack Roach St		Residential
Ray Richards St		Residential
Fred Jackman Ave		Residential
William Fair Dr		Residential
Bruce Cameron Dr		Residential
Arthur McLaughlin St		Residential
Henry Smith Ave		Residential
Temperance St	N-S	Minor Arterial
Silver St		Residential
Brown St		Residential
Church St	W-E	Minor Arterial
Horsey St		Residential
Beech Ave		Residential
Lowe St		Residential
Liberty Pl		Residential
Carlisle Ave		Residential
Centre St		Residential
Grants Ln		Residential
Alexander Blvd		Residential
Lovers Ln		Residential
Concession St W / E	W-E	Minor Arterial
O'Dell St		Residential
Prospect St	N-S	Minor Arterial
High St		Residential
Burk Crt		Residential
Borland Crt		Residential
Saunders Crt		Residential

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NON-PROPRIETARY INFORMATION

Name of Road / Highway / Station	Direction	Road Type
Lorraine Crt		Residential
Prout Dr		Residential
Lambs Ln		Residential
Elgin St E		Residential
First St		Residential
2nd St		Residential
3rd St		Residential
Bernard St		Residential
Summerfield Crt		Residential
Sunset Rd		Residential
Vanstone Crt		Residential
Sunicrest Crt		Residential
Veterans Ave		Residential
4th St		Residential
Hilltop Dr		Residential
Shoreview Dr / Crt		Residential
Meadowview Blvd		Residential
Aldcroft Cres		Residential
Clayton Cres		Residential
Argent St		Residential
Longworth Ave	W-E	Minor Arterial
Daley Ave		Residential
Hogan cres		Residential
Markham Tr		Residential
Streathern Way		Residential
Ken Bromley Lane		Residential
Somerscales Dr		Residential
Laurelwood St		Residential
Willey Dr		Residential
Birmingham Ave		Residential
Goodwin Ave		Residential

Name of Road / Highway / Station	Direction	Road Type
Honeyman Dr		Residential
Darryl Caswell Way		Residential
Allworth Cres		Residential
Allison St		Residential
Lander Cres		Residential
Colville Ave		Residential
Wyse Gt		Residential
Gimblett St		Residential
Courtney St		Residential
Brough Crt		Residential
McCorkell St		Residential
Jennings Dr		Residential
Keeler Cres		Residential
David Baker Crt		Residential
Bavin St		Residential
Higbee Ln		Residential
Ambereen Pl		Residential
Concession Road 3	W-E	Minor Arterial
Northglen Blvd	N-S	Minor Arterial
John Stalker Dr		Residential
Harry Lee Cres		Residential
Higham Pl		Residential
Rebecca Crt		Residential
Pamela Crt		Residential
Avi Crt		Residential
Sydel Crt		Residential
Gary Crt		Residential
Middle Rd	N-S	Minor Arterial
Concession Road 4	W-E	Minor Arterial

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Name of Road / Highway / Station	Direction	Road Type
Liberty St N	N-S	Arterial
Scugog St	N-S	Arterial
Soper Crt		Residential
Hobbs Dr		Residential
Duke St		Residential
Wharf St		Residential
Simpson Ave		Residential
Mearns Crt		Residential
Caristrap St		Residential
Lambs Rd	N-S	Minor Arterial
Haines St	N-S	Minor Arterial
Parkway Ave / Cres		Residential
Flett St		Residential
Southway Dr		Residential
Nelson St		Residential
Morgandale Cres		Residential
Deerpark Cres		Residential
Jane St		Residential
Wilde Crt		Residential
Hailey Crt		Residential
Ashdale Cres		Residential
Prince St		Residential
Frank St		Residential
Queen St	W-E	Minor Arterial
Mearns Ave	N-S	Minor Arterial
Lambert St		Residential
Church St		Residential
Kingscourt Rd		Residential
Galbraith Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Climie Crt		Residential
Royal Pines Crt		Residential
Orchard Park Dr		Residential
Peachtree Cres		Residential
Strathmanor Dr		Residential
Merryfield Crt		Residential
Trudeau Dr		Residential
Marchwood Cres		Residential
Orr Cres		Residential
Hendy Gt		Residential
Dadson Dr		Residential
Squire Fletcher Dr		Residential
McFeeters Cres		Residential
Clinton Crt		Residential
Soper Creek Dr		Residential
Downham Dr		Residential
Souch Crt		Residential
Barley Mill Cres		Residential
Farncomb Cres		Residential
Herriman St		Residential
Mann St		Residential
Tucker Rd		Residential
Baker Crt		Residential
Apple Blossom Blvd		Residential
Glanville Cres		Residential
Tilley Rd		Residential
Bradshaw St		Residential
Maconnachie Pl		Residential
Kershaw St		Residential
Chance Crt		Residential
Edgerton Dr		Residential

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Name of Road / Highway / Station	Direction	Road Type
Swindells St		Residential
Flaxman Ave		Residential
Forrester Dr		Residential
Redfern Cres		Residential
Elephant Hill Dr		Residential
Ireland St		Residential
Lyle Dr		Residential
Brent Cres		Residential
Scottsdale Dr		Residential
Assunta Ln		Residential
Courvier Cres		Residential
Quackenbush St		Residential
William Cowles Dr		Residential
Barlow Crt		Residential
Brooking St		Residential
Stephens Gulch Dr		Residential
Eldad Dr		Residential
Rickaby St		Residential
Dart Crt		Residential
Guildwood Dr		Residential
Lownie Crt		Residential
Budd Ln		Residential
Sprucewood Cres		Residential
Hutton Pl		Residential
Madden Pl		Residential
Cotton St		Residential
Taft Pl		Residential
Crough St		Residential
Hanna Dr		Residential
Laprade Sq		Residential
Lobb Crt		Residential

Name of Road / Highway / Station	Direction	Road Type
Fenwick Ave		Residential
Freeland Ave		Residential
Hanning Crt		Residential
Elford Dr		Residential
Pomeroy St		Residential
Bates Crt		Residential
Jollow Dr		Residential
Maxwell Crt		Residential
Hooper Sq		Residential
Champine Sq		Residential
Bethesda Rd	N-S	Minor Arterial
Stephen Mills Rd		Residential
Darlington Clarke Townline Rd	N-S	Minor Arterial
Bennett Rd	N-S	Minor Arterial
Baseline Rd E	W-E	Minor Arterial
Rickard Rd	N-S	Minor Arterial
Providence Rd	N-S	Minor Arterial
Bragg Rd	N-S	Minor Arterial
Taunton Rd	W-E	Arterial
Highway 2	W-E	Arterial
Cobbledick Rd	N-S	Minor Arterial
Lovekin Rd	W-E	Minor Arterial
Browview Rd	W-E	Minor Arterial
Gibson Rd		Residential
Pollard Rd	N-S	Minor Arterial

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Name of Road / Highway / Station	Direction	Road Type
Concession Road 5	W-E	Minor Arterial
Hwy 115	N-S	Arterial
Canadian Pacific Railway North of Hwy 401	W-E	Rail
Canadian Pacific Railway South of Hwy 401	W-E	Rail
Oshawa Executive Airport		Airport
Port of Oshawa East Pier		Pier

NEDO-33951 REVISION 2
NON-PROPRIETARY INFORMATION

APPENDIX C – List of Park Spaces and Water Bodies within the Survey Area

Park Spaces and Water Bodies	Location
Lakeview Park / Beach	Oshawa
Southmead Park	Oshawa
Lake Ontario	Multiple municipalities
Oshawa Creek	Oshawa
Cordova Park	Oshawa
Chopin Park	Oshawa
Eastview Park	Oshawa
Woodview Park	Oshawa
Connaught Park	Oshawa
Centennial Park	Oshawa
Central Park	Oshawa
Northway Court Park	Oshawa
North Oshawa Park	Oshawa
Hyde Park	Oshawa
Bathe Park	Oshawa
Conant Park	Oshawa
Kingside Park	Oshawa
Knights of Columbus Park	Oshawa
Eastbourne Park	Oshawa
Galahad Park	Oshawa
Attersley Park	Oshawa
Swiss Height Park	Oshawa
Iroquois Shoreline Park	Oshawa
Ridge valley Park	Oshawa
Corbett's Park	Oshawa
Harmony Valley Conservation Area	Oshawa
Easton Park	Oshawa
Baker Park	Oshawa

Park Spaces and Water Bodies	Location
Martindale Park	Oshawa
Harmony Village Park	Oshawa
Florell Park	Oshawa
Grandview North / South Park	Oshawa
Second Marsh Wildlife Area	Oshawa
McLaughlin Bay	Oshawa
McLaughlin Bay Wildlife reserve	Oshawa
Rosswell Park	Courtice
Terry Fox Park	Oshawa
"Oshawa Valleylands Conservation Area"	Oshawa
MacKenzie Park	Oshawa
Margate Park	Oshawa
Kettering Park	Oshawa
Pinecrest Park	Oshawa
Glenbourne Park	Oshawa
South Courtice Dog Park	Courtice
Gatehouse Parkette	Courtice
Glenabbey Park	Courtice
Courtice Duck Pond	Courtice
Tooley's Mill Park	Courtice
Courtice West Park	Courtice
Highland Park	Courtice
Penfound Park	Courtice
Bathgate Park	Courtice
Darlington Provincial Park	Bowmanville
Stuart Park	Courtice
Zion Park	Clarington
Avondale Park	Courtice
Alijco Beach	Courtice

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Park Spaces and Water Bodies	Location
W & D Courtice Memorial Park	Courtice
Moyse Parkette	Courtice
Darlington Hydro Soccer Fields	Clarington
Darlington Waterfront Trail	Clarington
Burk Pioneer Cemetery	Clarington
Harvey Jones Park	Bowmanville
Green Park	Bowmanville
Baxter Park	Bowmanville
Baseline Park	Bowmanville
West Side Park	Bowmanville
Landerville Parkette	Bowmanville
Northglen park	Bowmanville
Douglas Kemp Parkette	Bowmanville
Bons Park	Bowmanville
"Bowmanville Valley Conservation Area"	Bowmanville
Rotary Park	Bowmanville
Bowmanville Creek Barrier Dam	Bowmanville
Waverley Park	Bowmanville
"Bowmanville Westside Conservation Area"	Bowmanville
Bowmanville Harbour	Bowmanville
Port Darlington West / East Beach	Bowmanville
Lions Parkette	Bowmanville
Nelson Parkette	Bowmanville
Argent Park	Bowmanville
Barlow Court Parkette	Bowmanville
Elephant Hill Park	Bowmanville

Park Spaces and Water Bodies	Location
Bowmanville Cemetery	Bowmanville
"Bowmanville Soper Creek Playground"	Bowmanville
Guildwood Park	Bowmanville
Stephen Gulch's Conservation Area	Bowmanville
Samuel Wilmont Natural Area	Newcastle
Mearns Park	Bowmanville
Soper Creek Trail	Bowmanville



HITACHI

GE Hitachi Nuclear Energy

NEDO-33952

Revision 1

March 7, 2023

Non-Proprietary Information

**Ontario Power Generation Inc.
Darlington New Nuclear Project
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 3
Safety Objectives and Design Rules for
Structures, Systems and Components**

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
0	All	Initial Release
1	3.0	Updated to 300 MWe.
	3.1.6.2	Included full acronym name for Reactor Pressure Vessel for first time use.
	3.1.7.4	Updated acronym use of DL1.
	3.1.7.9.2	Updated use of RPV acronym.
	3.2.1.1	Edited Safety Category 3 wording.
	3.2.1.3	Edited Primary Function wording.
	3.2.1.4	Added details to Delayed Functions.
	3.2.1.6	Updated, added text and added reference to Table 3.2-2.
	3.2.3	Included full acronym name for Design Basis Earthquake and a pointer to Section 3.3-1.
	3.2.3.1	Added CSA N289.3 reference and updated text.
	3.2.4	Updated reference to Table 3.2-3 (from Table 3.2-2).
	3.2	Added new Table 3.2-2 for Safety Class for SSC.
	Acronym List	DGRS and NBC acronyms added.
	3.3	Updated pointer to Subsection 3.3.7.4
	3.3.1 – 3.3.7	Cross-references to Chapter 2 updated as required.
	3.3.1.1, 3.3.1.1.1-3.3.1.1.4	Updated to incorporate bounding information previously documented in Chapter 2, Section 2.7.
	3.3.1.1.6	Updated content on development of dynamic subgrade profiles and included pointer to Subsection 3.5.2.2.
	3.3.2.1 – 3.3.2.5	Updates made to decouple from Chapter 2 and present bounding design parameters.
	3.3.6.1	Removed reference to Chapter 19 for Fire Protection Program.
	3.3.8	References 3.3-12 to 3.3-20 and 3.3-26 and 3.3-28 added.
	Table 3.3-1	Added reference to CSA N289 series for basis.

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Revision #	Section Modified	Revision Summary
	Table 3.3-2 to 3.3-5	Tables added to supplement added bounding information content in 3.3.1.1.1 – 3.3.1.1.4.
	Figure 3.3-1, 3.3-2, 3.3-5, 3.3-12	Figures added to supplement added bounding information content in 3.3.1.1.1 – 3.3.1.1.4.
	3.4.1.1	Removed reference to Chapter 19 for Fire Protection Program.
	3.4.4	Rephrased reference to areas where postulated pipe breaks are excluded to indicate future analyses are required.
	3.4.4.2.2	Edited Location of Postulated Pipe Break subsection.
	3.4.4.2.3	Edited Location of Postulated Pipe Crack subsection.
	3.5.1, 3.5.2.7 and 3.5.4.1	Reference to NEDC-33926P added.
	3.5.2.2	Revised to incorporate bounding information previously documented in Chapter 2, Section 2.7. Additional text added regarding upper bound nominal water table levels.
	3.5.2.2.1	Bounding Equivalent Subgrade Static Profile Subsection updated.
	3.5.2.2.3	Edited to remove content covered in 3.5.2.2.1.
	3.5.4 and 3.5.4.4.1	Updated containment internal structure descriptions included.
	3.5.5.2.1	Pointer to Design Basis Threat subsection revised.
	3.5.5.4.1	Seismic and Extreme Wind sub-heading revised.
	3.5.7	References 3.5-11, 3.5-12 and 3.5-14 through 19 added to supplement Subsection 3.5.2.2 and 3.5.2.2.1 added information.
	Table 3.5-1 and 3.5-2	Tables added to supplement content update in 3.5.2.2.1.
		Minor editorial updates throughout.
	3.6.3.12	Safety Class 1 updated to Safety Category 1
	3.6.7.2.5	Editing Safety Category wording.
	3.9.2	Updated scope for DEC assessments.
	3.9.3.1	Added RD-2.5.2 reference and updated seismic categorization text.

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Revision #	Section Modified	Revision Summary
	3.9.3.2	Seismic interaction equipment details removed.
	3.9.3.2.1	Seismic test details added and edits made.
	3.9.3.3	Section 3.3.1.3 pointer added.
	3.9.3.5	Renamed from Seismic Margin to Beyond Design Basis Earthquake and updated content.
	3.9.4.1	Revised content and included additional references.
	3.9.4.4.1	Revised content including DBA groupings.
	Minor editorial updates made including Safety Class 1 updated to SC1 throughout	
	Table 3.12-1	SSC classification table updated to include the latest information (Radiation Monitoring Systems, Wide range pool level instrumentation, Leak detection equipment updated).
	Sections 3.13 – 3.18	Appendices 3B – 3G identifying and describing computer software have been updated to align with the latest information. Where there is a discrepancy identified between software version numbers in these appendices and other PSAR chapters, this appendix should be taken as correct.
	Appendix 3C	Title updated.
	Section 3.14	Introduction description of scope edited.
	Appendix 3D	Title updated.
	Section 3.15	Introduction description of scope edited.
	Appendix 3E	Title updated.
	Section 3.16	Introduction description of scope edited.
	Sections 3.16.18 and 3.17.10	Computer code descriptions updated.
	Appendix 3F	Title updated.
	Section 3.17	Introduction description of scope edited.
	Editorial changes made throughout.	

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ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
AEF	Annual Exceedance Frequency
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
API	American Petroleum Institute
ARS	Acceleration Response Spectra
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATH	Acceleration Time History
ATWS	Anticipated Transient Without Scram
AWWA	American Water Works Association
BDBA	Beyond-Design Basis Accident
BDBE	Beyond-Design Basis Earthquake
BDBT	Beyond-Design Basis Threat
BE	Best Estimate
BIS	Boron Injection System
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
BWRX	Boiling Water Reactor, 10th Design
CB	Control Building
CAD	Computer-Aided Design
CCF	Common Cause Failure
CAFTA	Computer Aided Fault Tree Analysis
CANDU	CANada Deuterium Uranium
CB	Control Building
CEPSS	Containment Equipment and Piping Support Structure
CGD	Canadian Geodetic Datum
CIV	Containment Isolation Valve

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Acronym	Explanation
CLE	Checking Level Earthquake
CNSC	Canadian Nuclear Safety Commission
CPR	Critical Power Ratio
CR	Control Room
CRD	Control Rod Drive
CSA	Canadian Standards Association
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
D-in-D	Defence-in-Depth
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DBT	Design Basis Threat
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DGRS	Design Ground Response Spectrum
DL3	Defense Line 3
DL	Defense Line
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
FE	Finite Element
FIA	Foundation Interface Analysis
FIRS	Foundation Input Response Spectra
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analyses
FPC	Fuel Pool Cooling and Cleanup System
FSF	Fundamental Safety Function
FW	Feedwater
GEH	GE-Hitachi Nuclear Energy
GMRS	Ground Motion Response Spectra

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Acronym	Explanation
GUI	Graphical User Interface
HCLPF	High Confidence of Low Probability of Failure
HCU	Hydraulic Control Unit
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HFE	Human Factors Engineering
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation and Air Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
ILRT	Integrated Leak Rate Test
ISRS	In-Structure Response Spectra
LB	Lower Bound
LL	Live Load
LMP	Licensing Modernization Program
LMS	Lumped Mass Stick
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LOPP	Loss of Preferred Power
LR	Lower Realization
LS	Level Switch
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MAPE	Mean Annual Probability of Exceedance
MCA	Main Condenser and Auxiliaries
MCNP	Monte Carlo N-Particle
MCR	Main Control Room
MS	Main Steam

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Acronym	Explanation
NBC	National Building Code of Canada
NPP	Nuclear Power Plant
NSCA	Nuclear Safety and Control Act
NS-DBE	Non-Seismic Design Basis Earthquake
OBE	Operating Basis Earthquake
OLC	Operational Limits and Conditions
OPG	Ontario Power Generation
P&ID	Piping and Instrumentation Diagram
PAM	Post-Accident Monitoring
PBIRS	Performance Based Intermediate Response Spectra
PBSRS	Performance Based Surface Response Spectra
PCW	Plant Cooling Water System
PIE	Postulated Initiating Event
PLSA	Plant Services Area
PMF	Probable Maximum Flood
PRA	Probability Risk Assessment
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSHA	Probabilistic Seismic Hazard Assessment
QA	Quality Assurance
RAM	Reliability, Availability, and Maintainability
RB	Reactor Building
RBV	Reactor Building Vibration
RCS	Reactor Coolant System
RG	Regulatory Guide
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3

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Acronym	Explanation
SCCV	Steel-plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Seismic Design Category
SDE	Site Design Earthquake
SEI	Structural Engineering Institute
SIL	Safety Integrity level
SIR	Seismic Interface Restraint
SIT	Structural Integrity Test
SMAMP	Structures Monitoring and Aging Management Program
SMR	Small Modular Reactor
SPSA	Seismic Probabilistic Safety Assessment
SRA	Site Response Analysis
SRSS	Square-Root-of-the Sum of the Squares
SSC	Structures, Systems, and Components
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
TB	Turbine Building
TBD	To Be Determined
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TRACG	Transient Reactor Analysis Code General Electric
UB	Upper Bound
UHRS	Uniform Hazard Response Spectrum
UL	Underwriters Laboratory
UR	Upper Realization
USNRC	U.S. Nuclear Regulatory Commission
V/H	Vertical to Horizontal
ZPA	Zero Period Acceleration

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3.0 SAFETY OBJECTIVE AND DESIGN RULES FOR STRUCTURES, SYSTEMS AND COMPONENTS

This chapter introduces the safety objectives and the Safety Strategy to meet those objectives for the design and construction of the Boiling Water Reactor, 10th Design – 300 MWe (BWRX-300) Small Modular Reactor (SMR) facility at the Darlington site in Ontario, Canada.

Additionally, this chapter describes the methodology for classification of Structures, Systems, and Components (SSC), the design measures for protection against external and internal hazards, the general design aspects, and codes and standards applied to the BWRX-300 design to meet the requirements of the Nuclear Safety and Control Act (NSCA) and associated Canadian Nuclear Safety Commission (CNSC) Regulations and relevant Regulatory Documents.

3.1 General Safety Design Basis

The overall safety philosophy for the design of the BWRX-300 is referred to as the Safety Strategy. The objective of the Safety Strategy is to establish a design with a high level of safety. This is accomplished through incorporation of design requirements as set forth in CNSC REGDOC-2.5.2, (Reference 3.1-1) which to a large degree are based on the principles set forth in the International Atomic Energy Agency (IAEA) document SSR-2/1 (Reference 3.1-2).

The establishment of the BWRX-300 design basis is achieved through an iterative safety framework wherein the design is implemented to meet defined safety objectives and safety goals that are confirmed via deterministic and probabilistic safety analyses. Results of safety analyses then provide feedback into the design and the process is repeated as required until adequate design and regulatory safety margins are achieved.

3.1.1 Safety Objectives

In CNSC REGDOC-2.5.2 Section 4 (Reference 3.1-1), the CNSC endorses the safety objectives established by the IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles (Reference 3.1-3) which when followed ensure that reactor facilities are operated, and activities conducted to achieve the highest standards of safety that can be reasonably achieved. These safety objectives are described below:

General Nuclear Safety Objective: Reactor facilities are designed and operated in a manner that will protect individuals, society, and the environment from harm by establishing and maintaining effective defences against radiological hazards due to ionizing radiation. The general nuclear safety objective is supported by the following three complementary safety objectives:

1. **Radiation Protection Objective:** Radiation exposures within the reactor facility during normal operations, during anticipated operational occurrences or due to any planned release of radioactive material from the reactor facility are kept below prescribed limits and As Low As Reasonably Achievable (ALARA). Provisions are made for the mitigation of the radiological consequences of accidents.
2. **Technical Safety Objective:** All reasonably practicable measures are taken to prevent accidents in the reactor facility and to mitigate the consequences of events should they occur.
3. **Environmental Protection Objective:** All reasonably practicable mitigation measures to protect the environment during the operation of a reactor facility and to mitigate the consequences of an accident are provided. The design includes provisions to control, treat and monitor releases to the environment and minimize the generation of radioactive and hazardous wastes.

The high-level safety objectives inform the principal safety objectives in the design and safety analyses.

3.1.2 Radiation Protection and Radiological Acceptance Criteria

3.1.2.1 Radiation Protection

The BWRX-300 is designed to meet the Radiation Protection Objective by ensuring that potential radiation dose to workers and the public is kept below prescribed regulatory limits per the Radiation Protection Regulations (Reference 3.1-4) and ALARA.

This is achieved by a comprehensive and appropriately conservative source term derivation identifying radiation sources during the design phase to ensure means are provided to reduce occupational exposure during plant operation, maintenance, and decommissioning.

Safety features and measures include:

- Passive engineered safety features
- Active engineered safety features
- Administrative safety measures

Engineered safety features include shielding, containment, ventilation, remote handling, and interlocks. Administrative safety measures that reduce exposure to the hazard during planned operations include restrictions on occupancy, monitoring arrangements, pre-planning of exposure and the use of barriers and notices. Passive engineered safety measures (e.g., containment or shielding) are preferred before active engineered safety features and administrative safety measures. Human factors considerations are incorporated into the engineered and administrative measures (See Chapter 18 for details).

System design evaluations are performed in parallel with other activities to ensure systems support operational objectives. These evaluations include the development of reasonable and practical measures to achieve minimal dose to workers and the public.

Details on how radiation protection is considered in the design for operational states and accident conditions are provided in Chapter 12.

3.1.2.2 Radiological Acceptance Criteria

Limits on radiation dose are established by the CNSC through the Radiation Protection Regulations (Reference 3.1-4). The expectation established is that during normal operation, including maintenance and decommissioning, dose to workers and the public are ALARA.

Per CNSC Radiation Protection Regulations (Reference 3.1-4), the effective dose limit for a nuclear energy worker is an average of 20 mSv effective dose per year over a five-year period (100 mSv over five consecutive years), with no single year exceeding 50 mSv effective dose. The effective dose limit for a member of the public is 1 mSv per year from all sources of radiation other than natural background and medical exposures. Additional details are provided in Reference 3.1-4.

In addition to design features, administrative measures such as radiation protection and environmental protection programs are established to ensure worker and public dose is maintained below limits. Action levels are established for effluent releases and expressed in a form that compliance can be demonstrated in a practical manner. These action levels are not limiting but, are values at which actions must occur to reduce the effluent releases from the plant. Chapter 20 discusses Effluent Dose Levels to the General Public.

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Deterministic safety analyses are conducted in accordance with CNSC REGDOC-2.4.1 (Reference 3.1-5) to confirm that the BWRX-300 is designed to ensure that potential radiation doses to the public from Abnormal Operating Occurrences (AOOs) and Design Basis Accidents (DBAs) (defined in Subsection 3.1.3) do not exceed dose acceptance criteria per Section 4.2.1 of CNSC REGDOC-2.5.2 (Reference 3.1-1). In the deterministic safety analysis, the committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated for a period of 30 days after the analyzed event to confirm that for AOOs and DBAs, doses are less than or equal to the following:

- 0.5 millisievert (mSv) for any AOO or
- 20 mSv for any DBA

Chapter 15, Subsection 15.3.1, describes the dose calculation methodology used in the deterministic safety analysis. Results of the analyses are summarized in Section 15.7 demonstrating that the radiological consequences of the analyzed events do not exceed the acceptance criteria for AOOs and for DBAs.

3.1.2.3 Safety Goals

In addition to the deterministic dose acceptance criteria, Probabilistic Safety Analysis (PSA) is used to assess risks posed by reactor facility operation through the application of quantitative safety goals. These include core damage frequency, and small and large release frequency.

Core damage frequency is a measure of the capability of the design to prevent an accident that leads to core damage. Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of reactor facilities. The quantitative goals as established by CNSC REGDOC-2.5.2, Section 4.2.2 (Reference 3.1-1) are:

- **Core damage frequency** - The sum of frequencies of all fault sequences that can lead to significant core degradation shall be less than 1E-5 per reactor-year.
- **Small release frequency** - The sum of frequencies of all fault sequences that can lead to a release to the environment of more than 1E15 becquerels of Iodine-131, shall be less than 1E-5 per reactor-year.
- **Large release frequency** - The sum of frequencies of all fault sequences that can lead to a release to the environment of more than 1E14 becquerels of Cesium-137 shall be less than 1E-6 per reactor-year.

The PSA is described in detail in Chapter 15, Section 15.6, Probabilistic Safety Analyses.

3.1.3 Plant States Considered in the Design Basis

The range of conditions and events considered are categorized into plant states based on their frequency of occurrence. Plant states include operational states and accident conditions. Operational states included in the design basis are Normal Operation and AOOs. Accident conditions considered in the design basis are DBAs. Design Extension Conditions (DECs) are accident conditions considered in the design but are outside of the design basis based on their lower expected frequency of occurrence.

These four plant states considered in the BWRX-300 Safety Strategy as described below are consistent with CNSC REGDOC-2.5.2, Section 7.3 (Reference 3.1-1):

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- **Normal Operation** is operation within specified Operational Limits and Conditions (OLCs) (see Chapter 16) and includes the following Normal Plant Operational Modes: Power Operation, Startup, Hot Shutdown, Stable Shutdown, Cold Shutdown, and Refueling. (The normal plant operating modes are described in Chapter 16).
- **Anticipated Operational Occurrences** are deviations from normal operation that are expected to occur at least once during the operating lifetime of the reactor facility but that, with the appropriate design measures, do not cause any significant damage to safety class components, or lead to accident conditions.
- **Design Basis Accidents** are conditions for which a reactor facility is designed according to established design criteria, and for which damage to the fuel and the release of radioactive material are kept within regulated limits.
- **Design Extension Conditions** are postulated accident conditions that are less frequent than DBAs. DECs are a subset of beyond-design-basis accidents (BDBA), and are therefore, not part of the design basis. DECs are considered in the design process of the facility in accordance with best-estimate methodology DECs can occur without core damage or with core damage where releases of radioactive material are reasonably contained and kept within acceptable limits.

BDBAs other than DECs are accidents for which confinement of radioactive materials cannot be reasonably achieved. These are referred to as severe accidents and involve a catastrophic failure, core damage, and fission product release. A severe accident is generally considered to begin with the onset of core damage.

Representative DECs with core damage are postulated to provide inputs for the design of the containment and of the safety features ensuring containment functionality. This set of accidents is considered in the design of corresponding safety features for DECs and represents a set of bounding cases that envelope other severe accidents with more limited degradation of the core.

These accidents scenarios are considered for practical elimination as described in Subsection 3.1.8.

Events are assigned to a plant state based on the expected frequency of the fault sequence, which includes a Postulated Initiating Event (PIE) and, in some cases, additional failures of mitigating functions. As described in CNSC REGDOC-2.5.2, Section 7.4 (Reference 3.1-1), PIEs are the events that lead to deviations from normal operation. PIEs originate from operating errors, equipment failures, or internal or external hazard of natural or human origin.

Frequency ranges for plant states are:

- AOO (greater than 1E-02 per reactor-year)
- DBA (1E-02 to 1E-05 per reactor-year)
- DEC (less than 1E-05 per reactor-year)

The design requirements of SSC are developed to ensure that the plant is capable of meeting applicable requirements for each plant state. This is demonstrated through safety analyses as described in Chapter 15.

The facility is operated, monitored, and maintained within safe operating configurations or is transitioned to a safe operating configuration in accordance with operating procedures that are consistent with the design. (See Chapter 13, Section 13.4 for details.)

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Acceptance criteria are assigned to each plant state in the design, considering the principle that frequent fault sequences have only minor or no radiological consequences, and that any fault sequences that may result in severe consequences are of extremely low probability.

For normal operating modes, the OLCs serve as acceptance criteria as they are the set of limits and conditions within which the facility must be operated to ensure it is operated safely. OLCs are established as discussed in Chapter 16.

For each AOO and DBA fault sequence, acceptance criteria are defined and met to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. These acceptance criteria are discussed and summarized in Chapter 15, Section 15.3.

For DEC fault sequences, the safety objectives are to prevent significant core damage, mitigate accident consequences, and protect containment integrity. These objectives are demonstrated in PSA by showing that the plant meets the established safety goals (described in Subsection 3.1.2.3). (PSA is described in detail in Chapter 15, Section 15.6.) Also, it is demonstrated that procedures and equipment put in place to handle accident management needs are effective in responding to DEC. This is accomplished through the operating procedures described in Chapter 13 and through complementary design features described in Chapter 15, Appendix 15B.

The general approach to defining the design basis for the BWRX-300 involves establishing the plant states described above, identifying the PIEs leading to a deviation from normal operation and categorizing mitigating functions based on their ability to prevent and mitigate the progression of events ensuring that the safety objectives are met.

3.1.4 Prevention and Mitigation of Accidents

The design of the BWRX-300 includes provisions to prevent and to mitigate the consequences of accidents and to ensure that the likelihood that an accident will have harmful consequences is extremely low.

The primary means of preventing and mitigating the consequences of accidents is through the application of Defence-in-Depth (D-in-D). The application of D-in-D for the BWRX-300 design is described below in Subsection 3.1.6.

3.1.5 Fundamental Safety Functions

The design of the BWRX-300 fulfills Fundamental Safety Functions (FSFs) at all plant states (defined in Section 3.1.3) which ensures the design meets the safety objectives consistent with CNSC REGDOC-2.5.2, Section 6.2 (Reference 3.1-1). The FSFs for the BWRX-300 are:

- Control of reactivity
- Removal of heat from the fuel (in the reactor, during fuel storage and handling, and including long-term heat removal)
- Confinement of radioactive materials, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental releases

The FSF prevent or mitigate radiological releases by ensuring the physical barriers to releases (fuel matrix, fuel cladding, Reactor Coolant Pressure Boundary (RCPB), and containment) remain effective. In addition to the protection of barriers, a means of monitoring the status of key plant parameters is provided for ensuring that the FSF are fulfilled. From this perspective, the monitoring function is treated as inherent to the design of the FSF. Other considerations for the monitoring function are as follows:

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1. If a manual operator action plays a role in performing an FSF, the monitoring function of the equipment used to display key plant parameters that are necessary for the operator to perform the manual action successfully are also considered part of the FSF.
2. Certain monitoring functions allow the operator to confirm ongoing effectiveness of the FSFs during all plant states, to implement post-accident procedures, and to make decisions in support of emergency planning.
3. Post-Accident Monitoring (PAM) is important for operator decision making such as taking manual actions and implementing functions. Therefore, the designation, treatment and display of certain plant parameters or measurements as post-accident monitoring variables is a supporting design feature.
4. A minimum set of monitoring functions and display of parameters that do not support the operator actions are provided to support accident assessment.

Preservation of the FSFs is intrinsic to BWRX-300 Safety Strategy. A systematic approach is taken to identify the FSFs and those SSC necessary to fulfill the FSFs following a PIE. This systematic approach is detailed in the D-in-D discussion below.

3.1.6 Defence-in-Depth

3.1.6.1 BWRX-300 Defence-in-Depth Concept

The implementation of D-in-D in the BWRX-300 design is the basis for the Safety Strategy for ensuring an adequate level of safety is achieved by the design.

The concept of D-in-D (consistent with CNSC REGDOC-2.5.2, Section 6.1 (Reference 3.1-1)) involves the provision of multiple layers of defence against some undesirable outcome rather than a single, strong defensive layer. In the case of a nuclear power plant, the undesirable outcome is the exposure of workers, the public or the environment to radioactivity exceeding levels determined to be safe.

There are two types of defensive layering considered:

1. Physical barriers in place to prevent the release of radioactivity: The fuel matrix, fuel cladding, RCPB, and containment. The integrity of one or more physical barriers must be maintained to prevent unacceptable releases.
2. A combination of active, passive, and inherent safety features used to minimize challenges to the physical barriers, to maintain the integrity of the barriers and, in case a barrier is breached, to ensure the integrity of the remaining barriers.

While the physical barriers themselves represent multiple layers of defence against radioactive releases, in the BWRX-300 D-in-D application, the physical barriers are not themselves referred to as “defense lines”. That term is reserved for the layers of defence comprising features, functions and practices that protect the integrity of the barriers. The D-in-D concept applied is largely focused on identifying and organizing features, functions, and practices into defense lines without explicit acknowledgment of the physical barriers. The fundamental purpose of the defense lines is to ensure the integrity of the physical barriers by applying multiple levels of protection.

The BWRX-300 D-in-D concept uses the FSFs described above to define the interface between the defense lines and the physical barriers. In a given plant scenario, if the FSFs are performed successfully, then the corresponding physical barriers remain effective.

3.1.6.2 Defense lines

Five Defense Lines (DLs) (or levels), DL1 through DL5, are adopted consistent with CNSC REGDOC-2.5.2, Section 6.1 (Reference 3.1-1) and IAEA SSR-2/1 (Reference 3.1-2). Figure 3.1-1 illustrates the defense lines as they correspond to the plant states.

The first defense line (DL1) does not include plant functions. It minimizes potential for PIEs to occur in the first place and minimizes potential for failures to occur in subsequent defense lines by assuring high quality and conservatism in design, construction, and operation. The second, third, and fourth defense lines (DL2, DL3, and DL4) comprise plant functions that act to prevent PIEs from leading to significant radioactive releases. The fifth defense line (DL5) involves off-site emergency preparedness to protect the public in case a substantial radioactive release occurs.

The defense lines include measures such as engineering and operational practices, plant features, and plant functions. These measures are incorporated such that:

- The normal operation of the plant is monitored and controlled such that PIEs that lead to AOOs can be mitigated before evolving into DBAs
- The consequences are limited if a DBA does develop
- Multiple defense lines are capable of independently performing the FSFs. While this means that more than one DL is capable of independently performing the FSFs for D-in-D, DL independence from all other DLs is based on how specific DLs are credited for specific fault sequences.

Table 3.1-1 provides a high-level description of the objective, and the design means and operational means for supporting the defense lines. The following is a brief description of each of the defense lines.

Defense Line 1 (DL1)

The purpose of the first level of defence is to prevent deviations from normal operation and the failure of important SSC. This is achieved through the quality measures taken to minimize potential for failures and for initiating events to occur in the first place and to minimize potential for failures to occur in subsequent lines of defence. These quality measures cover the design, construction, inspections, operation, use of operational experience, periodic safety reviews, and maintenance, and testing of the plant.

DL1 measures may support the basis for assumptions made in safety analyses. For example, the use of a high-quality design process and stringent equipment qualification for the most important components support the assumption that only a single failure is considered in the Conservative Deterministic Safety Analysis discussed in Chapter 15, Subsection 15.2.1.

Examples of DL1 measures include:

- The clear definition of normal and abnormal operating conditions
- Maintenance and implementation of a quality assurance program consistent with nuclear regulations and industry standards
- Application of appropriate industry standards to the design of SSC
- Adequate design margins
- Robust design processes including design verifications
- Comprehensive testing programs

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- Provisions for adequate time for operators to respond to events and appropriate human-machine interfaces, including operator aids, to reduce the burden on the operators
- Deterministic safety analyses including appropriate conservatism, supplemented by Probabilistic Safety Analysis to produce risk insights
- Categorization and qualification of SSC according to their safety significance
- Operational Limits and conditions
- Application of lessons learned through operating experience

Defense Line 2 (DL2)

The purpose of the second level of defence is to detect and control deviations from normal operational states to prevent AOOs from escalating to accident conditions. Functions that normally operate to maintain key reactor parameters (e.g., pressure, reactor level, and reactivity) within normal ranges are part of DL2.

Examples of DL2 measures include:

- Anticipatory plant trips
- Maintain target power
- Maintain target level
- Maintain target pressure
- Control Rod Block

Defense Line 3 (DL3)

For the third level of defence, it is assumed that, although very unlikely, the escalation of certain AOO or DBA PIEs might not be controlled at a preceding level and that an accident could develop. In the design of the plant, such accidents are postulated to occur. DL3 contains plant functions that act to mitigate a PIE by preventing fuel damage, when possible, which assures the integrity of the release barriers are maintained, and the plant is maintained in a safe state until normal operations are resumed.

The systems and equipment involved in performance of DL3 functions are designed for high reliability. Examples include eliminating the need for active support systems such as power supplies, ventilation, or cooling water, and minimizing the need for active control functions such as pumps and actively controlled valves.

The DL3 functions and equipment performing those functions are subject to functional and design requirements derived from the Conservative Deterministic Safety Analysis as described in Chapter 15, Subsection 15.2.1.

Examples of DL3 measures include:

- Reactor Scram
- Isolation Condenser Initiation
- Main Steam isolation
- Containment Isolation
- Reactor Pressure Vessel (RPV) Isolation

Defense Line 4 (DL4)

The purpose of the fourth level of defence is to mitigate DEC's.

For the BWRX-300, DL4 is comprised of two subsets of functions that are designated as DL4a and DL4b functions. DL4a functions mitigate DEC's that occur without core damage. DEC's progressing to core damage are mitigated by DL4b functions.

DL4a

DL4a functions are those that place and maintain the plant in a safe state in scenarios involving:

- DBAs sequences combined with multiple failures that prevent the DL3 SSC from performing their intended function (i.e., Common Cause Failure (CCF) which is a failure of two or more SSC due to a single specific event or cause.)
- DEC PIEs considered as credible events that may involve multiple failures causing the loss of a FSF to be fulfilled as part of normal operation

Examples of DL4a measures include:

- Diverse means of achieving the FSFs that are independent of and diverse from the SSC carrying out the DL3 functions that are presumed to have failed.
- Scrams initiated by the Diverse Protection System

DL4b

DL4b includes:

- Functions provided in scenarios leading to core damage to limit the radiological releases in case of core damage and are aimed at maintaining the containment functions for extreme events, multiple events, or multiple failures that defeat DL2, DL3, and DL4a.
- Functions provided to mitigate the effects from a damaged core and to preserve the FSF of confinement of radioactive material while limiting radioactive releases to acceptable levels.
- Safety features designated for DEC's with core damage may, if practicable and available, also be used for preventing or minimizing significant core damage if it can be demonstrated that such use will not undermine the ability of these systems to perform their primary functions if conditions evolve into a severe accident.

Examples of DL4b measures include:

- DL4b measures carried out by complementary design features such as diverse and flexible equipment and portable components such as, portable uninterruptible power supplies and portable pumps
- Containment venting and overpressure protection
- Boron injection

A list of complementary design features is provided in Chapter 15, Appendix 15B.

Defense Line 5 (DL5)

The purpose of the fifth and final level of defence is to mitigate the radiological consequences of radioactive releases that could potentially result from accidents.

DL5 includes emergency preparedness measures to cope with potential unacceptable releases in case the first four defense lines are not effective. These are largely off-site measures taken to protect the public in a scenario involving substantial release of radiation.

Examples of DL5 measures:

- Severe accident management procedures
- Emergency response procedures and equipment (peripheral systems such as meteorological monitoring)
- On/off-site emergency response facilities, and certain communication systems may play a role in DL5. Chapter 19 discusses emergency response arrangements such as procedures and facilities. Communication systems are discussed in Chapter 9A, Section 9A.9.1. (Note that these measures may be initiated earlier in an event prior to progression to a severe accident)

3.1.6.3 Defense Line Independence

The BWRX-300 design incorporates independence in the application of D-in-D. Defense lines that mitigate the same event are independent as far as is practicable to avoid the failure of one level reducing the effectiveness of other levels. Some examples include:

1. Among DL2, DL3 and DL4a, at least one defense line can mitigate a PIE caused by or concurrent with a CCF in another defense line, with the mitigation means being independent from the effects of the initiating CCF.
2. All PIEs with a frequency greater than $1E-05$ caused by a single failure can be mitigated by DL3 and independently by DL2, DL4a, or a combination of DL2 and DL4a functions that are unaffected by the PIE. To the extent practicable, DL3 functions are independent and diverse from those in DL2 and from those in DL4a. This is because DL3 functions provide a backup to DL2 functions, and DL4a functions provide a backup to DL3 functions but DL4a functions are not needed to provide a direct backup to DL2 functions to maintain D-in-D for the same event.
3. The DL4b functions intended for mitigating DECs are functionally and physically separated from the systems intended for other DL functions.
4. DL4b features specifically designed to mitigate the consequences of accidents with core damage are independent from systems used in normal operation or used to mitigate AOOs as far as is practicable and with exceptions justified.
5. Exceptions to rules of independence are described, assessed, and justified. If equipment supports functions in more than one defense line, there is an increased focus on their reliability in the application of DL1 compared to a design feature credited in only one defense line.

3.1.6.4 Safety Strategy Process for Implementing Defence-in-Depth

The BWRX-300 Safety Strategy implements the D-in-D concept into the design through evaluations and analyses as shown in Figure 3.1-2. These include:

- Hazard Evaluations
- Fault Evaluation
- Deterministic Safety Analyses
- PSA

The elements of Figure 3.1-2 are briefly described below.

3.1.6.4.1 Hazard Evaluations

The first step is to identify PIEs using a systematic methodology considering both direct and indirect events through hazard evaluations. The BWRX-300 Safety Strategy includes the following four types of hazard evaluations which are summarized in Chapter 15, Subsection 15.1.3:

- Functional Failure Hazard Evaluation – assessment of failures of SSC
- External Hazard Evaluation - assessment of external events such as earthquakes or aircraft crashes that have the potential to impact plant safety
- Internal Hazard Evaluation – assessment of hazards originating within the facility such as missiles from rotating equipment, fires, collapse of structures
- Human Operation Hazard Evaluation – human errors which could reasonably be expected to occur based on industry operating experience

The output of the four hazard evaluations are the potential PIEs for consideration in the Fault Evaluation.

3.1.6.4.2 Fault Evaluation

The Fault Evaluation process evaluates the PIEs determined as a result of the hazard analyses. PIEs are selected and organized along with fault sequences. As used herein, a fault is essentially a failure or a hazard and could be the initiator for or result from a PIE. A PIE is an event that initiates a fault sequence. A fault sequence consists of a PIE, and responses by mitigation functions (including both failed responses and successful responses). This is consistent with the description of event combinations per CNSC REGDOC-2.4.1, Section 4.2.2.5 (Reference 3.1-5).

The Fault Evaluation establishes traceability between the plant design and the safety analysis bases. The Fault Evaluation process including the selection and categorization of PIEs and fault sequences for deterministic safety analysis is described in Chapter 15, Section 15.2.

3.1.6.4.3 Deterministic Safety Analyses

The objective of deterministic safety analysis for nuclear power plants is to confirm that:

- FSFs can be performed
- SSC performing the FSF are designed with adequate margins
- physical barriers to radioactive release maintain their integrity as required

Deterministic safety analysis is supplemented by insights obtained from fabrication, testing, inspection, operating experience, and PSA. It demonstrates that the source term and the potential radiological consequences of different plant states are acceptable. It also demonstrates that the possibility of certain conditions arising that could lead to an early or a large radioactive release can be considered as 'practically eliminated'.

The output of the Fault Evaluation process which includes the selection of PIEs and fault sequences organized by frequency are analyzed in deterministic safety analysis. Chapter 15, Subsection 15.2.1, provides more detail on the deterministic safety analysis process.

3.1.6.4.4 Probabilistic Safety Analyses

PSA are performed to understand the overall risk presented by the facility and to allow comparisons to be made against safety goals (defined in Section 3.1.2.3) They also provide

essential understanding of strengths and weaknesses of a design with complex systems and interdependencies. They are used for evaluating complementary design feature concepts or changes in operating conditions and have many other applications to enhance safety decision

To supplement quantitative PSA results, a severe accident analysis is performed to understand the complex physical phenomena associated with a reactor core damage scenario. This analysis supports confirmation that the radioactive release sequences modeled in the Level 2 PSA adequately reflect associated phenomena.

Severe accident analyses are used to complement the design deterministic safety and PSA in situations where the consequence is large, even if the calculated risks are low and/or the deterministic safety analysis provides a robust demonstration of fault tolerance. The severe accident analysis is not considered standalone piece of analysis deriving scenarios from first principles, but instead builds upon other types of analysis to create an overall safety case that is adequate in its coverage.

Detailed discussion of PSA and Severe Accident Analysis is provided in Chapter 15, Section 15.6.

3.1.7 Application of General Design Requirements and Technical Acceptance Criteria

3.1.7.1 Deterministic Design Principles in Codes & Standards

A fundamental aspect of the BWRX-300 Safety Strategy is that the overall plant design applies good engineering practices for design, construction, operation, maintenance, and testing which relates to conformance to regulatory requirements, as well as industry codes and standards and norms for achieving high dependability in performance.

Engineering design rules are established and applied, as appropriate by the specific design discipline based on relevant codes, standards, and proven engineering practices.

Because codes or standards for the different design disciplines (e.g., mechanical, civil, and electrical) are not always based on compatible safety criteria, consistent acceptance criteria are established, and good engineering practices are used, to provide consistency in the application of selected codes and standards in design. Analyses and evaluation of the codes and standards to be applied in the design, fabrication and construction of the plant is performed. The results of this analysis and evaluation are documented as part of the management system.

The plant architecture and systems design specifications demonstrate that the plant and the SSC are designed, implemented, constructed, installed, operated, and maintained safely with respect to their application and maintenance of these guiding fundamental design principles that follow. Additionally, changes are performed using the same guiding fundamental design principles, using the same or better methods and processes to avoid compromising safety.

3.1.7.2 Minimize Probability of Failure Structures, Systems, and Components

The probability of failure of systems and equipment is minimized through a design which provides predictable and repeatable performance of the FSFs. This is achieved by deploying highly reliable and dependable SSC.

DL3 systems and equipment are designed to fail to a safe state or to a known, defined state to ensure safety is not jeopardized. Thus, reactor trip systems fail to the safe state, but engineered safety features systems may fail-safe or are non-actuated (e.g., isolation condenser cooling function). Fail-safe design is achieved through systematic identification of failure modes through Failure Modes and Effects Analyses (FMEA).

Systems are required to be testable to provide assurance of continued operability and availability when required. System maintainability is a fundamental aspect of the design, extending down to software by ensuring documented, well-designed, understandable code.

Chapter 13 describes how fitness for service is addressed in established programs that include: Reliability, Maintenance, Aging Management, Chemistry Control, Periodic and In-Service Inspections. Programmatic requirements addressing fitness for service span the full life cycle of the facility beginning with inclusion in facility design decision making.

3.1.7.3 Independence

The most plausible reason for the failure of FSFs is the occurrence of dependent failures. Dependent failures are identified, and where practicable, measures are implemented in design, construction, and operation to eliminate the dependencies or reduce their potential effect. The application of independence is used in the Safety Strategy to enhance reliability and reduce potential for dependent failures. Independence is an essential aspect of effectiveness in the implementation of D-in-D.

The determination of independence of SSC required to mitigate the consequences of a single or a likely combination of internal or external hazards on the plant is conducted through the Fault Evaluation introduced in Section 3.1.6.4.2 and described in more detail in Chapter 15, Section 15.2 and confirmed via the PSA in Chapter 15, Section 15.6.

The PSA is also used to confirm the adequacy of the independence measures.

Independence is achieved by addressing the main causes of CCFs: functional, spatial, inherent, and human error dependencies as discussed in Subsection 3.1.7.5.

3.1.7.4 Diversity

Diversity is the provision of dissimilar means of achieving the same objective. Diversity involves the use of design features which differ in the physical means of achieving a specific objective or use of different equipment made by different manufacturers. Diversity is achieved by incorporating different attributes into the systems or components. Such attributes could be different principles of operation, different physical variables, different conditions of operation, or production by different manufacturers, for example. It is necessary to ensure that the diversity attribute achieves the desired increase in reliability in the as-built design. For example, to reduce the potential for CCFs the designer should examine the application of diversity for any similarity in materials, components and manufacturing processes, or subtle similarities in operating principles or common support features. If diverse systems or components are used, there is a consideration that reasonable assurance that such additions are of overall benefit, including consideration of the associated disadvantages such as the increased operational complication, additional maintenance and test procedures, and the potential for lower reliability.

Diversity is considered for digital equipment and active mechanical/electrical equipment. Diversity is not included for passive equipment such as pipes and tanks. Diversity is a DL1 provision used to strengthen subsequent defense lines.

3.1.7.5 Separation

Functional isolation is used to reduce the likelihood of adverse interactions between equipment and components resulting from normal or abnormal operation or failure of any component in the systems. For example, in a power supply, functional isolation is commonly achieved using fuses and circuit breakers.

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Separation supports defense line function independence discussed in Subsection 3.1.6.3. System layout and design uses physical separation to increase assurance that independence will be achieved, to preclude certain CCFs.

- Physical separation includes separation by geometry (such as distance or orientation); barriers; or a combination of these. The choice of the means of separation will depend on the PIEs considered in the design basis, such as the effects of fire, chemical explosion, aircraft crash, missile impact, flooding, extreme temperature, or humidity.
- In a redundant system and despite diverse provisions, the threat of CCFs from hazards such as fire may be reduced by system segregation. Segregation is the separation of components by distance or physical barriers. An example is the use of fire barriers to indicate individual fire zones, which may also serve as barriers to other hazards.
- Plant barriers that provide protection against certain faults or hazards are assessed to ensure that the barriers remain operable and accessible in the event of those faults or hazards occurring. This is particularly important where SSC that perform defense line functions are co-located with other plant equipment that do not.

3.1.7.6 Redundancy

Redundancy is the provision of more than the minimum number of nominally identical equipment items required to perform a specific safety function. Such redundant provisions allow a safety function to be satisfied when one or more systems or components (but not all) are unavailable, due to a variety of unspecified potential failure mechanisms or maintenance (e.g., identified faults or hazards). Redundancy enables failure or unavailability of at least one set of systems or components without loss of the function. For example, three or four pumps may be provided for a particular function when any two would be capable of carrying it out. For the purposes of redundancy, identical or diverse components may be used.

The application of independence, diversity, separation, and redundancy in the design is described in each system design description.

3.1.7.7 Single Failure Criterion

The BWRX-300 design addresses the single failure criterion through design and safety analyses to ensure reliability of DL3 functions. Consistent with CNSC REGDOC-2.5.2, Section 7.6.2, each safety group (DL3 function) is assessed for capability in fulfilling its required function even if a failure of a single component occurs within this group.

A single failure is one which results in the loss of capability of a single system or component to perform its intended DL3 function(s), and any consequential failure(s) which result from it.

For the BWRX-300, the single failure criterion is considered in two ways:

1. As a design attribute that is typically achieved through redundancy in the system architecture of the SSC carrying out DL3 functions. This involves a systematic search for potential single failure points and their effects on prescribed missions (i.e., FMEA).
2. As an assumption made in the conservative deterministic safety analysis, in addition to the PIE and any additional failures, all identifiable undetectable faults are included to demonstrate a high degree of confidence that acceptance criteria will be met.

During the design process, systems that are designed to carry out a DL3 function must be capable of carrying out their mission despite the failure of any single component within the system or in an associated system that supports its operation. Design measures for ensuring high reliability

of SSC carrying out DL3 functions include incorporating, independence, diversity, and redundancy, and also through the incorporation of passive and fail-safe features.

The PSA is used for identifying single failures for consideration in the deterministic safety analysis and is also a complementary means of demonstrating the insensitivity to single failures.

3.1.7.8 Common Cause Failures

3.1.7.8.1 Background Information and General Approach

CCFs are functional failures of multiple components due to a single specific event or cause. Such failures may affect several different safety class components simultaneously or may affect multiple components of the same type at the same time.

The event or cause of CCFs may be a design deficiency, a manufacturing deficiency, an operating or maintenance error, a natural phenomenon, a human induced event or an unintended cascading effect from any other operation or failure within the plant. Appropriate measures to minimize the effects of CCFs, such as the application of redundancy, diversity, and independence, are taken as far as practicable in the design.

Multiple failures can occur due to common weaknesses or dependencies shared by components. Such failures can cause failure of all redundant components in a single protection system or failure of components in more than one system. Dependent failures can considerably reduce the reliability of the protection systems relative to that expected from consideration of random failure mechanisms occurring in isolation. Identification of dependent failures is assessment by Functional Failure Hazards Evaluations.

The main types of failure dependencies that can cause a potential loss of safety function are as follows:

- **Functional Dependencies**, which arise from shared or common functional features such as a common electrical power source, a common cooling water system or a shared process fluid.
- **Spatial Dependencies**, which arise from physical features shared by components located in a common location such as common radiation or chemical conditions, a common environment and common support structures, and vulnerability to leaks of dangerous fluids (high temperature, corrosive or toxic).
- **Inherent Dependencies**, which arise from shared characteristics such as a common principle of operation or technical embodiment and a common failure mechanism such as mechanical overload or overpressure.
- **Human Error Related Dependencies**, which arise from human errors affecting some shared or common human process such as human error in design or manufacture, or operating staff error during operation and maintenance.

The general protective approach used for addressing postulated vulnerabilities to CCFs is diversity in the design. Dissimilarities in technology, function, implementation, and so forth, can mitigate the potential for common faults. The diversity approach to ensuring safety uses different (e.g., dissimilar) means to accomplish the same or equivalent function to compensate for a CCF that disables one or more levels of defence. Diversity is complementary to the principle of defence-in-depth, and it increases the chances that a defense line function will be available when needed. Different defense lines that mitigate the same event are diverse from each other to the extent practicable.

Another means of protecting against CCF is through feedback from operating experience that could identify weaknesses in the design, construction, operation and testing of equipment. In addition, conducting periodic inspection, surveillance, and testing provides opportunities to detect degradation or common causes before failures of SSC. Quality assurance and quality control measures applied to SSC commensurate with their importance help reduce preclude potential CCFs.

3.1.7.8.2 Common Cause Failures of Digital Instrumentation and Control Software

The BWRX-300 approach to assessment of CCF of Digital Instrumentation and Control (I&C) software is through a consequence-based approach.

Even when functional dependencies are addressed through rigorous design and application of codes and standards, operating experience shows that software CCFs occur. Validating assumptions and modeling of software CCF modes can be challenging due to uncertainty as each Digital I&C system is unique, and extrapolation of failure data from one system to another may not be meaningful making the identification of failure scenarios difficult. Analyzing each postulated CCF scenario is not practicable; therefore, using a consequence-based approach can limit the number of CCF scenarios is considered. This approach considers the radiological or dose consequences that could result due to CCFs in the software.

3.1.7.8.3 Defense Line Approach to Common Cause Failure

A multi-pronged approach and the systematic integration of CCFs in defense line functions, both as PIEs and as failures affecting fault sequence mitigation, are applied in deterministic safety analyses for prevention and mitigation in the D-in-D approach. Examples include:

1. DL3 systems and functions are designed and rigorously qualified to be resistant to the effects of environments that could cause common failures, including DBA environments.
2. For internal and external events resulting in DECAs, the design includes independent and diverse system functions to cope with the effects of common cause failure (e.g., DL4a).
3. Diverse accident monitoring instrumentation for severe accident management (e.g., DL4b) is provided.

The defence-in-depth approach is designed to include analyses of a reasonable set of CCF scenarios to provide assurance that the plant is protected against CCF phenomena. This approach is implemented using a set of CCF application guidelines to define the CCF modes that are included, how the failure modes are applied, and which assumptions can be made regarding equipment operability.

3.1.7.9 Other Approaches for Ensuring Safety

In addition to the design principles discussed above, the BWRX-300 design incorporates the following approaches to ensure safety.

3.1.7.9.1 Simplicity in Design

An implicit approach to reliability is to deploy the design with minimal complexity, with the knowledge that complexity may be required to enhance reliability or reduce the potential for human error. Where complexity is required (e.g., self-diagnostics, redundancy within the equipment in a single division), the complexity is documented and justified as necessary and appropriate for enhancing reliability, surveillance, calibration, and other required system or equipment attributes. There are tradeoffs in complexity, such as increasing the complexity by designing the system to reduce the human actions necessary for surveillance which also decreases the potential for human error, which enhances system reliability.

The BWRX-300 is specifically designed to enhance safety through simplification and reducing its dependence on human intervention. This is achieved through increasing its reliance on natural circulation and natural phenomena-driven safety systems (these are passive features as discussed below). These safety enhancements, in combination with its reduction in scale and complexity including a reduction in total number of active SSC, simplifies operations and maintenance. Some of the simplified design features are described in Chapter 1.

3.1.7.9.2 *Passive Safety Features*

The design of the BWRX-300 uses passive functions that do not require external sources of power or operator actions. DL3 functions are passive to the extent that is practicable and, therefore, have significantly less reliance on supporting systems or operator actions.

Examples of the BWRX-300 passive design features include:

1. Safety Class 1 batteries are capable of powering loads for a minimum of 72 hours. The design ensures that plant safety is maintained even after battery depletion.
2. BWRX-300 utilizes natural circulation and passive natural circulation for fuel cooling and passive containment heat removal. The plant is designed with the capability to cope with decay heat for seven days using only installed systems with no reliance on significant operator actions or external resources.

The mitigation of loss-of-coolant accidents is built on utilization of inherent margins (e.g., larger water inventory) to eliminate system challenges, reduced number, and size of RPV nozzles as compared to predecessor designs, and elimination of fluid system nozzles located below a level well above the top of active fuel to conserve inventory. The relatively large reactor pressure volume of the relatively tall chimney region provides a substantial reservoir of water above the core. This ensures the core remains covered following fault sequences involving feedwater flow interruptions or loss-of-coolant accidents without the need for active components (such as pumps). Additionally, the RPV is equipped with isolation valves attached directly to the reactor vessel for large bore piping systems to preserve reactor coolant inventory ensuring that adequate core cooling is maintained.

The application of these design concepts is described in each system design description.

3.1.7.10 Technical Acceptance Criteria

To meet the radiological acceptance criteria, derived accepted criteria are defined for the fuel pellet, fuel cladding, RCPB and containment. Deterministic safety analyses are performed to demonstrate that these criteria have been met. A description of acceptance criteria is provided in Chapter 15, Section 15.3. Details of the deterministic safety analysis are presented in Chapter 15 Section 15.3. Table 15.3-1 for AOOs and 15.3-2 for DBAs.

3.1.8 Practical Elimination

Consistent with CNSC REGDOC-2.5.2 Section 7.3.4 (Reference 3.1-1) and IAEA SSR-2/1(Reference 3.1-5), the BWRX-300 design is such that fault sequences that could lead to an early or large radioactive release are practically eliminated.

The definition of early and large radioactive release (from IAEA SSR-2/1) (Reference 3.1-5) in this context are:

1. An early radioactive release is a release of radioactive material for which off-site protective actions would be necessary but would be unlikely to be fully effective in due time.

2. A large radioactive release is a release of radioactive material for which off-site protective actions that are limited in terms of lengths of time and areas of application would be insufficient for the protection of people and of the environment.

Fault sequences with early or large releases could be considered to have been practically eliminated if either of the following apply:

- It is physically impossible for the accident sequence to occur.
- The fault sequence can be considered with a high degree of confidence to be extremely unlikely to arise.

Practical elimination is considered to refer only to those fault sequences leading to or involving core damage (e.g., a severe accident) for which the confinement of radioactive materials cannot be reasonably achieved.

The aim of the practical elimination concept is to reinforce D-in-D by focused analysis of those conditions having the potential for early radioactive release or a large radioactive release.

The justification of practical elimination preferably relies on a demonstration of physical impossibility for the accident sequence to occur. If this is not achievable, a demonstration of an extremely low likelihood of occurrence with a high level of confidence is provided. Sufficiently robust arguments and evidence are used to demonstrate the reliability of the lines of defence. If additional features are identified that prevent accidents or mitigation accident consequences, these features are considered for implementation as far as practicable.

The set of individual fault sequences that might lead to an early radioactive release or a large radioactive release are grouped to form a limited number of bounding cases or type of accident conditions.

Severe accident phenomena based on operating experience with predecessor advanced light water reactors serve as a starting point for consideration for practical elimination. Analyses demonstrating practical elimination are described in Chapter 15, Appendix 15A.

3.1.9 Safety Margins and Avoidance of Cliff-Edge Effects

A cliff-edge effect is described as a small change of conditions that may lead to a significant increase in the severity of consequences per CNSC REGDOC-3.6 (Reference 3.1-7).

In the BWRX-300 Safety Strategy, the principle of multiple physical barriers to the release of radioactive material and protection of those barriers is incorporated in the design as a DL1 measure. Margins are incorporated into the design of the physical barriers to demonstrate their capability in postulated scenarios that are more severe (by a small amount) than those in the design basis without incurring cliff-edge effects.

Conservative safety margins and sensitivity analyses are applied in safety analyses to account for assumptions and uncertainties. Additional details on the application of safety margins in Deterministic Safety Analysis are described in Chapter 15, Subsection 15.5.1.1. As part of the PSA, sensitivity and uncertainty analysis is conducted to demonstrate consideration of potential cliff-edge effects. (See Chapter 15, Subsection 15.6.1).

3.1.10 Design Approaches for the Reactor Core and for Fuel Storage

3.1.10.1 Design Approach for Reactor Core

The reactor core is designed to maintain the integrity of the fuel and the fuel cladding. The fundamental safety functions of control of reactivity, removal of heat from the reactor and fuel, and confinement of radioactive materials are inherent design features for the reactor core.

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The reactor core, the fuel, and fuel assemblies, including fuel channels and control blades, are designed such that the reactor can be shut down, cooled, and held subcritical with adequate margin in operational states, DBAs, and DECAs. Reactivity control ensures shutdown margin for shutdown states and any credible changes in core configuration. The design ensures that the fission chain reaction is controlled during operational states. The design limits positive reactivity through inherent neutronic and thermal-hydraulic characteristics, means of shutdown, and control to protect the reactor pressure boundary and prevent fuel damage.

The reactor core (including associated structures and cooling systems) is designed to withstand static and dynamic loading and vibration, to be compatible with expected chemicals, and to meet thermal material and radiation damage limits.

The reactor core design also provides for certain operator actions in accident scenarios to maintain the reactor in a shutdown condition, such as actions that might be addressed in emergency operating procedures or severe accident management guidelines.

3.1.10.2 Design Approach for Fuel Handling and Storage

The design of fuel handling and storage systems is consistent with the D-in-D approach applied to the reactor core with slightly different fundamental safety functions.

The design approach is to identify fuel handling and storage SSC that are necessary to fulfill the following fundamental safety functions for all plant states:

- Maintaining subcriticality of the fuel
- Removal of the decay heat from irradiated fuel
- Confinement of radioactive material, shielding against radiation as well as limitation of accidental radioactive releases

The Safety Strategy principle for fuel handling and storage is to leverage design and safety features in relation to fuel handling and storage that have been proven either in predecessor BWR applications or are based on operating experience.

Subcriticality is maintained by preventing criticality through use of geometrically safe configurations. The design of fuel storage systems considers the use of physical means or physical processes to increase subcriticality margins in normal operation to avoid reaching criticality during PIEs, including those PIEs arising from the effects of internal hazards and external hazards.

Fuel handling and storage systems are designed to maintain adequate fuel cooling capabilities for irradiated fuel ensuring that the fuel cladding temperature limits and/or the coolant temperature limits, as defined for operational states and accident conditions, are not exceeded.

The fuel storage and handling, radioactive waste, and other systems that may contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions. These systems are designed:

- With a capability to permit appropriate periodic inspection and testing of components safety features,
- With suitable shielding for radiation protection,
- With appropriate containment, confinement, and filtering systems,

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- With a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and
- To prevent significant reduction in fuel storage coolant inventory under accident conditions.

Appropriate systems are provided in fuel storage and radioactive waste systems and associated handling areas:

- To detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and
- To initiate appropriate safety actions

Refer to Chapter 9A, Section 9A.1 for a detailed description of the Fuel Handling and Storage Systems.

3.1.11 Considerations of Interactions Between Multiple Units

Operating experience has demonstrated that interactions or shared equipment between multiple units can cause problems for the plant and for personnel. Lessons learned include:

- Significant interactions between multiple co-located radiological sources (e.g., reactor units, spent fuel pools, or dry fuel storage facilities) could result due to concurrent or consequential initiators.
- The timing of concurrent accident sequences involving multiple radiological sources on a site can challenge shared SSC, as well as resources available for severe accident management and emergency response to the event.

Site evaluations would address multiple reactors or other co-located facilities and determine if these need to be treated as external hazards (e.g., external radiation sources) in the design of the BWRX-300. See Chapter 2, Subsection 2.2.5 for more details.

Each BWRX-300 unit would have its own safety class systems and its own safety features for DEC's.

If multiple units are to be co-located, emergency planning and design and safety analyses, including consideration of CCFs in similarly design units, would demonstrate that sharing resources of equipment and personnel, including temporary equipment and emergency response personnel, would not be detrimental to plant operation, fuel storage, emergency planning, or accident management.

3.1.12 Design Considerations for Aging Management

Aging of SSC is considered in the basic assumptions and in the input data to the safety, thermohydraulic and stress analyses. All system and component design specifications reference design requirements on aging, including those in the applicable codes and standards.

Aging and equipment qualification considerations are important aspects, complementary to each other in plant design. Equipment qualification is discussed in Section 3.9.

In designing components, system designers consider aging mechanisms and their effects on the safety, reliability, and performance of SSC for those that are well known and understood. Additionally, system designers collect information from operations feedback, research and development, vendor recommendations, maintenance and operating manuals, and expert insight, and make design decisions based upon shared knowledge. For BWRX-300 there exists significant operating experience and insights regarding individual degradation mechanisms that have been considered in the aging management programs. For example, the United States

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Nuclear Regulatory Commission has developed a consistent approach to aging management in connection with licence renewal for operating plants.

Known aging phenomena are quantified and considered in the design of SSC. The design includes the effects of wear and all other known age-related degradation to ensure that safety and performance are maintained for the duration of their lifetime. If the component lifetime extends to the plant service life, as is the case for passive non-replaceable components, the design considers all normal and transitory operating conditions, including testing stressors, maintenance interventions and the consequences of plant and system outages. Analyzed DBAs are considered as part of the operating life and hence part of the design calculations.

In general, margins consist of design margins, operational margins, and safety margins. They account for uncertainties, assumptions, instrument feedback tolerances and ranges, unexpected transitory peaks, contingencies, and operating flexibility. Margins are mainly set to minimize the probability of component failure. Only the unquantifiable aging effects are included in the margin estimates.

Design documents include as a minimum, the following aging management topics:

1. A recommended strategy for aging management and prerequisites for its implementation.
2. Identification of safety class SSC in the plant that could be affected by aging.
3. Proposals for appropriate materials monitoring and sampling programs, where aging may affect the capability of critical SSC to perform their functions throughout the lifetime of the plant.
4. Appropriate consideration of operating experience with respect to aging.
5. Recommendations for aging management for safety class SSC (concrete structures, mechanical components, electrical and instrumentation and control components, cables, etc.) and measures to monitor and mitigate their degradation.
6. Equipment qualification requirements of safety class SSC.
7. General principles stating how the environment of structures, systems, and components are to be maintained within specified service conditions (location of ventilation, insulation of hot SSC, radiation shielding, damping of vibrations, submerged conditions and water chemistry, selection of cable routes, etc.).

3.1.13 References

- 3.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.1-2 IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design" International Atomic Energy Agency.
- 3.1-3 IAEA Safety Standards Series No. SF-1, "Fundamental Safety Principles," International Atomic Energy Agency.
- 3.1-4 Government of Canada SOR/2000-203, "Radiation Protection Regulations,"
- 3.1-5 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 3.1-6 IAEA TECDOC-1791, "Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants," International Atomic Energy Agency.
- 3.1-7 CNSC Regulatory Document REGDOC-3.6, "Glossary of CNSC Terminology."

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- 3.1-8 CNSC Regulatory Document REGDOC-2.4.2, "Safety Analysis – Probabilistic Safety Assessment (PSA) for Nuclear Power Plants."
- 3.1-9 IEC 60880, "Nuclear power plants – Instrumentation and control systems important to safety – Software aspects for computer-based systems performing category A functions," International Electrotechnical Commission.

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Table 3.1-1: Identification of Defence Levels

Level of Defence/DL	Objective	Design Means	Operational Means
Level 1/DL1	Prevention of abnormal operation and failures	Conservative design and high quality in construction of normal operation systems, including monitoring and control systems	Operational rules and normal operating procedures
Level 2/DL2	Control of abnormal operation and detection of failures	Limitation and protection systems and other surveillance features (Safety Category 3)	Abnormal operating procedures/emergency operating procedures
Level 3/DL3	Control of design basis accidents	Engineered safety features (Safety Category 1)	Emergency operating procedures
Level 4a/DL4a	Control of design extension conditions to prevent core melt	Safety features for design extension conditions without core damage (Safety Category 2)	Emergency operating procedures
Level 4b/DL4b	Control of design extension conditions to prevent or mitigate the consequences of severe accidents	Safety features for design extension conditions with core damage (Safety Category 3)	Complementary emergency operating procedures/severe accident management guidelines
Level 5/DL5	Mitigation of radiological consequences of significant releases of radioactive materials	On-site and off-site emergency response facilities	On-site and off-site emergency plans

Plant Design Envelope					
Defense Line 1					
Normal Functions	Defense Line 1	Defense Line 2	Defense Line 3	Defense Line 4a	Defense Line 4b
	Operational States			Accident Conditions	
Normal Operation		AOO Fault Sequences	DBA Fault Sequences	DEC Fault Sequences	Practically Eliminated Conditions
				Design Extension Conditions	
	Design Basis Conditions				Severe Accidents (core damage)
				No Core damage	
				Beyond Design Basis Conditions	

Figure 3.1-1: Defence-in-Depth – Plant States and Defense lines

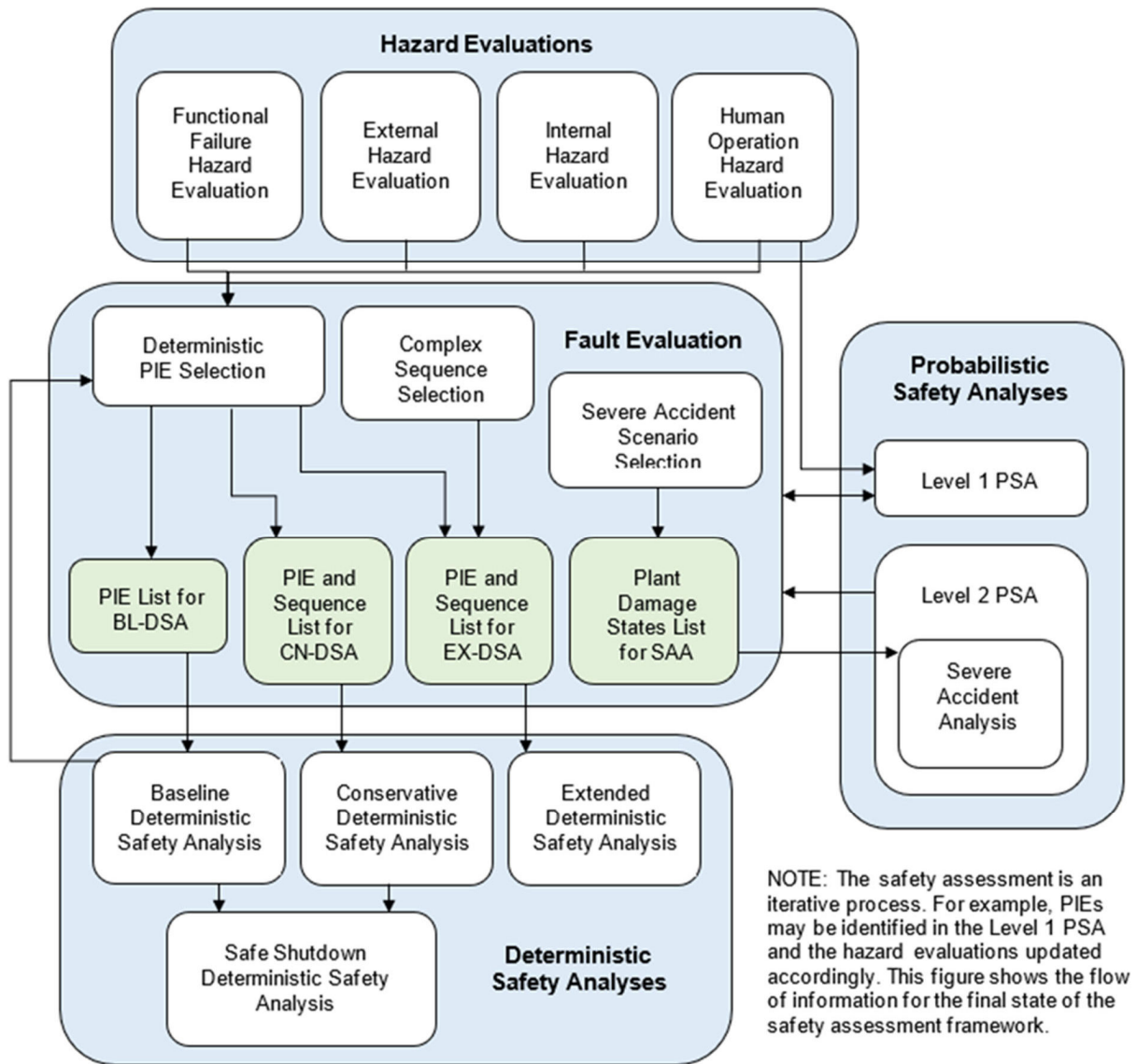


Figure 3.1-2: BWRX-300 Safety Strategy Implementation Process

3.2 Classification of Structures, Systems and Components

The BWRX-300 approach to classifying SSC is consistent with IAEA SSR-2/1 (Reference 3.2-1) and IAEA SSG-30 (Reference 3.2-2) and aligns with CNSC REGDOC-2.5.2 (Reference 3.2-3). Classification of SSC is conducted to identify the importance of the SSC with respect to safety.

This section described how BWRX-300 SSC are classified by:

- Safety Class (SC)
- Seismic Category
- Quality Group

Classification of SSC provides a means for applying appropriate design requirements and establishes a graded approach in the selection of materials, and application of codes and standards used in design, manufacturing, construction, testing and inspection of individual SSC. Sections 3.5 through 3.8 describe the codes and standards applicable to civil, mechanical, I&C, and electrical SSC based on classification.

The classification of SSC also determines the degree of redundancy, diversity, separation, and reliability/availability required as described in Subsection 3.1.7. The requirement for environmental qualification is based on the classification of SSC as described in Section 3.9. In addition, SSC classification informs procurement and quality assurance requirements as discussed in Chapter 17.

3.2.1 Safety Classification Background

The BWRX-300 approach to classifying SSC by safety class is based primarily on deterministic methods and is directly traceable to the safety functions performed by the SSC. This approach aligns with CNSC REGDOC-2.5.2, Section 7.1, as it reflects:

- Consequences of the SSC failure to perform its safety functions
- Expected frequency of the SSC being called upon to perform its safety functions
- Time following a PIE at which, or the period for which, the SSC may be called upon to perform a safety function

A fundamental element of the BWRX-300 SSC classification approach is the direct correlation between the Defense Line in which an SSC performs a function, and the relative safety importance of that function. Functions are categorized into three safety categories, Safety Category 1, Safety Category 2, and Safety Category 3, with Safety Category 1 being the most important.

3.2.1.1 Primary Function Categorization

Primary functions are those that directly perform the FSFs in support of DL2, DL3, DL4a or DL4b. Safety Categories are applied to the primary functions as follows:

1. Safety Category 1 is assigned to DL3 primary functions. DL3 functions assure the integrity of the barriers to release, place and maintain the plant in a safe state, and provide independence and diversity for all DL2 and DL4a functions caused by a single failure (and many CCFs). Accordingly, DL3 primary functions are the most important from a safety standpoint.
2. Safety Category 2 is assigned to DL4a primary functions. Both DL2 and DL4a provide a redundant means to address PIEs (generally independent of DL3 functions) and are therefore important from a safety standpoint, although less important than DL3 functions.

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DL4a functions are a backup to DL3 functions, in the unlikely event a DL3 functions fails, and therefore have a higher consequence of failure than DL2 functions and are more important from a safety standpoint than DL2 functions.

3. Safety Category 3 is assigned to DL2 and DL4b primary functions as they are relatively the least important. DL4b functions address severe accidents, which are extremely unlikely because failure of both DL3 and DL2 or DL4a functions would have to occur. Accordingly, DL4b functions are considered relatively the least important defense line functions, despite the high consequence of failure.
4. Non-Safety Category is assigned to all other functions.

The assignment of DL4a functions to Safety Category 2, to address the low probability but high consequences of failure, and the assignment of DL4b functions to Safety Category 3, based on the extremely low probability of being called upon, is consistent with CNSC REGDOC-2.5.2, Section 7.1 (Reference 3.2-3), which provides guidance on the treatment of complementary design features called upon to mitigate DEC's.

In addition to categorizing primary functions by the defense line they support, function that provide a supporting role and functions that are not immediately required following a PIE are assigned to a Safety Category as described below and summarized in Table 3.2-1.

3.2.1.2 Integral Support Functions

Integral support functions are functions that support the primary function and are required to be performed concurrently with the primary function (e.g., an HVAC system maintaining the temperature of a space or area within an acceptable range during performance of the primary function (i.e., following the initiating event) to maintain equipment in an acceptable condition).

Integral support functions are considered part of the defense line function (and therefore subject to defense line function "rules," such as independence and diversity) and are assigned the same safety category as the primary function they support.

3.2.1.3 Make-Ready Support Functions

Make-ready support functions are continuously available online functions that maintain the primary function, or a component required to perform the primary function, in a state of readiness but are not required to be performed at the time the primary function is performed. Make-ready functions must have monitoring, such that plant operators would be alerted if the make-ready support function were lost, or the readiness of the primary function or component were compromised. For example, maintaining the temperature of a pool of cooling water within acceptable limits, with monitoring by pool temperature indication is an example of a make-ready support function.

Make-ready functions are not required at the time the primary function is performed and are not considered part of the defense line function (and therefore not subject to defense line function "rules," such as independence and diversity). The primary function would eventually be considered unavailable if the make-ready function were compromised to the extent that the primary function might be compromised. Accordingly, make-ready functions are not required to be assigned the same safety category as the primary function. However, make-ready functions are important and are therefore assigned to safety categories as follows:

- Make-ready functions that support DL3 or DL4a functions are assigned to Safety Category 3
- All other make-ready functions can be assigned to Safety Category N.

3.2.1.4 Delayed Functions

Delayed functions are primary or support functions that are not required to be performed until sometime after the initiating event. Because there would be ample time during the event to ensure these functions are available, delayed functions are not required to be assigned the same safety category as functions required immediately after the initiating event. If the function is not needed until after 72 hours into the event (but before seven days), it can be classified as Safety Category 2 (instead of Safety Category 1), and if the SSC is not needed until after seven days into the event, it can be classified as Safety Category 3 (instead of Safety Category 1 or Safety Category 2). Delayed functions are not subject to defense line function "rules," such as independence and diversity.

3.2.1.5 Normal Functions

Normal functions that perform an FSF during normal plant operation or that maintain key reactor parameters (e.g., reactor pressure and temperature) within normal ranges, and their integral support functions, are assigned to Safety Category 3. Make-ready functions for normal functions can be assigned to Safety Category N. If failure of a normal function would likely result in an initiating event that could challenge an FSF, the function should be assigned to Safety Category 3.

3.2.1.6 Assignment of Safety Class to Components

Safety Class is assigned to components based on the safety category of the functions they perform as follows:

- Safety Class 1 (SC1) is assigned to SSC that perform a Safety Category 1 function
- Safety Class 2 (SC2) is assigned to SSC that perform a Safety Category 2 function
- Safety Class 3 (SC3) is assigned to SSC that perform a Safety Category 3 function
- Non-Safety Class (SCN) is assigned to all other SSC

Just as with functions, a time-dependency is introduced for components that perform or support DL3 and DL4a functions. Specifically, if the component is not needed until after 72 hours into the event (but before seven days), it can be classified as SC2 (instead of SC1), and if the component is not needed until after seven days into the event, it can be classified as SC3 (instead of SC1 or SC2) because there would be ample time during the event to ensure those components are available. (See Table 3.2-2)

Functions typically have a mission time, which is the time period during which the function is required to be performed. Only SSCs that are required during the mission time of the function are required to be assigned to the safety classes discussed above.

Some component classifications are made for components that perform FSFs but may not be explicitly defined as part of a defense line function. For example:

- Components that are part of design provisions that perform a FSF, whose failure is considered "practically eliminated," are assigned to SC1. An example is the RPV.
- Components that make up the fission product barriers are assigned to SC1.
- Components that are part of the RCPB are assigned to SC1.

The safety classification of a system is the highest safety classification of any components within the system; however, the component safety classification, and not the system safety classification, defines the design rules applied to components. Assignment of safety

classifications to systems is for convenience in understanding the relative importance of plant systems.

Not all components or parts of a system are necessarily assigned to the same safety class as the system itself. For example, a process system may be classified as SC 1 because one or more of its components support a DL3 function; however, the system may also contain components that support functions associated with other defense lines or components that support no defense line functions. These components are classified in accordance with the defense line functions they support.

Appendix 3A provides a list of the BWRX-300 principal components organized by system and includes their safety classification.

Structures are assigned a safety classification based on the highest safety classification of the components they house or support, excluding components whose failure, due to loss of functionality of the structure, would result in fail-safe performance of the component's safety category function(s). Design rules and performance requirements for structures are derived from their seismic category. Seismic categorization methodology is described in Subsection 3.2.3. The seismic category assigned to a structure is commensurate with its safety classification as listed in Section 3.3, Table 3.3-1.

3.2.2 Safety Classification Process

In alignment with both IAEA and CNSC guidance, this method of classifying the safety significance of SSC is based primarily on deterministic methods because the DL functions are identified using deterministic safety analyses. The deterministic methods are complemented (where appropriate) by probabilistic methods and engineering judgment.

Design rules are then applied to SSC based on their safety classification and the DL functions they support. The safety classification process is iterative with the deterministic and probabilistic safety assessment and is maintained and updated throughout the design phase.

The following outlines the BWRX-300 classification process.

Review and Definition of PIEs – Hazard evaluations are performed (as introduced in Section 3.1.6.4.1) to identify hazards with potential to challenge an FSF. The output of these hazard evaluations are potential PIEs.

Grouping and Identification of Bounding PIEs – Potential PIEs are grouped by plant effect and occurrence frequency. Bounding or representative PIEs and fault sequences are selected for deterministic safety analyses as described in Chapter 15, Section 15.2.

Identification of Plant-Specific Safety Functions to Prevent or Mitigate the PIEs – The deterministic safety analyses are performed and updated iteratively with design activities to establish the plant-specific functions responsible for maintaining the FSFs during PIEs and fault sequences. The identification of plant-specific functions and their assignment to a Defense Line is carried out in the Fault Evaluation described in Chapter 15, Section 15.2 with traceability of each function to each PIE and PIE sequence in which it is credited.

Safety Categorization of the Safety Functions – Functions are categorized in accordance with their safety significance and role in performing FSFs. As such, each function receives a safety categorization directly based on its assignment to a DL (as described in Subsection 3.2.1 above).

Identification of SSC that Provide the Safety Functions

Plant-level requirements are created for each DL function and decomposed into system-specific functional requirements to implement the credited DL functions, consistent with the plant

performance modeled in the safety analyses. These requirements are then allocated to the applicable system design description which identifies the components that support the system DL functions.

Assignment of SSC to a Safety Class Corresponding to the Safety Category

Safety Class is assigned to SSC based on the SSC's role in ensuring plant safety, and the defense line and FSF supported as described in Subsection 3.2.1.6 above.

Verification of SSC Classification

The deterministic safety analyses are maintained and updated as the plant design matures. Confirmation of SSC classification is achieved when the deterministic safety analyses models reflect the final plant design and demonstrate compliance to the analysis acceptance criteria (which include rules governing how classified equipment can be credited in each analysis case). This verification is complemented, as appropriate, by insights from the PSA.

Identification of Engineering Design Rules for Classified SSC

Engineering design rules are applied to SSC based on several factors including their SC, their DL role, their status as a pressure boundary component, their role during and following earthquakes, and their operational environment. The design rules establish the scope of codes and standards applied to an SSC, as well as requirements for reliability, diversity, redundancy, and independence applicable to an SSC. These design rules are discussed in Subsection 3.1.7.

3.2.3 Seismic Categories

Seismic Category reflects SSC requirements during and after a seismic event and governs how the SSC is seismically designed and qualified. Seismic Category is assigned based on the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13 (Reference 3.2-3), and CSA N289.1, Clause 5.2.5.2 (Reference 3.2-4) as follows:

1. **Seismic Category A/B** - DL3 functions are credited with remaining operable during and after a seismic event associated with a Design Basis Earthquake (DBE) as defined in Section 3.3.1. Accordingly, SSC that perform or support DL3 functions are categorized as Seismic Category A for passive SSC or Seismic Category B for active SSC. Other SSC that are classified as SC1 per Subsection 3.2.1.6, are categorized as Seismic Category A or B. Any other SSC that are a significant contributor to PSA risk for seismic events are categorized Seismic Category A or B.
2. **Seismic Category RW-IIa** - SSC for management and storage of radiological material that, if released would exceed the dose limits defined in CNSC REGDOC-2.5.2, Section 4.2.1, are categorized as Seismic Category RW-IIa per guidance in U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide (RG) 1.143, (Reference 3.2-7). These RW-IIa SSC are seismically qualified for one-half of the site-specific DBE. This approach is in accordance with CNSC REGDOC-2.5.2, Section 7.13.1, which permits the use of ASCE/SEI 43 (Reference 3.2-8) graded approach for the seismic classification of SSCs with justification. Based upon the consequences of failure, one-half of the site-specific DBE is justified as it would bound the ground motion spectra for seismic categories identified in ASCE/SEI 43 (Reference 3.2-8) for SSCs used for handling and storage of highly radioactive materials. This justification is described in NEDC-33974P (Reference 3.2-18).
3. **Non-Seismic** - All other SSC are categorized as Non-Seismic and are designed based on applicable non-nuclear requirements, such as those stipulated in the National Building Code of Canada (Reference 3.2-19).

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The BWRX-300 Containment and the Reactor Building (RB) are the only structures that house, support, or protect Seismic Category A or Seismic Category B SSC. These two structures are therefore categorized as Seismic Category A structures in the BWRX-300 design per Clause 5.2.5.2 of CSA N289.1 (Reference 3.2-4).

3.2.3.1 Seismic Interaction

SSC that are not Seismic Category A or B but whose failure during a seismic event could adversely affect the ability of any Seismic Category A or B SSC to accomplish its safety function are evaluated for seismic interaction to demonstrate that these SSC:

- Will not collapse or collide with the Seismic Category A and Seismic Category B SSC and will maintain their stability during a DBE or other relevant extreme external hazard event; or
- Impact loads that result from collapse or collision on the Seismic Category A and Seismic Category B SSC are either negligible or smaller than those considered in the design.

In accordance with requirements of Clause 7.2.1.2 of CSA N289.3 (Reference 3.2-6) and Section 6.0 of NEDO-33914 (Reference 3.2-9), interaction evaluations are performed of the Power Block structures and foundations adjacent to the Seismic Category A RB, as described in Subsection 3.3.1.2.8, to ensure:

- These structures and foundations do not collapse to compromise the safety functions of those SSC that are required to remain functional following a DBE or design tornado level event for the first 72 hours.
- The CB structure, which includes the Main Control Room (MCR) does not collapse and result in incapacitating injury to the main control room occupants or prevent their egress to the RB.

Table 3.3-1 in Section 3.3 lists the seismic category and seismic interaction evaluation requirements for structures..

Evaluations for seismic interaction of systems and components is conducted as the design advances and details supporting these evaluations are available.

3.2.4 Quality Group

In alignment with CNSC REGDOC 2.5.2, Section 7.7 (Reference 3.2-3), BWRX-300 pressure-retaining components are designed to ensure they are protected against overpressure conditions, and are classified, designed, fabricated, erected, inspected, and tested in accordance with established standards. The selection of codes and standards is commensurate with the safety class and is adequate to provide confidence that plant failures are minimized. CNSC REGDOC-2.5.2 points to ASME Boiler and Pressure Vessel Code (BPVC) (Reference 3.2-11) to meet the requirements of different classes of pressure-retaining systems, components, piping and their supports.

BWRX-300 design utilizes a Quality Group designation per the guidance in USNRC RG-1.26 (Reference 3.2-10) as a method for establishing the appropriate codes and standards based on the importance of the pressure-retaining function of the component. Items are classified as Quality Group A, B, C or D. The guidance and classification method are used with some clarification based on the unique design of the BWRX-300.

Table 3.2-3 tabulates the design and fabrication requirements for each Quality Group. For mechanical equipment that does not fall within the scope of USNRC RG 1.26 (Reference 3.2-10),

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appropriate industrial codes and standards are applied. Per CNSC REGDOC-2.5.2, alternative codes and standards may be used with justification and consistent with a graded approach.

Appendix 3A provides a list of the BWRX-300 principal components organized by system and includes their Quality Group. The Quality Group for structures is listed in Section 3.3, Table 3.3-1.

3.2.5 References

- 3.2-1 IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design," International Atomic Energy Agency.
- 3.2-2 IAEA Safety Standards Series No. SSG-30, "Safety Classification of Structures, Systems, and Components in Nuclear Power Plants," International Atomic Energy Agency.
- 3.2-3 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.2-4 CSA N289.1, "General Requirements for Seismic Design and Qualification of Nuclear Power Plants," CSA Group.
- 3.2-5 USNRC Regulatory Guide 1.29, "Seismic Design Classification for Nuclear Power Plants."
- 3.2-6 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.2-7 USNRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."
- 3.2-8 ASCE/SEI 43, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," American Society of Civil Engineers.
- 3.2-9 NEDO-33914, "BWRX-300 Advanced Civil Construction and Design Approach," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.2-10 USNRC Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."
- 3.2-11 ASME (BPVC), "Section III," American Society of Mechanical Engineers.
- 3.2-12 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 3.2-13 API 620, "Design and Construction of Large, Welded, Low-Pressure Storage Tanks," American Petroleum Institute.
- 3.2-14 API 650, "Welded Steel Tanks for Oil Storage," American Petroleum Institute.
- 3.2-15 AWWA D100-11, "Welded Carbon Steel Tanks for Water Storage," American Water Works Association.
- 3.2-16 ASME B96.1, "Welded Aluminum-Alloy Storage Tanks," American Society of Mechanical Engineers.
- 3.2-17 TEMA, "Standards of the Tubular Exchanger Manufacturers Association," Tubular Exchange Manufacturers Association.
- 3.2-18 NEDC-33974P, "BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report," GE-Hitachi Nuclear Energy Americas, LLC.

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- 3.2-19 Canadian Commission on Building and Fire Codes, "National Building Code of Canada,"
National Resource Council of Canada.

Table 3.2-1: Safety Category for Functions Based on Defense Line Assignment

Safety Category	Defense Line 3 Functions	Defense Line 4a Functions	Defense Line 2/4b Functions	Normal Functions
1	<ul style="list-style-type: none"> Primary and Integral support functions required within the first 72 hours of an event 			
2	<ul style="list-style-type: none"> Primary and integral support functions required after 72 hours but before 7 days after an event 	<ul style="list-style-type: none"> Primary and integral support functions required within the first 7 days of an event 		
3	<ul style="list-style-type: none"> Primary and integral support functions required after 7 days after an event Make-ready support functions 	<ul style="list-style-type: none"> Primary and integral support functions required after 7 days Make-ready support functions 	<ul style="list-style-type: none"> All primary and integral support functions 	<ul style="list-style-type: none"> Normal functions that perform a fundamental safety function Normal functions that maintain key reactor parameters (e.g., pressure and temperature) within normal ranges Integral support functions
N			<ul style="list-style-type: none"> Make-ready support functions 	<ul style="list-style-type: none"> Make-ready support functions

Table 3.2-2: Safety Class for SSC

Safety Class	Safety Category 1 Functions	Safety Category 2 Functions	Safety Category 3 Functions	Safety Category N Functions	Other
1	<ul style="list-style-type: none"> SSCs required within first 72 hours of event 				<ul style="list-style-type: none"> Components that are part of design provisions that perform a FSF, whose failure is considered "practically eliminated" Components that make up the fission product barriers Components that are part of the reactor coolant pressure boundary
2	<ul style="list-style-type: none"> SSCs required after 72 hours but before 7 days 	<ul style="list-style-type: none"> SSCs required within first 7 days of event 			
3	<ul style="list-style-type: none"> SSCs required after 7 days 	<ul style="list-style-type: none"> SSCs required after 7 days 	<ul style="list-style-type: none"> All SSCs 		
N				<ul style="list-style-type: none"> All SSCs 	

Note: Only SSCs that are required during the mission time of the function are required to be assigned to the safety classes discussed above.

Table 3.2-3: Codes and Standards for Pressure-Retaining Equipment

Quality Group	ASME BPVC Section III Code Classes	Pressure Vessels and Heat Exchangers ⁽⁴⁾	Pipes, Valves, and Pumps	Storage Tanks 0-103 kPaG (0-15 psig)	Storage Tanks Atmospheric	ASME BPVC Section III Component Supports	Non-ASME BPVC Section III Component Supports	Core Support Structures and Reactor Internals	Containment Boundary
A	1	NCA and NB	NCA and NB	—	—	NCA and NF	—	—	—
B	2	NCA and NCD	NCA and NCD	NCA and NCD	NCA and NCD	NCA and NF	—	—	—
	MC	—	—	—	—	—	—	—	NCA and NE ⁽¹⁾
	CS	—	—	—	—	—	—	NCA and NG	—
C	3	NCA and NCD	NCA and NCD	NCA and NCD	NCA and NCD	NCA and NF	—	—	—
D	—	ASME BPVC Sect. VIII Division 1	ASME B31.1 for piping and valves ⁽²⁾	API 620 or equivalent ³	API 650 AWWA D100-11 ASME B96.1 or equivalent ⁽³⁾	—	Manufacturer Specified Standards, e.g., ASME B31.1, AISC	—	—

(1) Excluding the Steel-plate Composite Containment Vessel. See Section 3.5.3 for applicable codes and standards.

(2) For pumps classified in Quality Group D, the ASME BPVC, Section VIII, Division 1 is used as a guide in determining the wall thickness for pressure-retaining parts and in sizing the cover bolting.

(3) Tanks are designed to meet the intent of American Petroleum Institute (API) Standard 620 (Reference 3.2-13), API 650 (Reference 3.2-14), American Water Works Association (AWWA) (Reference 3.2-15), and/or ASME B96.1 standards (Reference 3.2-16, as applicable).

(4) For Tubular Exchanger Manufacturers Association (TEMA)-style heat exchangers, both the ASME Code and TEMA standard (Reference 3.2-17) are considered. Other heat exchanger design styles/configurations are not subject to the TEMA standard.

(5) Acronyms used in Table 3.2-2 refer to the ASME BPVC Section III (Reference 3.2-11) subsections as follows:

- Subsection NCA-General Requirements for Division 1 and Division 2
- Division 1 Subsections:
 - Subsection NB – Class 1 Components

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- Subsection NCD - Class 2 and 3 Components
- Subsection NE- Metal Containment (MC)
- Subsection NF – Supports
- Subsection NG – Core Support Structure (CS)

3.3 Protection Against External Hazards

Complying with Section 7.4.2 of CNSC REGDOC-2.5.2 (Reference 3.3-1), the BWRX-300 design considers natural and human-induced external hazards that may be linked with significant radiological risk. This section discusses external hazards relevant to the DNNP site and the BWRX-300 approach to prevent and mitigate their effects on Safety Class 1 (SC1) Structures, Systems and Components (SSC). SC2/SC3 SSC that are credited in the fault evaluation with mitigating fault sequences initiated by external hazards, and SSC whose failure can affect the structural integrity or safety class functions of adjacent SC1 SSC are also protected against external hazards.

The determination of the external hazards considered in the BWRX-300 design relies on the collection of the geotechnical, seismological, hydrological, hydrogeological, and meteorological reference data, and human-induced external events presented in Chapter 2, Section 2.2, Section 2.4, Section 2.5, Section 2.6 and Section 2.7. For external hazards, the main protection is provided by the civil structures. The design against external hazards is such that a design basis external hazard does not lead to a Design Basis Accident (DBA) or a Beyond Design Basis Accident (BDBA). Significant safety margins are included in the evaluation of the design basis external hazards and the associated design aspects to ensure a conservative design. Assurance that the overall reactor plant is resilient to external hazards is provided by the demonstration that SSC do not fail when subject to these hazards and generated loadings. Demonstration of the adequacy of protection measures is provided in the applicable PSAR chapters covering the design of SSC.

Malevolent acts considered in the robustness design are discussed in Subsection 3.3.7.4.

Protection and mitigation methods considered in the design are in line with the design safety objectives and Defence-in-Depth (D-in-D) concept discussed in Subsections 3.1.1 and 3.1.6, respectively. They include the use of physical separation, barriers/shielding, qualification of equipment and instrumentation for the hazards environment and monitoring programs to preclude unacceptable radiation releases following accidents due to external hazards.

When applicable, loads generated by external hazards are considered in the BWRX-300 design following requirements in Section 7.15.1 of CNSC REGDOC-2.5.2 and CSA N291 (Reference 3.3-2). Combination of loads from randomly occurring individual external hazards is considered in the design to ensure structures are adequately protected against external hazards.

A principal safety objective of the BWRX-300 Safety Strategy is the demonstration that the overall reactor plant design is resilient to hazards through D-in-D. This means that the design provisions optimize protection to provide the highest level of safety that can reasonably be achieved such that relevant dose targets on-site and off-site are met and the resilience of the reactor plant to external hazards reduces risk. The process of demonstrating that the reactor plant is resilient starts with the systematic identification of Postulated Initiating Event (PIEs) with a potential to challenge a fundamental safety function, and to organize them into the fault list developed as per Chapter 15, Section 15.2. Combinations of randomly occurring individual events are considered in these evaluations in accordance with requirements in CNSC REGDOC-2.5.2, Section 7.4.3. Deterministic and probabilistic safety analyses are then performed as discussed in Chapter 15, Sections 15.5 and 15.6 to confirm the design adequacy and its resilience to these hazards. Summary of results of the safety assessments are presented in Section 15.7.

3.3.1 Seismic Design

For seismic design, BWRX-300 SSC are categorized as Seismic Category A, Seismic Category B, Seismic Category RW-IIa and/or Non-Seismic Category as discussed in Subsection 3.2.3. This

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seismic categorization reflects SSC's functional and performance requirements during or after a seismic event and impacts their design.

Following the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, CSA N289.1 (Reference 3.3-3) and U.S. NRC RG 1.208 (Reference 3.3-4), Seismic Category A and Seismic Category B SSC are seismically qualified to withstand the effects of a Design Basis Earthquake (DBE) that is developed:

1. Based on the geological, seismological, and geotechnical conditions at the site described in Chapter 2, Section 2.7
2. Following the performance-based approach of ASCE/SEI 43 (Reference 3.3-5) Section 2 for development of DBE for seismic design of structures achieving a target performance goal of $1E-5$ per year
3. Meets the minimum earthquake requirements of CSA N289.3 (Reference 3.3-6), Clause 4.2

The development of the 5% damped Acceleration Response Spectra (ARS) defining the amplitude and frequency content of the bounding site-specific DBE input ground motion used for the seismic qualification of Seismic Category A and B SSC is discussed in Subsection 3.3.1.1.

Table 3.3-1 provides the seismic categorization of BWRX-300 structures. Per Subsection 3.2.3, the containment, and Reactor Building (RB) are the only structures that house, support, or protect Seismic Category A or Seismic Category B SSC. As a result, the integrated RB structure, which consists of the RB, containment and containment internal structures is the only structure categorized as Seismic Category A in the BWRX-300 design. As shown in Table 3.3-1, the seismic design of the Seismic Category A structures considers Limit State LS-D response defined in Table 1-2 of ASCE/SEI 43 as essentially elastic response without any significant permanent deformation. According to U.S. NRC RG 1.208, this ensures a consistent level of safety from earthquake-caused failures defined by level of response resulting in an onset of significant inelastic deformations with a probability of unacceptable performance:

- Less than 1% for a DBE ground motion level
- Less than 10% for ground motion with 1.5 times the DBE intensity

The Radwaste Building (RWB) which processes and houses liquid, solid and gaseous radwaste is categorized as Seismic Category RW-IIa as shown in Table 3.3-1. The remaining BWRX-300 Power Block structures, which consist of the Control Building (CB), Turbine Building (TB) and Reactor Auxiliary Bay (See Chapter 1, Appendix A, Figure A1.4-1) are categorized as Non-Seismic.

Due to their proximity to the Seismic Category A RB, the RWB, CB, TB and Reactor Auxiliary Bay are evaluated for interaction with the integrated RB structure per the requirements in SSR-2/1 (Reference 3.3-7), Section 5.19, as discussed in Subsection 3.2.3.1. The interaction evaluation methodology is presented in Subsection 3.3.1.2.8. Table 3.3-1 summarizes the seismic design basis for the BWRX-300 structures based on their seismic categories. Per Table 3.3-1, the RW-IIa structures are designed per CSA N291 and U.S. NRC RG 1.143 (Reference 3.3-8), while Non-Seismic Category structures are designed in accordance with the National Building Code of Canada (NBC) (Reference 3.3-9). The primary focus of this section is on the seismic qualification of Seismic Category A and Seismic Category B SSC. The seismic design of the RW-IIa and Non-Seismic Category structures is further discussed in Chapter 9B, Section 9B.3.

Seismic robustness of Seismic Category A structures is evaluated for a Design Extension Condition (DEC) Checking Level Earthquake (CLE) as described in Subsection 3.5.6.1.

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The BWRX-300 design considers Operating Basis Earthquake (OBE) and Site Operating Earthquake loads as 1/3 of the DNNP site-specific DBE. Per Appendix S to 10 CFR 50 (Reference 3.3-10), design load combinations that consider OBE and Site Operating Earthquake loads are not required, except for the design of metal containment components where the OBE loads are considered for post-flooding condition and cyclic loading considerations, as noted in Table 9B-1 in Chapter 9B. OBE is not used as reference earthquake for the BWRX-300 DNNP plant shutdown.

The DNNP BWRX-300 seismic instrumentation is discussed in Subsection 3.3.1.5. As described in Subsection 3.3.1.5, the criteria for seismic instrumentation, plant shutdown, evaluation and inspection are in accordance with the guidelines of CSA N289.5 (Reference 3.3-11) and Clause 6.5 of CSA N289.1.

3.3.1.1 Bounding Seismic Design Parameters

Consistent with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, the design of the BWRX-300 is based on DNNP site-specific geotechnical and seismic inputs. Bounding seismic design parameters are developed based on the data that was available prior to the completion of the characterization of geotechnical and seismic conditions at the DNNP site presented in Chapter 2, Section 2.7. These conservative site-specific seismic inputs adequately address uncertainties related to the use of incomplete (preliminary) characterizations of the DNNP geotechnical and seismic conditions.

The 5% damped spectra defining the magnitude and frequency content of the DNNP bounding site-specific design ground motion are developed based on the results of probabilistic Site Response Analysis (SRA) presented in Subsection 3.3.1.1.2 using as input the dynamic subgrade properties dynamic subgrade properties described in Subsection 3.3.1.1.1.

The results of the probabilistic SRA are also used for the development of bounding stiffness and damping properties of subgrade materials that are compatible with the free-field strains generated by a typical design level earthquake event.

The bounding DBE ground motion response spectra in Subsection 3.3.1.1.3 and the bounding strain-compatible dynamic subgrade profiles discussed in Subsection 3.3.1.1.6 provide a conservative seismic design that adequately address the aleatory variabilities and epistemic uncertainties in the geotechnical properties of the DNNP site.

Five sets of ground motion time histories compatible to the bounding DBE ground motion response spectra are developed, as described in Subsection 3.3.1.1.4, for use as input for the linear seismic Soil-Structure Interaction (SSI) analysis.

3.3.1.1.1 Bounding Dynamic Subgrade Properties

The bounding seismic design parameters are developed using dynamic properties for the subgrade rock, in-situ soil, and engineered fill that are determined based on the data obtained from multiple geotechnical investigations that were completed at the vicinity of the DNNP site prior to the geotechnical site investigations and laboratory tests described in Chapter 2 Section 2.7.3.

For use as input for the probabilistic SRA described in Subsection 3.3.1.1.2, bounding subgrade dynamic profiles are developed reflecting anticipated as-built conditions at the site after construction of the BWRX-300 SMR that include compacted fill from about elevation 80 to 82 m Canadian Geodetic Datum (CGD) to the final grade at elevation 88 m CGD. The layering of the in-situ soil materials is determined based on the stratigraphy obtained from the studies presented in:

- NK054-REP-01210-00098 (Reference 3.3-12) providing data from multiple borings near the proposed BWRX-300 SMR site (B-104, B-113, B-116, and B-118), and

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- NK054-REF-01210-0418696 (Reference 3.3-13) providing data from deeper borings close to the BWRX-300 SMR (AMC-03ALT).

It is anticipated that the loose surficial soil materials that are not competent for supporting the heavy foundations of the power block buildings and have potential for liquefaction during earthquakes will be excavated and replaced with an engineered fill obtained by reconditioning and compacting the in-situ soils from the fill layer, surficial lacustrine layer, and upper till materials excavated from the upper 6 to 8 m of the site. Results of compaction tests of the in-situ soil materials in 2009 NK054-REP-07730-00005 (Reference 3.3-14) are used as basis for development of the engineered fill dynamic properties.

The probabilistic SRA, described in Subsection 3.3.1.1.2, explicitly consider the epistemic uncertainties in the estimation of subgrade dynamic properties by using 50th percentile Best Estimate (BE), 10th percentile Lower Realization (LR), and 90th percentile Upper Realization (UR) values for the shear wave velocities and kappa representing the dissipation of the energy for the site. For the different subgrade materials, standard deviation for the natural log of the shear wave velocity is assigned to adequately define the aleatory variability of subgrade dynamic stiffness properties.

The profile of bounding rock dynamic properties is developed directly from the recommended shear wave velocity profiles in 2012 NK054-REF-01210-0418696 (Reference 3.3-13). The Base Case values and variations for dynamic properties of rock are presented in Table 3.3-2. The compression wave velocities, shear wave velocities, and Poisson's ratio for the bedrock rock units are obtained from the measured values from 2012 NK054-REF-01210-0418696 (Reference 3.3-13) without modification. The rock Poisson's ratio was calculated from the measured compression and shear wave velocities following the recommendation of the NEDO-33914 (Reference 3.3-15).

The profile of base case dynamic properties presented in Table 3.3-2 considers the following:

1. The "Top of Bedrock Rock" elevation is 64.1 m CGD with a σ_{TOR} of ± 1 m
2. The variation in the rock layers assumes ± 2 m
3. The $\sigma_{\mu \ln}$ represents the epistemic uncertainty for estimating LR (10th percentile) and UR (90th percentile) profiles
4. The $\sigma_{\ln V_s}$ represents the aleatory uncertainty for randomization of the shear wave velocities.

Epistemic uncertainty in the distribution of the shear wave velocity profiles ($\sigma_{\mu \ln}$) was estimated based on the range of V_s values measured in each bedrock layer; however, the estimated values were lower than the typical estimate of 0.35 in the 2013 EPRI TR-1025287 (Reference 3.3-16). Based on a comparison with the estimated $\sigma_{\mu \ln}$ values, a $\sigma_{\mu \ln}$ of 0.10 is selected based on the similar results from all three borings, as described in the 2012 NK054-REF-01210-0418696 (Reference 3.3-13). Using a higher $\sigma_{\mu \ln}$ value was not justified by the site data. Aleatory uncertainty considers a standard deviation for the natural log of the shear wave velocity ($\sigma_{\ln V_s}$) of 0.15 for the bedrock layers based on the 2013 EPRI TR-1025287 (Reference 3.3-16).

Table 3.3-3 presents the small-strain dynamic properties of the engineered fill and the in-situ soil. The small-strain values of the soil materials are estimated from the measured SPT N60 values provided in the NK054-REF-01210-0418696 (Reference 3.3-13) and the NK054-REP-01210-00098 (Reference 3.3-12). Three sets of shear wave velocities are estimated for each soil layer using the average, lowest, and highest N60 values. The results for the average, lower, and upper estimates were then combined using weights of 0.4, 0.3, and 0.3, respectively, to approximate a normal distribution, per the 2013 EPRI TR-1025287 (Reference 3.3-16).

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The uncertainty in the estimates of soil V_s is considered using a $\sigma_{\mu \ln}$ of 0.35 to 0.40. Per recommendations in the 2013 EPRI TR-1025287 (Reference 3.3-16), a value of 0.35 is intended for sites with limited shear wave velocity data while a value of 0.50 is appropriate for a site without shear wave velocity data. The selected $\sigma_{\mu \ln}$ values generally cover the range of estimated V_s values in each soil layer at the 10th and 90th percentile.

Dynamic fill properties are estimated from the N60 values. Two correlations are used to estimate V_s for the N60 values, per the 2012 PEER Report 2012/08 (Reference 3.3-17). The selected V_s correlations use the N60 values and are appropriate for fill using a range of soils. The average of the two correlations was used as the shear wave velocity in each fill layer. A $\sigma_{\mu \ln}$ of 0.40 was selected. The selected $\sigma_{\mu \ln}$ value is considered reasonable due to the limited information on the fill materials. A $\sigma_{\ln V_s}$ value of 0.25 is used for the fill and upper till and a value of 0.15 is used for the deeper in-situ soil layers.

The BE, LR, and UR variations of the kappa parameter, used to establish consistent damping ratios for the rock layers at the site are presented in Table 3.3-4. The kappa value was estimated following the guidance of the 2013 EPRI TR-1025287 (Reference 3.3-16) for CEUS firm rock profiles with a thickness of less than 1000 m and a total standard deviation of 0.47 for kappa based on the 2014 PEER Report No. 2014/12 (Reference 3.3-18).

The BE, LR and UR of the shear wave velocity profile representing the assumed as-built conditions are presented in Figure 3.3-1.

The dynamic subgrade stiffness properties of in-situ soil and engineered fill materials in Table 3.3-3 correspond to small-strain levels. To account for the nonlinearity of the engineered fill and in-situ soil materials. The following two sets of strain-dependent property curves are recommended in EPRI TR-1025287 (Reference 3.3-16, Section B-3.3):

- EPRI curves from the 1993 EPRI TR-102293, "Guidelines for determining design basis ground motions (Reference 3.3-19)
- Peninsular Range curves, Silva, W.J., N. Abrahamson, G. Toro and C. Costantino. (1996). Description and validation of the stochastic ground motion model (Reference 3.3-20)

The Peninsular Range curves are used for the development of bounding seismic design parameters to account for the strain-dependance of the soil and engineered fill dynamic stiffness and damping properties. The EPRI curves are not considered because the results of SRA indicated excessive softening of the soil and fill layers which can result in unconservative estimates of the seismic response at the ground surface, per the 2013 EPRI TR-1025287 (Reference 3.3-16, Section 5.0, and Figure 5-7).

3.3.1.1.2 Site Response Analyses

Probabilistic Site Response Analyses (SRA) are performed to accommodate the effects of overlying materials on the seismic hazard considering the epistemic uncertainties and aleatory variabilities in the site parameters to preserve the desired hazard levels and performance goals per requirements of CSA N289.2 (Reference 3.3-21) and regulatory guidelines of U.S. NRC RG 1.208. These SRA consider as-built conditions at the DNNP site after the excavation, construction, and backfilling. The equivalent linear approach is used for the SRA to account for the non-linear response of the soil. Curves representing the shear modulus reduction (G/G_{max}) and damping of the soil materials as a function of strain are used to iteratively adjust the shear modulus and damping ratio of the soil based on the calculated effective soil shear strain until convergence is obtained.

As discussed in Subsection 3.3.1.1.1, epistemic uncertainties in the shear wave velocities and the dissipation of energy for the site represented by the coefficient kappa are explicitly considered in

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the evaluation of DNNP bounding seismic parameters. To account for the epistemic uncertainties, the probabilistic SRA consider three sets of values BE, LR, and UR for shear wave velocity, presented in Figure 3.3-1 and kappa values presented in Table 3.3-4, resulting in a total of 9 sets of base case analyses. Per 2013 EPRI TR-1025287 (Reference 3.3-16), weight factors of 0.4, 0.3, and 0.3 are assigned for the BE, LR and UR cases, respectively. The cases considered for the epistemic uncertainties and their associated weight factors are presented in Figure 3.3-2.

The SRA consider aleatory variabilities related to variations in layer thicknesses including rock depth, shear wave velocities, non-linear degradation curves for the engineered backfill and soil layers, and rock damping. The aleatory variabilities are included in the site response analysis by randomization of the BE, LR and UR shear wave velocity base case profiles, using a sample size of 60 with log-normal distributions.

The range of simulated shear wave velocities is limited to two log-standard deviations above and below the specified median value to bound the randomized profiles within physically plausible limits.

Toro's site variation model (Reference 3.3-22) is used for the randomization of the thickness of soil and rock layers. The site variation model parameters are modified to capture a value of 1 m for the variation of rock depth without regards to the thickness variation in the soil layers above or the rock layers below the rock top elevation. This is a reasonable approximation since:

- The top layer is engineered backfill
- The effects of the thickness variations within the soil and rock layers on the site response are insignificant compared to the variation of the elevation of the rock and soil interface

Figure 3.3-3 shows the suite of 60 random shear wave profiles that include the thickness variations obtained from the randomization of the BE shear wave and BE kappa value (BE-BE) base case profile. The thick black line in the plot designates the resulting mean profile.

The curves representing the shear modulus reduction (G/G_{max}) and damping of the soil materials with strain are randomized into 60 realizations with correlated log-normal distribution using the Darendeli model (Reference 3.3-23). The damping of subgrade materials is limited to 15% in accordance with the regulatory guidance of ASCE/SEI 4 (Reference 3.3-24), Section C5.2 and U.S. NRC RG 1.208, Appendix E. Figure 3.3-4 shows examples of randomized modulus reduction and material damping curves. The thick black lines in these plots designates the resulting mean curves.

Approach 1, from the approaches defined in NUREG/CR-6728 (Reference 3.3-25), is implemented for the SRA, where the reference site Uniform Hazard Response Spectra (UHRS) with Mean Annual Probability of Exceedance (MAPE) of $1E-3$, $1E-4$ and $1E-5$, are directly used as input control motions and propagated from the bedrock with reference shear wave velocity of 2,800 m/sec through the randomized subgrade profiles. This allows the 5% damped ARS results of Approach 1 SRA to be directly used for the development of the UHRS representing the seismic hazard at the horizons of interest.

Approach 1 is selected as appropriate approximation for the purposes of development of bounding seismic parameters using a preliminary site information.

The reference site UHRS at $1E-03$, $1E-04$, and $1E-05$ MAPE levels are developed using the results of the PSHA documented in NK38-REP-03611-10041 (Reference 3.3-26). Between the different options considered in this PSHA, Option 2 for CAV filtering of magnitudes 5 and above is used as input for the Approach 1 SRA, as it provides the greater seismic hazard. Figure 3.3-5 shows the bedrock UHRS used as input for the SRA.

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Using the random vibration theory, power spectral density functions for the reference site motions are calculated iteratively from the input UHRS and propagated throughout the randomized shear wave profiles to calculate power spectral density functions at the horizons of interest. 5% damped ARS at each horizon of interest are then calculated from their corresponding power spectral density functions implementing the random vibration theory approach.

For each of the 9 base cases shown in Figure 3.3-2 and MAPE considered, log-mean (μ_i) and log-Standard Deviation (σ_i) 5% damped ARS results are calculated from the SRA of the 60 random profiles. UHRS representing the mean estimate of the seismic hazard at the horizons of interest are calculated by applying weight factors (w_i) to the log mean ARS results from the different base case analyses as follows:

$$UHRS = \sum_i w_i \mu_i$$

Figure 3.3-6 and Figure 3.3-7 show with thick solid red lines the MAPE 1E-4 and 1E-5 UHRS representing the seismic hazard at the ground and top of rock surfaces, respectively, together with the corresponding log-mean ARS calculated from the analyses of 9 base cases.

Log-Standard Deviation values σ_T and σ_{Ep} are calculated as follows, representing the composite (total) uncertainty and epistemic uncertainty of the calculated hazard at the horizons of interest, respectively:

$$\sigma_T = \sqrt{\sum_i w_i ((\mu_i - \mu_T)^2 + \sigma_i^2)}$$
$$\sigma_{Ep} = \sqrt{\sum_i w_i (\mu_i - \mu_T)^2}$$

Figure 3.3-8 and Figure 3.3-9 present the composite and epistemic uncertainties for the MAPE 1E-4 and 1E-5 seismic hazard for the responses at the ground and top of rock surfaces, respectively. The figures also show the log-Standard Deviation of the ARS results for the 9 base cases.

Upper Bound (UB) estimates of the UHRS ($UHRS_{UB}$) are developed to account for the epistemic uncertainties related to the site inputs and simplified SRA methodology by applying one epistemic log-normal Standard Deviation (σ_{Ep}) increments to the mean hazard estimate UHRS as follows:

$$UHRS_{UB} = UHRS \times e^{\sigma_{Ep}}$$

Figure 3.3-6 and Figure 3.3-7 show with thick dashed lines the UB UHRS for MAPE 1E-4 and 1E-5 representing the UB estimates of the seismic hazard at the ground and top of rock surfaces, respectively.

3.3.1.1.3 Design Basis Seismic Ground Motion Response Spectra

Acceleration response spectra at 5% damping define the amplitude and frequency content of the BWRX-300 design ground motion consistent with Clause 4.3 of CSA N289.3. In accordance with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, the horizontal ground motion design spectra are developed following the methodology specified in Section 2 of ASCE/SEI 43 using the UHRS results with annual probability of exceedance of 1E-4 and 1E-5 per year.

Additional requirements for developing the site-specific DBE for the design of the deeply embedded Seismic Category A integrated RB structure are provided in Section 5.2.2 of NEDO-33914 Revision 2 (Reference 3.3-15).

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The following horizontal and vertical spectra define the amplitude and frequency content of the DNNP site-specific DBE ground motion for the SSI analysis of the BWRX-300 deeply embedded RB structure:

1. Foundation Input Response Spectra (FIRS) defining the DBE ground motion at bottom of RB Foundation.
2. Performance Based Surface Response Spectra (PBSRS) defining the DBE ground motion at the finished plant grade elevation.
3. Performance Based Intermediate Response Spectra (PBIRS) defining the DBE ground motion at intermediate embedment depth elevation established, following the guidelines in NEDO-33914 Revision 2, Section 5.2.2 at the top of the rock elevation having a significant contrast between rock and overlaying soil shear wave velocities.

The purpose of PBIRS is to ensure the ground motions used as input for the SSI analyses of deeply embedded structures are adequate throughout the depth of the embedment.

Horizontal FIRS, PBSRS and PBIRS are developed following the performance-based approach criteria of ASCE/SEI 43, Section 2 for DBE with a target performance goal of 1E-5. Instead of using UHRS representing the mean estimate of the seismic hazard as mandated by ASCE/SEI 43, the bounding FIRS, PBSRS and PBIRS are conservatively developed using the 1E-4 and 1E-5 MAPE UHRS representing UB estimates of the seismic hazard. These UB UHRS are developed as described in Subsection 3.3.1.1.2 to account for the epistemic uncertainties related to the site inputs and simplified SRA methodology. The resulting horizontal Ground Motion Response Spectra (GMRS) are further adjusted to meet the minimum required response spectra requirement using the generic spectrum in CSA N289.3, Clause 4.3.2 anchored at the minimum peak ground acceleration value of 0.1g.

Horizontal reference site hard rock GMRS is also developed following the ASCE/SEI 43 performance-based approach using the UHRS obtained from the PSHA documented in NK38-REP-03611-10041 (Reference 3.3-26) representing the reference site hazard with MAPE of 1E-4 and 1E-5. This reference site hard rock spectrum is used to conservatively neglect the de-amplifications of the reference hazard motion as it propagates through the rock column. A single rock design ground motion response spectrum is developed as a conservative representation of the amplitude and frequency content of the horizontal rock GMRS by enveloping, as shown in Figure 3.3-10 the three GMRS representing the seismic hazard at FIRS, PBIRS and reference site hard rock horizons.

The horizontal PBSRS representing the amplitude and frequency content of the design motion at the ground surface are increased to conservatively account for the uncertainties in the soil column properties that may result in spectral peak shifts by connecting the spectral peaks in the PBSRS at frequencies of 8.3 Hz and 20.4 Hz using linear interpolation in the logarithmic space.

Figure 3.3-11 presents the development of the enveloping 5% damped PBSRS representing the amplitude and frequency content of the horizontal design ground motion at the finished grade elevation.

Vertical rock GMRS and PBIRS are developed by applying frequency-dependent Vertical-over-Horizontal (V/H) ratios to the bounding horizontal spectra, in accordance with the requirements of CSA N289.3, Clause 4.3.3.3 and U.S. NRC RG 1.208.

The rock V/H ratios that are used for calculation of vertical rock GMRS, are constructed using the CEUS hard rock V/H ratios from NUREG/CR-6728 (Reference 3.3-25). The vertical PBSRS are calculated using soil V/H that are constructed following the methodology for CEUS soil sites using the procedure outlined in Appendix J of NUREG/CR-6728 (Reference 3.3-25). The rock and soil

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V/H ratios used for calculation of the bounding vertical ground motion design spectra are presented in Figure 3.3-12.

Figure 3.3-13 presents the site-specific horizontal and vertical rock Design Ground Response Spectrum (DGRS) and PBSRS defining the bounding design ground motion for the seismic analysis of the BWRX-300 Seismic Category A structures and for the seismic interaction evaluations discussed in Subsection 3.3.1.2, and compares these bounding values to the corresponding ground motion response spectra developed using the latest available geotechnical and seismological data (described in Chapter 2, Section 2.7), which were not available at the time of development of the bounding seismic design parameters.

The bounding horizontal and vertical peak ground accelerations for the rock design ground motion is 0.31 g. For the surface ground motion, the bounding peak accelerations are 0.532 g and 0.516 g for the horizontal and vertical directions, respectively. Peak ground acceleration values are defined as the ground motion acceleration values at 100 Hz.

NEI checks are performed following the procedure described in Section 5.3.4 of NEDO-33914 Revision 2 to ensure the ground motion used as input for the deterministic SSI analyses of deeply embedded RB structure at the RB foundation bottom elevation meets the regulatory guidance of U.S. NRC DC/COL-ISG-017 (Reference 3.3-27) to be hazard consistent with the results of probabilistic SRA. Horizontal and vertical rock design GMRS input motions are propagated upward through the strain-compatible soil profiles, developed as described in Subsection 3.3.1.1.6, from the bottom of foundation to the profiles surface. The envelope of the 5% damped ARS results for the responses at surface of the profiles are compared to the PBSRS. When the enveloped ARS do not meet or exceed the PBSRS, the design spectra are augmented to ensure that the augmented motion satisfies the NEI check. The augmented spectra are further increased to smooth spectral peaks and fill the valleys. Figure 3.3-14 presents the NEI check augmented and smoothed horizontal and vertical 5% damped spectra defining the amplitude and frequency content of the SSI input control motion applied to the SSI model at the RB foundation bottom.

As shown in Figure 3.3-13, in the frequency range of 0.5 to 50 Hz, which is of interest for the seismic design, the bounding horizontal Rock DGRS and PBSRS envelop the corresponding updated design response spectra discussed in Chapter 2, Section 2.7. Exceedances can be observed in the vertical Rock DGRS of up to 10% for frequencies up to 15 Hz. There are also exceedances in the vertical PBSRS of up to 20% for frequencies ranging from 2 Hz to 30 Hz.

The results of the sensitivity evaluation discussed in Chapter 9B Appendix 9B.C indicate the conservatism introduced in the bounding DNNP site-specific seismic design by using the enhanced input ground motion in Figure 3.3-14. Considering this and the other sources of conservatism in the analysis inputs and methodology as well as the considerable margins in the site-specific design of the RB integrated structures demonstrated by the structural design evaluations discussed in Chapter 9B Appendices 9B.E – 9B.G, the conclusions of the bounding seismic SSI evaluations are not expected to be affected by the relatively small exceedances of bounding ground motion Design Response Spectra observed in Figure 3.3-13.

3.3.1.1.4 Design Time Histories

Design ground motion acceleration time histories used as input to the seismic SSI analyses of RB are developed by spectral matching seed ground motion records to the ground motion design response spectra presented in Figure 3.3-14. Per the guidelines of NEDO-33914 Revision 2, Section 5.2.3, five sets of three design motion time histories, in the two horizontal and in the vertical directions, are developed for the design to mitigate uncertainties due to the phasing of the time history frequency components.

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Time histories are developed by fitting recorded seed time histories to the 5% damped target design spectra to meet the requirements of Clause 4.4.4 of CSA N289.3 and Section 5.2.3 of NEDO-33914 Revision 2.

Per the recommendations of NEDO-33914 Revision 2, seed time histories are selected from the NUREG/CR-6728 database of ground motion records. The selected seed time histories include records with different magnitude and distance bins that have spectral shapes reasonably consistent with the spectral shape of the design target spectrum over the frequency range of interest and characteristics that reasonably represent the earthquake motions expected at the site. Since only a limited number of records for moderate and larger magnitude earthquakes are available for the Central and Eastern United States in the NUREG/CR-6728 database, transformed records from the Western United States are used. The transformation of these time records is performed to modify the spectra to correspond to Central and Eastern United States site conditions while preserving the realistic phase and amplitude relationships of the original records. Based on the DNNP PSHA deaggregated seismic hazard results, the selection of seed time records considered multiple bins for rock seed time histories, including records from magnitude 6 to 7 earthquakes at distances of 10 to 50 km, and the magnitude 7+ earthquakes at 10 to 50 km, 50 to 100 km, and 100 to 200 km.

Table 3.3-5 provides details of the selected five sets of time history records used for the development of the design time histories for SSI analyses of DNNP BWRX-300 RB. The five selected time histories are all from the 1999 Chi-Chi Taiwan earthquake (magnitude 7.6) that had a reverse fault mechanism that is appropriate for eastern North America. These time history records had sampling rates (Δt) of 0.005 seconds, with a Nyquist frequency of 100 Hz, and were typically longer duration recordings. Records from the shorter distances of 10 to 50 km and 50 to 100 km better matched the shape of the bounding ground motion response spectra once scaled to match the target spectrum at 100 Hz. The magnitude 7+ earthquakes at shorter distances than the scenario earthquakes (e.g., 10 to 50 km) are consistent with the target ground motion response spectra that represent an UB estimate of the seismic hazard. Smaller magnitude earthquakes were not selected because of a deficit of low frequency energy and the need for larger scaling factors. Table 3.3-5 provides the scaling factors applied to the time histories prior to spectral matching to better align the seed response spectrum shapes to the target spectra.

The spectral matching procedure is implemented for fitting the seed time histories to the 5% damped target spectra that retains the phase spectra of the seed time histories, preserving the relative phasing between horizontal and vertical components, as well as, preserving the non-stationarity and randomness characteristics. The modified time histories are checked as follows to ensure they meet the criteria specified in CSA N289.3, Clause 4.4 and ASCE/SEI 43, Section 2.4:

1. The 5% damped ARS of the modified seed time history are computed at a minimum of 100 points per frequency decade per CSA N289.3, Clause 4.4.4.3, uniformly spaced over the log frequency scale. The average of 5% damped ARS of the five Acceleration Time Histories (ATHs) are compared to the 5% damped target acceleration spectrum at each frequency point in the range of 0.1 Hz to 100 Hz to ensure that:
 - a. The average ARS does not fall below the target spectra by more than 10% at any frequency point
 - b. The average ARS does not fall below the target spectra at more than nine adjacent frequency points and 6% of the total number of points where the ARS is calculated satisfying the requirements of CSA N289.3, Clause 4.4.4.4.

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2. In accordance with Clause 4.4.4.5 of CSA N289.3, the power spectral density of the modified ground motion history is computed as described in ASCE/SEI 4, Section 2.6.2, and shown not to have significant gaps in energy at any frequency over this frequency range.
3. The total duration of time histories has to be no less than 15 seconds with minimum strong motion duration of 6 seconds per CSA N289.3, Clause 4.4.4.2 and long enough to provide an adequate representation of the Fourier components at low frequency.
4. Time histories used as input for the seismic response analyses have a strong motion duration, and ratios V/A and AD/V^2 (where A , V , and D , are the peak ground acceleration, velocity, and ground displacement, respectively) that are consistent with those of appropriate controlling events developed using the disaggregation data from in NK38-CORR-03611-0847339 (Reference 3.3-28)
5. The set of three modified ATHs representing the ground motion in the three orthogonal directions (two horizontal and one vertical) are statistically independent. Each pair of ground motion histories is considered statistically independent when the absolute value of their correlation coefficient does not exceed 0.16, satisfying the requirement of CSA N289.3, Clause 4.4.4.6.
6. The ATHs are baseline corrected to ensure the ground velocity converges to zero at the end of the earthquake record and maintains a zero-mean value over the time history duration.

Per recommendations of NEDO-33914 Revision 2, Section 5.2.3, the time step of the modified time histories is refined to 0.0025 seconds for the purposes of calculating high frequency in-structural responses, which exceeds the requirements of CSA N289.3, Clause 4.4.4.2.

Spectral matching of the seed time histories is completed using the time domain spectral matching procedure proposed by Lilhanand and Tseng (Reference 3.3-29) and later modified by Abrahamson (Reference 3.3-20) and Al Atik and Abrahamson (Reference 3.3-31). Figure 3.3-151, Figure 3.3-16, and Figure 3.3-17 present an example comparison of the original and spectrally matched time histories for the HWA026 records matched to the target rock design ground motion response spectrum. These plots demonstrate the non-stationary characteristics of the time histories are preserved. The most noticeable changes to the time histories are due to low frequency wavelets added at later portions of the time histories.

Response spectrum of the generated acceleration time histories are computed and compared to the appropriate target response spectra. A small scaling factor is applied to the time histories to increase the spectra and meet the design criteria. Finally, the cross-correlation coefficients, peak values, Arias Intensity, and Power Spectral Density function are computed for the spectrally matched time histories.

Figure 3.3-18 presents the normalized Arias Intensity, and the power spectral density function for the horizontal HWA026 components that are spectrally matched to the rock design ground motion response spectrum. Figure 3.3-19 presents the response spectra for spectrally matched horizontal and vertical components of record HWA026.

3.3.1.1.5 Percentage of Critical Damping

Consistent with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, damping values assigned to the structures and components in the SSI analysis model are in accordance with provisions of CSA N289.3, Clause 6.6, and ASCE/SEI 43, Section 3.3.3. The damping ratio values specified in Table 4(a) of CSA N289.3, Table 3-1 of ASCE/SEI 43, and U.S. NRC RG 1.61 (Reference 3.3-32) are used to represent the dissipation of energy in different elements. Consistent with the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, lower (Response Level 1) damping ratios are used for generating in-structure demands for qualification

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of equipment and systems. The higher (Response Level 2) damping values can be used for development of seismic demands for structural design per ASCE/SEI 43, Section 3.3.3 and U.S. NRC RG 1.61, Section C.1.2, respectively.

The damping properties assigned to soil materials in the SSI analysis model take into account the stress-strain properties corresponding to the level of seismic input per requirements of CSA N289.3, Clause 6.6.3. Stiffness and damping properties of subgrade materials compatible to the strains generated by design level earthquake event are developed based on results of Approach 1 SRA in Subsection 3.3.1.1. The strain-compatible damping of the subgrade materials is limited to 15% in accordance with the recommendations of ASCE/SEI 4, Section C5.2 and the regulatory guidance of U.S. NRC RG 1.208, Appendix E.

Table 3.3-6 lists damping values used in the seismic analysis of structures and components. These damping values are applicable to all modes of a structure or component constructed of the same material.

Damping values for subsystems including piping and equipment are obtained using the procedures described in Subsection 3.3.1.3.

3.3.1.1.6 Supporting Media for Seismic Category A Structures

Consistent with regulatory guidelines of CNSC REGDOC-2.5.2, Section 7.13.1, the input subgrade properties for the site-specific SSI analysis of the BWRX-300 integrated RB structure are based on the geological, seismological, and geotechnical investigations and take into account the random nature and inherent uncertainties of soil material properties.

In accordance with the regulatory guidelines of CNSC REGDOC-2.5.2, Section 7.13.1, the SSI analysis uses at least three sets of subgrade profiles representing BE, UB, and Lower Bound (LB) estimates of the subgrade material properties. These profiles are representative of the as-built conditions at the DNNP site. The LB and UB shear wave velocities and damping reflect a minimum coefficient of variation of each layer properties of $\pm 50\%$. In accordance with CSA N289.3, Clause 5.2.3, the design uses an envelope of results from the SSI analysis of BE, LB and UB subgrade profiles to account for the variation and uncertainty in subgrade properties.

The effects of primary non-linearity of subgrade materials response are addressed by using dynamic stiffness and damping properties which are compatible to the free-field strains induced by an DBE level seismic event.

The strain-compatible subgrade dynamic properties for the DNNP soil materials are calculated in accordance with the requirements of CSA N289.3, Clause 5.2 and ASCE/SEI 4, Section 2.4. These properties are developed at strain levels consistent with the estimated site PBSRS based on the results of the probabilistic SRA presented in Subsection 3.3.1.1.2. The strain-compatible subgrade dynamic properties are developed using the approach described in Appendix B of the Screening Prioritization and Implementation Details document (Reference 3.3-19) as follows:

1. Strain-compatible shear wave velocity and damping ratios are obtained consistent with the 1E-04 and 1E-05 MAPE from the results of SRA of the BE-BE, LR-BE, and UR-BE randomized soil profiles discussed in Subsection 3.3.1.1.2.
2. The logarithmic mean and logarithmic standard deviation of the strain-compatible shear wave velocity and damping ratios at 1E-04 and 1E-05 MAPE are calculated for the considered cases at each soil layer. The results from different soil cases are combined using weight factors of 0.4, 0.3, and 0.3 for the BE-BE, LR-BE, and UR-BE base cases, respectively. The LR and UR kappa base cases (e.g., BE-LR and BE-UR) are not considered given their small effects on site response analysis results when compared to the alternative cases for shear wave velocity. The weighted logarithmic mean and logarithmic standard deviations of the strain-compatible

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properties are calculated at 1E-04 and 1E-05 MAPE. The weighted average logarithmic mean and logarithmic standard deviation profiles for shear wave velocity and damping ratio at 1E-04 and 1E-05 MAPE are presented in Figure 3.3-20 through Figure 3.3-23, respectively.

3. The logarithmic mean and logarithmic standard deviation of shear wave velocity and damping ratio at strains that are compatible with the 100 Hz value of PBSRS are calculated by linear interpolation in the logarithmic space between those compatible with the 100 Hz values at 1E-04 and 1E-05 UHRS.
4. The exponential of the logarithmic mean profiles shear wave velocity profile calculated above is referred to as the median shear wave velocity and is selected as the 100 Hz BE shear wave velocity profile (V_{SBE}). The LB and UB shear wave velocity profiles are calculated as the 16th and 84th percentiles, respectively, using the following equations:

$$V_{SLB} = \min \left\{ e^{\ln(V_{SBE}) - \sigma}, \frac{V_{SBE}}{\sqrt{1.5}} \right\}$$

$$V_{SUB} = \max \left\{ e^{\ln(V_{SBE}) + \sigma}, V_{SBE} \times \sqrt{1.5} \right\}$$

where σ is the logarithmic standard deviation and the terms $V_{SBE} \times \sqrt{1.5}$ and $V_{SBE}/\sqrt{1.5}$ reflect the minimum variation requirement of $C_v = 0.5$ on the shear modulus as specified in CSA N289.3, Clause 5.2.3 to ensure that adequate uncertainty in the shear modulus of the soil profiles are included.

The 100 Hz strain-compatible LB, BE, and UB shear wave velocity profiles are presented in Figure 3.3-24.

5. The BE, LB, and UB profiles for damping ratio are calculated similar to step 4, except that no minimum variations of $C_v = 0.5$ are used, and the damping ratios are limited to a maximum of 15%, based on the recommendations of ASCE/SEI 4, Section C5.2 and regulatory guidance of U.S. NRC RG 1.208, Appendix E. Consistent with non-linear behavior of soil layers, the 16th percentile of damping ratio profile is associated with the UB profile and the 84th percentile of damping ratios are associated with the LB profile. For the linear rock layers, a damping ratio logarithmic standard deviation of 0.6 is adopted. The 100 Hz strain-compatible LB, BE, and UB damping ratio profiles are presented in Figure 3.3-24.

$$D_{LB} = e^{\ln(D_{BE}) + \sigma}$$

$$D_{UB} = e^{\ln(D_{BE}) - \sigma}$$

6. The BE, LB, and UB profiles considering the interpolation at 1 Hz are established using the same approach described in Steps 3, 4 and 5 above.
7. The final BE profiles are calculated as the average of the BE profiles considering the 100Hz interpolated values and 1 Hz interpolated values. Similarly, the final LB and UB profiles are calculated as the average of their corresponding profiles for the 100 Hz and 1 Hz interpolations.
8. The compression wave velocity profiles (V_p) are calculated using the final strain-compatible shear wave velocity profiles (V_s) obtained in Step 7 and the Poisson's ratios (ν) recommended for each layer using the following equation. Note that below-ground water table, the minimum of the compression wave velocity of water (1,463 m/sec) and the compression wave velocity corresponding to a maximum Poisson's ratio of 0.48 is used.

The latter criterion is adopted to avoid numerical problems in subsequent SSI analysis of the structure.

$$V_P = V_S \sqrt{\frac{2(1-\nu)}{1-2\nu}}$$

9. The P-wave damping values used as input to the SSI analysis are limited to a maximum of 10% at large strains for soil layers above the ground water table.

The development of dynamic subgrade profiles considers the soils located below the nominal groundwater table to be fully saturated. The groundwater level at elevation of 85 m CGD corresponding to a depth of 3 m below the plant grade at elevation 88 m CGD is considered as noted in Subsection 3.5.2.2. Figure 3.3-25 presents the strain-compatible shear wave velocity, compression wave velocity and damping ratio profiles used for the bounding design seismic analyses of BWRX-300 Seismic Category A structures discussed in Subsection 3.3.1.2.

3.3.1.2 Seismic Analysis of Seismic Category A Structures

This section discusses the seismic analysis of the Power Block Seismic Category A structures which consist of the RB, containment, and containment internal structures.

In accordance with CSA N289.3, Clause 6.2.3, the seismic demands for the design of the BWRX-300 Seismic Category A and Seismic Category B SSC are obtained from the seismic response analyses of the Seismic Category A structures that consider:

- Effects of interactions of the structures and the foundations with the surrounding subgrade
- Variation in the soil and structural parameters
- Hydrodynamic loads (mass and stiffness effects)
- Structure-Soil-Structure Interaction (SSSI) effects with the adjoining RWB, CB, TB, and Reactor Auxiliary Bay structures

Per Subsection 3.2.3, the BWRX-300 Seismic Category A and B SSC are hosted in the integrated RB structure, with the majority of them, including most of the Reactor Pressure Vessel (RPV) and the containment structure, being located below the plant grade elevation.

Because a significant part of the RB structure is located below grade, the interaction of the structure with the surrounding soil is a very important factor for the integrity of the RB structure, its seismic response, and the distribution of seismic stress demands.

In order to adequately account for the SSI and SSSI effects per guidance of NEDO-33914 Revision 2, Section 5.1, the one-step approach, as defined in Section 3.1.2 of ASCE/SEI 4, is implemented for the design of the integrated RB structure. Seismic structural stress demands are obtained directly from the results of SSI analyses of combined models that include 3-Dimensional (3-D) Finite Element (FE) representations of the integrated RB structure and the surrounding soil and Power Block structures. The surrounding subgrade is represented by layered half-space continuum with equivalent linear elastic stiffness properties and complex damping. Simplified FE models represent the dynamic properties of the surrounding Power Block structures and their foundations.

The methodology used for development of the 3-D integrated RB FE model is described in Subsection 3.3.1.2.2, and the SSI modeling assumptions are presented in Subsection 3.5.1.1.2.

3.3.1.2.1 Seismic Analysis Method

One-Step Seismic Analysis Method

Seismic demands for the design of Seismic Category A and B SSC are obtained from SSI analyses performed in accordance with the provisions of CSA N289.3, Clause 5.3, and ASCE/SEI 4, Section

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5, following the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, and U.S. NUREG-0800 (Reference 3.3-33), SRP 3.7.2.

The BWRX-300 one-step seismic SSI analysis approach provides demands for the seismic design and qualification of SSC for all frequencies of interest and adequately captures the effects of SSSI for the integrated RB with adjacent structures and foundations. The BWRX-300 seismic analysis approach follows the guidance of NEDO-33914 Revision 2, Section 5.0 to address current limitations in U.S. NUREG-0800 SRP 3.7.2 when capturing the effects of seismic interaction of the deeply embedded RB structure with adjacent structures through the subgrade, as identified in NUREG/CR-7193 (Reference 3.3-34), Section 1.5.11.

The seismic SSI analyses are performed using the sub-structuring method in CSA N289.3, Clause 5.3.5, and ASCE/SEI 4, Section 5.4 and the ACS SASSI (a system for analyses of soil-structure interaction, see Appendix 3B) computer program to calculate the seismic response of the RB SSI system. The SSI analysis model consists of the integrated RB structure, the surrounding subgrade and the excavated volume of the subgrade materials replaced by the embedded portion of the RB structure, near field backfill materials and the models representing the dynamic properties of the foundations and structures surrounding the RB.

The sub-structuring method allows the seismic response of the SSI system to be obtained by subdividing the problem into a series of simple subproblems that can be solved separately. Using the principle of superposition, the results of different sub-analyses are combined to obtain the final solution for the SSI problem. The solution for the seismic response of the BWRX-300 RB structure, is obtained in the frequency domain for a selected set of frequencies and then interpolated for other frequency points.

The linear elastic SASSI analyses are performed on one-step structural models that accurately represent the geometry and dynamic properties of the integrated RB structure and its interaction with the subgrade. These structural models have a refined FE mesh that is identical to the mesh of the models used for the static analyses, and that can transmit the entire frequency range of interest for the seismic design of the RB SSC. These models assume isotropic elastic material properties of structural members and surrounding subgrade and neglect any non-linearity at the soil-structure contact interfaces.

The linear elastic assumption allows a set of design and sensitivity SASSI one-step approach analyses to be performed on refined RB structural models with a large number of interaction nodes. The superposition principle, which is applicable only for linear elastic analyses, allows the SASSI stress results obtained from different dynamic and static analyses to be combined with the results of static analyses in seismic design load combinations.

Far-field interaction nodes are established at the surface of each soil layer through the RB shaft embedment depth to capture the horizontal and vertical components of the far-field motion in the SSI model. The responses calculated from these far-field interaction nodes are used to monitor the propagation of the input control motion through the RB embedment depth.

To account for the non-linear response of subgrade materials, strain-compatible subgrade properties are used that are developed based on the results of equivalent linear probabilistic SRA as described in Subsection 3.3.1.1. The uncertainties related to variation of soil and rock properties are addressed in the design of RB SSC by using seismic demands calculated as an envelope of the results obtained from SSI analysis cases of BE, LB, and UB subgrade dynamic profiles.

Input ground motion ATHs are applied to the SASSI model at the RB foundation bottom elevation as vertically propagating coherent:

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- Shear waves for horizontal components of the input motion
- Compression waves for the vertical component of the input motion

The horizontal control motion is applied to the SASSI model in a manner that is consistent with the 1-D wave propagation SRA approach discussed in Subsection 3.3.1.1.

As described in Subsection 3.3.1.1, five sets of three input motion ATHs are used as input for the SSI analyses to mitigate the uncertainty in the computed responses due to the phasing of the time history frequency components.

As described in Subsection 3.3.1.2.3, uncertainties related to variations of the input SSI parameters are addressed by results of sensitivity analyses following the recommendations in Section 5.3 of NEDO-33914 Revision 2.

Frequencies of Analysis

Following the guidance of CNSC REGDOC-2.5.2, Section 7.13.1, the frequency range considered in the seismic SSI analysis is based on the frequency content of the input ground motion, the soil properties, the building dynamic properties, including properties of the subsystems, and the response parameter of interest.

The solution for the response of the SSI system is obtained at a selected set of frequency points and then interpolated for other frequency points. The analysis is performed for a cut off frequency value established based on the largest value required by the following four criteria of ASCE/SEI 4, Section 5.3.5(b):

1. Twice the highest dominant frequency of the coupled soil-structure system or
2. The highest structural frequency of interest, or
3. The frequency at which the Fourier amplitude of input motion has passed its peak value and has reached 10% of the peak value, and
4. 20 Hz.

Criteria used to determine the highest dominant frequency and lower cutoff frequency values are described in Section 5.3.2 of NEDO-33914 Revision 2.

Sensitivity SSI analyses required to determine lower cutoff frequency values are performed for the stiffest UB subgrade profile that provides bounding responses at high frequencies.

The value of cutoff frequency determined by the criteria described above is used for the analysis of the UB subgrade profile. The analyses of the softer BE and LB profiles may use lower values for the cutoff frequency. In this case, it shall be demonstrated that the analysis of the UB profile provides responses that are bounding for frequencies higher than the cutoff frequencies used for the analyses of the softer subgrade profiles by comparing transfer function and 5% damped In-Structure Response Spectra (ISRS) results for responses at key locations within the building, selected as described in Subsection 3.3.1.2.5.

The frequencies of analysis are selected at sufficiently small frequency intervals. Transfer function amplitude results for responses at the key locations, selected as described in Subsection 3.3.1.2.5, are inspected to detect any numerical anomalies in the interpolated transfer functions (e.g., sharp narrow spikes) that can potentially affect the accuracy of results. If present, the effects of these anomalies in the interpolated transfer function results are evaluated using additional frequencies of analysis to ensure the anomalies in the transfer function interpolations do not affect the accuracy of the calculated responses.

Acceleration transfer functions and 5% damped ARS are also calculated for the response of SSI model free-field interaction nodes to check the amplitude and frequency content of the in-column free-field motion throughout the RB embedment depth.

3.3.1.2.2 Procedures Used for Analytical Modeling

SSI analyses of the integrated RB structure, which is primarily constructed of Steel Bricks™ as described in Subsection 3.5.1, are performed on 3-D FE models that meet the structural modeling requirements of CSA N289.3, Clauses 5.3.2 and 6.2, and ASCE/SEI 4, Section 3.

In addition to the integrated RB structures, simplified models of the surrounding RWB, CB, TB, and Reactor Auxiliary Bay structures and their foundations are included in the model to capture the SSSI effects in the RB seismic design.

Dynamic Finite Element Modeling of Integrated RB Structure

In accordance with requirements of Clause 6.10.4 of CSA N291, U.S. NUREG-0800, SRP 3.7.2, Subsection III.3.D, and ASCE/SEI 4, Section 3.4.2, the integrated RB structural FE model represents all mass expected to be present at the time of the earthquake including mass due to:

- Weight of the structure
- Weight of permanent equipment
- Mass equivalent to floor load of 2.4 kPa for miscellaneous dead weights such as minor equipment, piping, and raceways
- Weight of building elements not represented in the structural model (e.g., secondary members, siding partitions)
- Expected live load, not less than 50% of the live load specified for the design
- At least 25% of the specified design snow loads

The dynamic FE model also includes the inertia associated with the hydrodynamic effects of the fluids contained in various pools inside the RB and tanks in the RWB. The hydrodynamic effects that consist of the impulsive and convective (or sloshing) components are considered in accordance with the requirements of Clause 6.9 of CSA N289.3, Section 3.6.3 of ASCE/SEI 4, and Chapter 5 of ACI 350.3 (Reference 3.3-35). The hydrodynamic mass is included in the model by.

- Distributing the horizontal impulsive fluid mass over the pool and tank walls that are perpendicular to the direction of motion in accordance with the guidelines in ACI 350.3
- Lumping the entire vertical fluid mass on the pool slab or tank bottom.

The convective (sloshing) component of the hydrodynamic mass is not explicitly included in the global analysis model since its contribution is small and is associated with very low frequencies insignificant for the overall response. To account for the sloshing hydrodynamic effects, the design considers quasi-static sloshing pressure loads applied on the pool and tank walls in accordance with Section 9.4 of ASCE/SEI 4.

Beam and shell elements are used to adequately represent the configuration of all main structural members in the integrated RB. The FE model includes gross discontinuities such as large openings and member eccentricity. Thick shell elements are used to model the Steel Bricks™ shear walls, slabs, and mat foundation. 3-D beam elements are used to model the steel columns, beams, and trusses. The shell and beam elements are established at the centreline of the wall, slab, beam, column, and truss elements. Rigid beam and shell elements are used to model

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member eccentricities and offsets or the section properties of the centreline modeled elements are appropriately adjusted to account for the effect of member offsets.

Local spring elements represent the stiffness of the connections between different structural members, such as the connections of the SCCV with the internal structures, RB walls and slabs that are designed to relief stresses due to thermal expansion.

Contact springs with stiffness properties appropriate to capture the interaction at the soil-structure interface connect the RB structural and subgrade FE models. The results obtained from the contact spring elements serve to:

- Calculate dynamic earth pressures on the below grade RB shaft exterior wall and basemat and
- Determine whether separation between RB shaft wall and soils occurs under DBE loading as discussed in Subsection 3.3.1.2.4.

The evaluation of effects of conditions at the contact interfaces with surrounding subgrade on the RB seismic response is discussed in Subsection 3.3.1.2.4.

The values of Young's modulus and Poisson's ratio representing the structural material stiffness properties are determined in accordance with the governing design codes in Section 3.5. BE stiffness properties are assigned to the concrete made structures in accordance with ASCE/SEI 4, Section 3.3.2.

The effective stiffness for analysis for the thick shell elements representing Steel Bricks™ members is determined in accordance with guidelines in ANSI/AISC N690 (Reference 3.3-36), Appendix N9, or equivalent guidelines that reflect the expected behavior of the structural components during the applicable loads. These guidelines are the same as those in NEDC-33926P (Reference 3.3-37), the licensing topical report providing design requirements for steel-plate composite containment vessel. The stiffness calculations account for the expected state of stress and level of cracking for different loading conditions during normal operation and accident conditions. An effective in-plane shear stiffness determined from ANSI/AISC N690 code Equation A- N9-12, may be used if seismic load is considered in combination with accident thermal loading.

ANSI/AISC N690, Equation A-N9-8 is used to calculate the effective flexural stiffness of Steel Bricks™ members based on the cracked transformed section, which accounts for stiffness from the steel faceplates as well as the cracked concrete infill. This equation is also used to account for reduction of flexural stiffness due to additional concrete cracking due to conditions related to accident thermal loading. The additional reduction in flexural stiffness due to accident thermal can be ignored for operating thermal conditions where thermal gradients are small and develop over longer periods of time.

For structural components whose behavior is controlled by membrane behavior, the effective stiffness for analysis for applicable loading conditions includes considerations to realistically represent the membrane stiffness calculated in accordance with industry accepted guidelines.

The effects of variation of structural stiffness and damping properties is considered in the modeling of the integrated RB structure to ensure accuracy of the calculated seismic responses and seismic demands. Section 5.3.5 in NEDO-33914 Revision 2 describes methods used and sensitivity analyses performed to evaluate possible amplifications of in-structure responses and load demands on the members due to the load redistribution effects.

The FE models used for seismic SSI analyses have a sufficiently refined mesh to be capable of transmitting the entire frequency range of interest for the seismic design of the RB SSC. In accordance with the requirements of ASCE\SEI 4, Section 5.3.4, the FE mesh is smaller than or

equal to one-fifth of the smallest wavelength transmitted through the soil model, i.e., the maximum mesh size:

$$d_{max} \leq \frac{V_s}{5 f_{cutoff}}$$

where: V_s is the shear wave velocity of the transmitting soil material; and

f_{cutoff} is the cutoff frequency of analysis determined as described in Subsection 3.3.1.2.1

Consistent with requirements of CSA N289.3, Clause 5.3.4.4, the integrated RB FE model is sufficiently refined to ensure:

- Accuracy of SSI solution and ability to capture modes of vibrations up to frequencies that are important for the design
- SSI model can accurately transmit seismic waves with frequencies equal or higher than the cutoff frequency of analysis

Finer meshes are used around penetrations and openings that are larger than half of the wall or slab thickness. Meshes of major walls and slabs consists of at least four shell elements along the short direction and at least six shell elements along the long direction.

The lower boundary of the SSI model is established at a distance that is deeper than at least two times the depth of the RB embedment and at least three times the largest foundation dimension from the bottom of the slab in accordance with requirements of CSA N289.3, Clause 5.3.4.3.

Dynamic Modeling of Subsystems, Components and Equipment

The dynamic properties of subsystems, components, and equipment are included in the integrated RB structural model based on the decoupling criteria of CSA N289.3, Clause 6.3, and ASCE/SEI 4, Section 3.7, depending on the ratios of the mass and first natural frequency of the subsystem, component, or equipment to those of the supporting structure. To capture the dynamic coupling effects of the RPV, the dynamic properties of the RPV and its components are represented by a Lumped Mass Stick (LMS) model capable of capturing all significant modes of the RPV seismic response. Procedures used to develop this LMS model are presented in Subsection 3.3.1.3. The RPV LMS model is connected to the RB structural model using local spring elements, representing the stiffness of the RPV support skirt and the horizontal stabilizers.

3.3.1.2.3 Seismic SSI Analyses Results and Comparison of Seismic Responses

Key Seismic Responses

Responses at key nodal locations are calculated to check the accuracy of the SSI analysis and to evaluate seismic responses and effects of variations of different SSI parameters. These key locations are selected based on the following criteria:

1. Nodes at intersections of main structural members (main structural walls) at ground and other major floor elevations to illustrate global responses that exclude possible local effects due to out-of-plane vibrations of slabs and walls, openings or connections with columns, beams or subsystem supports.
2. At least two roof nodes, one central and one corner node, to show all important modes of seismic response of structure including the effects of rocking and torsion.

3. At least two basemat nodes, one central and one corner node, to show the SSI effects on the translational as well as the rotational (rocking and torsion) responses of foundation.

The seismic demands on the below grade portion of the RB structure are affected by the deformations resulting from the response of the SSI system. Therefore, besides the in-structural responses, main stress demand components, such as in-plane shear force and vertical bending moment demands, are also compared to be able to gain a complete understanding of the effects of SSI parameters variations on the structural design. These comparisons are performed for the main below grade structural members at selected design cross-sections subjected to high seismic stress demands.

Seismic SSI Analyses Results

Refer to Appendix 9B.B in Chapter 9B for results obtained from the Seismic SSI Analyses of BWRX-300 Seismic Category A structures.

3.3.1.2.4 Seismic Soil-Structure Interaction Parameters

The following are key requirements and approaches considered in the seismic SSI analyses to ensure the structural integrity and stability of the deeply embedded BWRX-300 RB structure throughout the life of the plant and to address specifics related to its design and construction.

Implementation of ISG-017 Guidance

BWRX-300 approaches for meeting U.S. NRC DC/COL-ISG-017 guidance and addressing current limitations in DC/COL-ISG-017 related to the seismic analysis of deeply embedded structures, as identified in NUREG/CR-7193, Section 1.5.8 are described in NEDO-33914 Revision 2, Section 5.3.4.

The intent of U.S. NRC DC/COL-ISG-017 is to ensure that the deterministic SSI analysis of the embedded RB structure uses ground motion inputs that are hazard consistent with the results of probabilistic SRA at the foundation bottom elevation and at ground surface.

The consistency between free-field motion at the bottom of the RB foundation used as input for the deterministic SSI analysis and probabilistic SRA is checked as described in Subsection 3.3.1.1, using the procedure described in Section 5.3.4.1 of NEDO-33914 Revision 2.

The augmented and smoothed horizontal and vertical 5% damped spectra presented in Figure 3.3-14 define the amplitude and frequency content of the SSI input control motion applied to the SSI model at the RB foundation bottom that is hazard consistent with the results of the probabilistic SRA described in Subsection 3.3.1.1.

Coupling of Soil and Structures

The seismic SSSI of the RB with the adjacent RWB, CB, TB, and Reactor Auxiliary Bay is explicitly considered in the seismic analysis and design.

Simple FE models representing the BE dynamic properties of the surrounding buildings and foundations are included in the integrated RB FE model used for the seismic SSI analysis. These simple models are sufficiently refined to capture all global modes of vibration of the RWB, CB, TB and Reactor Auxiliary Bay structures with significant ($> 20\%$) modal mass participations in the three orthogonal directions.

Subsection 3.3.1.2.8 presents the approach for addressing the requirements related to the seismic interaction of the RB with the surrounding RWB, CB, TB, and Reactor Auxiliary Bay structures and foundations.

3.3.1.2.5 Effects of Parameter Variation on Responses

This section covers the effects of concrete cracking, excavation support and backfill, groundwater variation, soil separation, non-vertically propagating seismic waves and soil secondary non-linearity on the seismic response and design of the BWRX-300 RB. The evaluations are performed in accordance with the requirements of ASCE/SEI 4, Section 5.1, following the guidelines of NEDO-33914 Revision 2, Section 5.3. They are based on comparisons of key in-structure responses, defined in Subsection 3.3.1.2.5, obtained from sensitivity SSI analyses as described below.

Effects of Variation of Structural Stiffness and Damping Properties

Effective structural stiffness and damping properties developed as discussed in Subsections 3.3.1.2.2 and 3.3.1.2.3 are assigned to the SSI model following the recommendations in Section 5.3.5 of NEDO-33914 Revision 2. Effective stiffness assigned to concrete members takes into account the level of stress in the concrete members due to the most critical seismic load combinations.

To address the effects of structural stiffness variations, sensitivity SSI analyses are performed on models representing lower structural stiffness properties corresponding to accident thermal and high intensity load conditions. Higher Response Level 2 damping properties may be used for the analysis of the model with LB structural stiffness.

These sensitivity analyses are performed for BE subgrade profile to evaluate the significance of the structural stiffness variations on the RB in-structure responses and redistribution of load demands on the structural members. The effects of structural stiffness variations are assessed by comparing key in-structure responses, defined in Subsection 3.3.1.2.5, of the two sensitivity analyses of models with reduced stiffness properties with results of the design basis analysis performed on the model with effective stiffness properties.

Excavation Support and Backfill Effects

Excavation support and backfill effects are to be addressed following the guidelines of NEDO-33914 Revision 2, Section 5.3.8. Sensitivity seismic SSI analyses are to be performed using BE properties of surrounding in-situ subgrade materials on a RB FE model that includes the excavation support structure and the fill concrete to assess their effect on the BWRX-300 RB seismic response. Shell and beam elements are to be used to represent the BE dynamic properties of the excavation support structure. Solid elements are to be used to represent BE, and the dynamic properties of concrete fill material. The geometry of the excavation support and the lean concrete are to be modeled based on the nominal dimensions obtained from excavation plan drawings. To address the uncertainties related to the modeling of friction at the RB shaft interfaces, the sensitivity SSI analyses are performed considering two bounding conditions:

- A. Fully bonded conditions assuming no slippage between the RB shaft and surrounding materials
- B. No-friction conditions assuming no friction resistance of RB shaft exterior walls

Results of these sensitivity analyses for key in-structure responses, defined in Subsection 3.3.1.2.5, are compared with the corresponding results of the design basis SSI analyses of FE model that excludes the excavation support and the fill concrete. If the comparisons show significant exceedances ($> 10\%$) in the RB seismic response due to the interaction with the excavation support and fill concrete, the results of these sensitivity analyses are included in the RB seismic design basis.

Groundwater Variation Effects

The potential effects of groundwater level variability on the seismic design of the BWRX-300 RB are addressed as described in Section 5.3.10 in NEDO-33914 Revision 2.

The seismic design of RB is based on analysis of SSI models that reflect fully saturated conditions for all soil materials located below the nominal groundwater elevation. The potential effects of groundwater level variability on the seismic design are addressed by comparing the seismic responses obtained from two sensitivity analyses of:

- A. Fully saturated soil profile with BE soil dynamic properties representative of accidental flood groundwater level
- B. Dry soil profile with BE soil dynamic properties representative of the extreme conditions when the groundwater is located below the RB foundation bottom elevation

Results of these two sensitivity analyses for key in-structure responses, defined in Subsection 3.3.1.2.5, are compared with the results of the design bases SSI analyses based on fully saturated soil profiles below the nominal groundwater elevation. If the comparisons show that the effects of groundwater variation significantly exceed (>10%) the design basis, the results of the two sensitivity analyses are included in the RB seismic design basis.

Soil Separation Effects

The SSI analysis of the BWRX-300 RB addresses the uncertainties related to the inability of linear models used for the seismic design SSI analysis to explicitly represent the separation between the soil and the structure in accordance with the guidance of ASCE/SEI 4, Section 5.1.9(b).

The approach described in Section 5.3.9 of NEDO-33914 Revision 2 is followed to determine if the separation at soil-structure interfaces can have significant effect on the seismic response. A sensitivity SSI analysis is performed on a model where portions of the below grade shaft wall that may experience separation from the subgrade soil are assumed to remain unbonded for the total duration of the earthquake. The extent of soil separation is assessed by comparing the maximum lateral earth pressure calculated from the seismic SSI analysis of BE subgrade profile with a LB estimates of static earth pressures. The static lateral pressures calculated from static design SSI analysis with 1-g loading, described in Subsection 3.5.2.4, are reduced by 10% to account for uncertainties in calculation of soil unit weights and surcharge loads. The regions where the static lateral pressure is lower than the seismic lateral pressure are considered separated in the model used for the sensitivity analysis.

The key in-structure responses, defined in Subsection 3.3.1.2.5, and stress demands calculated from this sensitivity analysis are compared to the corresponding results of the SSI analysis of the model with BE properties representing fully bonded conditions. If the comparisons indicate that the seismic in-structure responses and stress demands from the fully separated model exceed those obtained from the SSI analysis of fully bonded models by more than 10%, the results of this sensitivity analysis are included in the RB seismic design basis.

Effects of Non-Vertically Propagating Seismic Waves

The potential for non-vertically propagating seismic waves at the DNNP site is to be assessed following the guidelines in Section 5.3.3 of NEDO-33914 Revision 2 based on the geological and seismological conditions of the site. The available site information does not indicate presence of dipping soil and rock layers or local seismic sources that can result in significant non-vertical seismic wave propagation at the DNNP site

3.3.1.2.6 Three Components of Design Ground Motion

Earthquake motion is three-dimensional and seismic design takes into account the effects of three orthogonal components (two horizontal and one vertical) of the prescribed design earthquake.

The SSI analyses are performed separately for each of the three directional components of input ground motion using five sets of time histories per Subsection 3.3.1.1. For each set of time histories used as analysis input, the seismic response parameters obtained from the analysis of each of the three ground motion components are combined to get the total co-directional response with either of the three methods permitted under ASCE/SEI 4, Section 4.2.2.

1. The time histories of responses due to the three earthquake components are combined algebraically on the time-step-by-time-step approach.
2. The maximum co-directional responses can be combined using the 100-40-40 method.
3. The maximum responses due to the three earthquake components can be combined using the Square-Root-of-the Sum of the Squares (SRSS) method.

The absolute sum method used in time domain may also be implemented (e.g., for calculations of seismic demands for foundation bearing pressure and stability evaluations) as a conservative alternative to performing the algebraic sum method for all possible combinations of the input motion directions.

3.3.1.2.7 Development of In-Structure Responses

ISRS and ATHs are developed from the seismic analysis to serve as input for the seismic design and evaluation of subsystems, components, and equipment.

In-Structure Response Spectra

The ISRS for the seismic design and evaluation of subsystem, components, and equipment are developed in accordance with the requirements of CSA N289.3, Clause 6.5.2.3 and ASCE/SEI 4, Section 6.2.

A set of ISRS are developed for required damping levels defining the amplitude and frequency content of in-structure design motion at different locations within the RB, in the two horizontal and the vertical directions for seismic qualification of substructures, systems, and components.

The ISRS for the design of subsystems for which dynamic properties are included in the global dynamic model using LMS models, are developed as an envelope of responses at the node locations where these LMS models are connected to the supporting structure provided that, per ASCE/SEI 4, Section 3.7.1(d), the LMS model adequately represents the major effects of interaction between the equipment and supporting structure.

The ISRS for the seismic design and evaluation of subsystems that are decoupled from the global model, and which location is known, are developed as an envelope of responses at the perimeter of the support footprint area to capture the effects of in-structure rotations. If the equipment or component is supported by flexible slabs or attached to flexible walls, ISRS are developed considering additional nodal responses that capture the local effects of out-of-plane vibrations of the supporting slab or wall.

If the LMS models are used to model the structure, substructure, or subsystem in the global dynamic model, the ISRS are developed as envelope of the responses of outrigger nodes located at the edges of the structure or subsystem.

In accordance with the requirements of ASCE/SEI 4, Section 6.2.1.1(a) and (b), the ISRS are developed from the calculated nodal in-structure responses by:

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1. First combining in the time domain the three co-direction responses due to the three orthogonal components of seismic input motion as an algebraic sum at each time step and then calculating the ARS of the combined ATHs, or
2. Combining the co-directional ARS results obtained from the analysis with the three orthogonal components of seismic input motion using the SRSS method specified in Subsection 3.3.1.2.6.

The spectra are calculated for frequencies ranging from 0.1 Hz to the highest frequency of interest meeting the requirements specified in Table 2 of CSA N289.3. In addition, the ISRS are developed at small frequency intervals to ensure they are sufficiently close to the peak response frequencies of the supporting structure. To satisfy this requirement, the ISRS are calculated at 301 frequency points equally distributed on the logarithmic scale at the frequency range from 0.1 Hz to 100 Hz.

The ISRS are calculated as an envelope of the results from the seismic design basis SSI analysis of all subgrade profiles. In accordance with the requirements of Clause 6.5.2.3 of CSA N289.3, the peaks of the enveloping ARS are broadened by a minimum of +/-15% to address uncertainties related to the modeling of natural frequencies of the supporting structure and the SSI analysis methodology. The sharp valleys between peaks are filled to account for the uncertainties in subgrade properties.

In-Structure Acceleration Time Histories

In accordance with the requirements of ASCE/SEI 4, Section 6.3, time histories used in the analysis of subsystems are obtained either:

- Directly from the results of the SSI analysis as time histories of nodal responses at reference of subsystem support locations; or
- By generating synthetic time histories compatible to multi-damping ISRS developed as described above.

When obtained directly from the SSI analysis results:

- Time histories of the co-directional in-structure responses due to the three components of the SSI analysis input motion are combined in the time domain
- Time histories are obtained from SSI analysis cases that are critical for the designed subsystem and include those obtained from BE soil case
- Time histories obtained from the BE soil case only can be modified by using time-shifting factors to address uncertainties related to the modeling of natural frequencies of supporting structure

Relative Displacements

Relative Displacement between different support points of subsystems with multiple or distributed supports are evaluated using displacement time histories.

The time history of the relative displacements corresponding to each SSI analysis is obtained by algebraic calculation of the different displacement time histories at the support locations. Directional combination of the support displacement time histories is carried on a time-step-by-time-step basis. Maximum design relative displacements are calculated as an envelope of the maximum relative displacements obtained for each SSI analysis case.

3.3.1.2.8 Seismic Interaction Evaluation

Consistent with CNSC REGDOC-2.5.2, Section 7.13.1, the BWRX-300 design ensures the ability of the RWB, CB, TB, and Reactor Auxiliary Bay to prevent adverse interactions with the Seismic category A and B SSC during a DBE event.

To meet the interaction requirements in Subsection 3.2.3.1, evaluations are performed of the lateral load resisting system of the RWB, CB, TB, and Reactor Auxiliary Bay structures following the approach in NEDO-33914 Revision 2, Section 6.2. These evaluations are based on seismic responses of RWB, CB, TB, and Reactor Auxiliary Bay obtained from the SSSI analyses that incorporate the dynamic response of the RB and surrounding Power Block structures. As described in Subsection 3.3.1.2.2, models used in the SSI analyses of the RB include FE representations of the surrounding RWB, CB, TB, and Reactor Auxiliary Bay structures and foundations. The FE models of the RWB, CB, TB, and Reactor Auxiliary Bay are refined sufficiently to provide accurate stress demands on the major lateral load resisting structural members and accurate seismic displacements in the direction of the adjacent RB.

The seismic interaction evaluations consider limited permanent deformations (LS-C) structural response to calculate DBE demands for the main lateral load resisting structural members in accordance with the guidance of NEDO-33914 Revision 2, Section 6.2.

The stability of RWB, CB, TB, and Reactor Auxiliary Bay foundations is checked following criteria in Subsection 3.5.2.2 using demands calculated per Subsection 3.3.1.2.10. No reductions are applied to seismic driving force demands used for the stability evaluations to account for inelastic responses of these structures.

The resistance to sliding is calculated as summation of the effective cohesion and static frictional resistance between foundation and subgrade. The frictional resistance is based on the effective weight of the building and includes the buoyancy and seismic loads in the vertical direction. The lateral passive resistance of the foundation embedment soil is also considered, as applicable.

The overturning stability evaluation is performed for each orthogonal horizontal axis of the building using the overturning demands calculated per Subsection 3.3.1.2.10 and the restoring moments calculated using the effective weight of the building. The energy method described in BC-TOP-4A (Reference 3.3-38) can be used for overturning stability evaluation, where factors of safety against overturning are calculated by comparing the maximum kinetic energy driving the system to overturning during a seismic event with the potential energy required to prevent overturning of the structure and foundation. For this approach, the minimum overturning factor of safety of 1.25 is used, consistent with CSA N289.3.

The gaps between the RB and adjacent structures are evaluated per guidance in NEDO-33914 Revision 2, Section 6.2, to ensure no physical interaction between the RB structure and surrounding structures. The gaps are evaluated along the entire height of the adjacent structures considering construction tolerances, inelastic deformations, and possible differential settlements.

3.3.1.2.9 Methods to Account for Torsion

Considerations are given in the modeling of the integrated RB structure to represent the actual locations of the centre of masses and centres of rigidity of structural elements to account for torsional effects.

In accordance with the requirements of ASCE/SEI 4, Section 3.1, the seismic design of the RB structure also considers accidental torsion to account for:

- Non-vertically propagating seismic waves
- Rotational components of ground motion

- Possible distributions of structural mass and stiffness that differ from those represented in the 3-D FE model used for the seismic response analysis per the requirements in Clause 6.10 of CSA N289.3

Accidental torsional moment demands may be calculated at each floor level as the product of the story shear and 5% of the floor plan dimension perpendicular to the story shear direction. Alternatively, the horizontal shear force demands on all walls may be conservatively increased by 5% to account for the accident torsion.

3.3.1.2.10 *Determination of Seismic Overturning Movement, Sliding Forces and Dynamic Bearing Pressures*

Contact spring elements installed in the SSI models at interfaces between the structure and the subgrade are used for calculation of seismic driving forces and overturning moments on the BWRX-300 foundations. As described in Subsection 3.3.1.2.6, time histories of the horizontal and vertical seismic forces in the three directions are calculated as the algebraic sum of the spring forces in the three directions at each step for all contact spring elements. Overturning moments about the two horizontal axes are calculated as the algebraic sum of the moments resulting from each spring force with respect to the foundation bottom centreline. Conservatively, the spring force results for calculation of seismic driving force demands may be combined using the absolute sum time domain method instead of using the algebraic sum method for all possible combinations of the input motion directions.

The seismic inertia forces and overturning moments for the foundation stability evaluations and seismic bearing pressure calculations are obtained from SSI models with higher (Response Level 2) structural damping values.

Seismic stability of the surface mounted foundations surrounding the RB are evaluated by calculating safety factors for seismic sliding and overturning stability for each time step. These safety factors are calculated for the total duration of each of the five sets of ATHs described in Subsection 3.3.1.1. The average value of the minimum safety factors obtained from the five sets of ATHs is used to demonstrate the seismic stability criteria described in Subsection 3.5.2.2 are met.

The seismic bearing pressure demands are also calculated in the time domain. Maximum bearing pressure values are calculated for the total duration of earthquake for each of the five sets of ATHs used as input for the SSI analysis discussed in Subsection 3.3.1.1. The dynamic bearing pressure demand under each foundation is defined as the average of the results obtained from the five sets of ATHs.

3.3.1.3 *Seismic Analysis of Seismic Category A and B Subsystems*

This section applies to the Seismic Category A and Seismic Category B subsystems. Input motions for the qualification of these systems are usually in the form of floor response spectra or ATHs obtained from the primary system dynamic analysis discussed in Subsection 3.3.1.2. Input motions in terms of acceleration time histories are generally used. Dynamic qualification can be performed by analysis, testing, or a combination of both, or by the use of experience data. This section addresses the aspects related to analysis only.

3.3.1.3.1 *Seismic Analysis Methods*

Seismic analysis of subsystems can be performed using one of the following methods:

- Time History Analysis
- Response Spectrum Analysis

- Static Coefficient

The time history and the response spectrum methods are utilized in the piping analysis as required. The procedure for multi-support excitation described in Subsection 3.3.1.3.9 is followed with both methods. When the multi-support Response Spectrum Method is used to calculate the dynamic response of the piping system, all multi-support response spectra components are simultaneously applied to each piping model for each load case.

The time history and Response Spectrum Methods are also utilized in the equipment analysis as required. When the equipment is supported at two or more points located at different elevations in the building, the response spectrum for the most severe single point of attachment is chosen as the design spectra. Alternatively, the multi-support excitation procedure described in Subsection 3.3.1.3.9 is used.

Vertical analyses of the RPV and internals are performed using in-structure responses obtained from the results of one-step analyses of the RB discussed in Subsection 3.3.1.2.

RPV and internal components such as fuel, guide tubes, and Control Rod Drive (CRD) System housing are included in the integrated RB model as discussed in Subsections 3.3.1.2 and 3.3.1.3.3. As a result, the evaluation of RPV internals components in the horizontal direction is performed using a Two-Step analysis approach, where seismic loads are applied to more detailed horizontal beam models of the RPV and internals. The first step of the Two-Step analysis consists, therefore, of obtaining ATHs or ISRS developed as described in Subsection 3.3.1.2 at the RPV/RB interface locations from the RB SSI analyses discussed in Subsection 3.3.1.2. The second step is a multi-support excitation time history analysis of the RPV, and internals subjected to the ATHs generated in the first step. The procedure for multi-support excitation time history analysis, as described in Subsection 3.3.1.3.9, is followed in the second step analysis of the RPV and internals.

Time History Analysis

Assuming velocity proportional damping, the dynamic equilibrium equations for a lumped mass, distributed stiffness system are expressed in matrix form as:

$$[M] \{ \ddot{u}(t) \} + [C] \{ \dot{u}(t) \} + [K] \{ u(t) \} = \{ P(t) \}$$

Where:

- $\{ u(t) \}$ = time dependent displacement of nonsupport points relative to the supports.
- $\{ \dot{u}(t) \}$ = time dependent velocity of nonsupport points relative to the supports.
- $\{ \ddot{u}(t) \}$ = time dependent acceleration of nonsupport points relative to the supports.
- $[M]$ = mass matrix.
- $[C]$ = damping matrix.
- $[K]$ = stiffness matrix.
- $\{ P(t) \}$ = time dependent applied force column vector.

The above equation can be solved by modal superposition or direct integration in the time domain.

Modal Superposition involves two steps. First, the characteristic equation corresponding to undamped, free vibration of the model is solved to obtain the eigenvalues, eigenvectors, and generalized masses. The system coupled equations are then decoupled via the eigenvector transformation matrix which is simply the matrix of eigenvectors written as columns. The equations are decoupled in the generalized coordinate system because of the orthogonality of the matrix of eigenvectors with respect to the “weighted” mass and stiffness matrices. The decoupled modal

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equations are then solved independently to obtain the generalized coordinates. The physical solution is then given by the eigen transformation once the generalized coordinates are known.

The direct integration method involves the numerical integration of the simultaneous differential equations of equilibrium in their original form, without transformation to the generalized coordinates. For systems subjected to short duration, high frequency excitation (such as those due to LOCA acoustic, blast and jet loads), the direct integration method requires less computation and is recommended over the modal superposition method.

For the time domain solution, the numerical integration time step is sufficiently small to accurately define the dynamic excitation and to render stability and convergence of the solution up to the highest frequency (or shortest period) of significance. This condition is satisfied if Δt is selected to limit the amplitude decay per cycle of free vibration of the highest significant mode to less than 20 percent. This corresponds to approximately 3.5 percent numerical damping for that highest significant mode. The integration time step for both the direct numerical integration of the system coupled equations of motion and the numerical integration of the n decoupled equations (Modal Superposition) satisfies the following requirement:

$$\Delta t \leq T_m/10$$

where Δt is the numerical integration time step magnitude and T_m is the period of the highest significant mode considered in the analysis or the reciprocal of the cutoff frequency in Hz as defined in Subsection 3.3.1.3.4.

Response Spectrum Analysis

This method is used if only peak dynamic responses are required.

The response spectrum method is a modal superposition analysis in which only the peak values of the solution of the decoupled modal equations are obtained. The method is based on writing the solution of each decoupled modal equation in terms of the convolution integral. The major advantage of this form of solution is that for a given input motion the only variables under the integral are the damping factor and the frequency. Thus, for a specified damping factor, it is possible to construct a curve which gives the maximum value of the integral as a function of frequency. This curve is called a response spectrum for the particular input motion and the specified damping factor. The integral has units of velocity, consequently the maximum of the integral is called the spectral velocity.

For a subsystem analysis of a secondary system the input floor response spectra, obtained from a time history analysis of the primary system, is broadened ± 15 percent to account for modeling uncertainties in both the primary and secondary systems in accordance with ASCE/SEI 4, Section 6.2.3.

Using the calculated natural frequencies of vibration of the system, the maximum values of the modal responses are determined directly from the appropriate response spectrum. The modal maxima are then combined as discussed in Subsection 3.3.1.2.

Static Coefficient

The static coefficient method may be applied to certain equipment in lieu of the required dynamic analysis. Response loads are determined statically by multiplying the equipment mass by a static coefficient equal to 1.5 times the maximum spectral acceleration that corresponds to the first mode of the equipment. This coefficient is intended to account for the effect of both multi-frequency excitation and multi-mode response. This method is applicable only to equipment corresponding

to a simple column, beam, or frame type structure supported at a single point. Justification is required for applying this method or coefficient to equipment having configurations other than simple frame or beam type structures.

A factor of less than 1.5 may also be used if adequate justification is provided. For example, if the equipment is simple enough such that it behaves essentially as a single degree-of-freedom model and is greater than the seismic excitation frequency, the factor 1.0 can be used instead of 1.5.

If the fundamental frequency of the equipment is greater than the cutoff frequency but less than the Zero Period Acceleration (ZPA) frequency, the static coefficient can be taken as 1.5 times the peak spectral acceleration which occurs between the cutoff frequency and the ZPA frequency in the equipment input response spectra.

3.3.1.3.2 Determination of Number of Earthquake Cycles

The BWRX-300 Seismic Category A and Seismic Category B SSC are seismically qualified to withstand the effects of the DBE defined in Subsection 3.3.1.1. RW-IIa SSC are seismically qualified for one-half (1/2) of this DBE as stated in Table 3.3-1.

The determination of the number of earthquake cycles for subsystem analysis is in accordance with U.S. NUREG-0800, SRP 3.7.3.

3.3.1.3.3 Procedures Used for Analytical Modeling

The mathematical model for each Seismic Category A and B component to be analyzed is prepared to realistically reflect the dynamic characteristics of that component. Each component is discretized into a series of interconnected beam elements or finite elements. The node points are generally selected to coincide with the locations of large masses, such as at structure floors or at heavy equipment supports, and at all points corresponding to any significant change in physical geometry.

The number of mass node points in the model is sufficient if additional node points (independent of number) do not result in more than 10 percent increase in the responses in the frequency range below the cutoff frequency specified in Subsection 3.3.1.3.4.

The node point spacing is selected such that the maximum length L of the finite element between any two node points, in the direction of the stress wave propagation, satisfies the condition

$$L \leq \frac{\lambda}{4} = \frac{v}{4f} = \frac{vT}{4}$$

where: λ and v are the wavelength and wave velocity, respectively.

The frequency f , or period T , correspond to the cutoff frequency of Subsection 3.3.1.3.4.

Modeling of Equipment

For dynamic analysis, Seismic Category A and B equipment is represented by lumped mass systems which consist of discrete masses connected by weightless beam elements and/or by any other appropriate finite element representation. The criteria used to lump the masses are:

- A. The number of modes of dynamic system is controlled by the number of masses used. Therefore, the number of masses is chosen so that all significant modes are included. The modes are considered as significant if the corresponding natural frequencies are less than the cutoff frequency specified in Subsection 3.3.1.3.4.
- B. Mass is lumped at any point where a significant concentrated weight is located.

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- C. For equipment with a free-end overhang span whose flexibility is significant compared to the centre span, a mass is lumped at the overhang span.
- D. When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to conservatively lower the natural frequencies of the equipment. Similarly, in the case of live loads and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen to yield the lowest frequency content for the system.

Modeling of Piping Systems

Mathematical models for Category A and B piping systems are constructed to realistically reflect the dynamic characteristics of the system. The continuous system is modeled as an assemblage of pipe elements (straight sections, elbows, and bends) supported by hangers and anchors, and restrained by pipe guides, struts, and snubbers. Pipe and fluid masses are lumped at the nodes and connected by the weightless elastic beam elements which reflect the physical properties of the corresponding piping segment. The mass node points are selected to coincide with the locations of large masses, such as valves, pumps, and motors, and with locations of significant geometry change. All concentrated weights on the piping system, such as the valves, pumps, and motors, are modeled as lumped mass rigid systems if their fundamental frequencies are greater than the cutoff frequency in Subsection 3.3.1.3.4. The torsional effects of valve operators and other equipment with off-set centre of gravity with respect to the piping centreline are included in the analytical model. The pipe length between mass points is no greater than the length with a fundamental frequency equal to the cutoff frequency stipulated in Subsection 3.3.1.3.4 when calculated as a simply supported beam with uniformly distributed mass.

Branch lines with a run to branch moment of inertia ratio of 25 to 1 or greater are excluded from the piping model of the main line in accordance with CSA N289.3.

All pipe guides and snubbers are modeled to produce representative stiffness to reduce model uncertainties. Snubbers are modeled with an equivalent stiffness based on dynamic tests or on data provided from the vendor. The stiffness of the supporting structures is included in the analysis unless the supporting structure is shown to be rigid.

Modeling of Reactor Pressure Vessel and Internals

Because of the significant dynamic interaction between the RB and RPV and internals, the latter are integrated into the RB model as discussed in Subsection 3.3.1.2.

The mathematical model of the RPV and internals consists of a LMS model connected by linear elastic members and 3D finite element models. Using the elastic properties of the structural components, the stiffness properties of the model are determined. This includes the effects of both bending and shear.

To facilitate hydrodynamic mass calculations, mass points (e.g., representing the fuel, shroud, vessel) are selected at the same elevation. The various lengths of CRD housings are grouped into two representative lengths. These lengths represent the longest and shortest housings to adequately represent the full range of frequency response of the housings. In order to reduce the complexity of the dynamic model, the light components (such as in-core guide tubes and housing, sparger, and their supply headers) are excluded from the RPV mathematical model. However, the dynamic response of selected components is determined from a subsystem analysis after the system response is found.

Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix, which serves to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped inside the RPV vessel. Although the

dynamic coupling between the vertical hydrodynamic masses is not considered, the vertical hydrodynamic masses themselves are properly accounted for. Dynamic loads due to vertical motion are added to, or subtracted from, the static weight of component, whichever is more conservative.

The shroud support plate is modeled as a rigid link in the translational direction since it is loaded in its own plane during a horizontal dynamic event. The shroud support legs, and the local flexibilities of the vessel and shroud contribute to the rotational flexibilities and are modeled as an equivalent torsional spring.

Due to the small clearances in the horizontal directions, the fuel assembly is adequately modeled as a linear system for subsystem and system analysis. In the vertical direction, the fuel assembly has the potential to lift off from its seat and a non-linear representation is required if the vertical applied and reaction forces are sufficient to cause fuel lift. Furthermore, the interface between the fuel channel and lower plate tie plate is not rigid and a non-linear model to account for slippage may be appropriate.

The weight of asymmetric secondary components, such as attached equipment, is uniformly redistributed around the node point circle. Asymmetric equipment is modeled using finite element or LMS methods.

3.3.1.3.4 Basis of Selection of Frequencies

The cutoff frequency selected in the time history and response spectrum analyses ensures that all significant modes are included in the superposition. Higher modes which cumulatively contribute less than 10% of the total system response are not considered in the superposition of the individual modal values.

The cutoff frequency for seismic and other dynamic loads follows Subsection 3.3.1.2. For seismic load, it is estimated that all modes up to 100 Hz are included.

For all other dynamic analysis, it is estimated that the cutoff frequency will be 100 Hz, as long as no more than 5 percent of the total strain energy of the system remains beyond this cutoff frequency.

Where practical, to avoid adverse resonance effects, equipment and components are designed/selected such that their fundamental frequencies are approximately less than half or more than twice the dominant frequencies of the support structure. Moreover, in any case, the equipment is analyzed or tested or both to demonstrate that it is adequately designed for the applicable loads considering both its fundamental frequency and the forcing frequency of the applicable support structure.

3.3.1.3.5 Analysis Procedure for Damping

Damping of Primary Subsystems

Primary Subsystems consist of the RPV and internals.

Damping values for seismic analysis of primary subsystems using the Modal Superposition are presented in Table 3.3-7. These damping values are in accordance with ASCE/SEI 43 and CSA N289.3.

α , β –damping curves for the axis-symmetric finite element analysis of primary subsystems completed by Direct integration are defined per Table 3.3-8 and the following equation:

$$\lambda = \frac{\alpha}{2\omega} + \frac{\beta\omega}{2}$$

Damping values for dynamic loading beam analysis, performed by modal superposition, are identical to those for DBE provided in Table 3.3-7.

Damping of Secondary Subsystems

Damping coefficients used in the seismic analysis of Seismic Category A and B piping, equipment, equipment supports and intermediate structures between subsystems are presented in Table 3.3-9.

Damping coefficients used for all other non-seismic loads are presented in Table 3.3-10.

These damping values are in accordance with ASCE/SEI 43 and CSA N289.3.

3.3.1.3.6 Three Components of Design Ground Motion

Applicable methods for spatial combination of responses due to each of the three input motion components are described in Subsection 3.3.1.2.

3.3.1.3.7 Combination of Modal Responses

Applicable methods for combination of modal responses are described in Subsection 3.3.1.3.1.

3.3.1.3.8 Interaction of Other Subsystems with Seismic Category A and B SSC

Non-Seismic Category systems are designed to be isolated from Seismic Category A and B systems by either a constraint or barrier or are remotely located with regard to the Seismic Category A and B systems.

If it is not feasible or practical to isolate the Seismic Category A or B system, adjacent Non-Seismic Category systems are analyzed according to the same seismic criteria as applicable to the Seismic Category A and B systems. Consistent with the approach used for evaluation of structures discussed in Subsection 3.3.1.2, limited inelastic deformation responses LS-C are considered for the seismic interaction evaluations of equipment by using inelastic absorption factors per ASCE/SEI 43, Section 8.2.2.2, and Table 8-1. For Non-Seismic Category systems attached to Seismic Category A and B systems, the dynamic effects of the Non-Seismic Category systems are simulated in the modeling of the Seismic Category A or B system. The attached Non-Seismic Category systems, up to the first anchor beyond the interface, are also designed in such a manner that during DBE level event it does not cause failure of the Seismic Category A or B system.

3.3.1.3.9 Multiply Supported Equipment and Components with Distinct Inputs

This section discusses the analytical method used for obtaining multi-support loadings and for dynamically analyzing Category A and B systems with multiple supports (or one support with many excitations), with different dynamic excitations. This analytical method is in accordance with CSA N289.3.

The time history Direct Integration, time history Modal Superposition and Response Spectrum Modal Superposition methods discussed in Subsection 3.3.1.3.1 can all be used in Multi-Support Excitation analysis. However, the mode superposition procedure described in Section 3.3.1.3.1 for an applied load vector is replaced with the corresponding mode superposition procedure for multi-support excitation analysis.

When using the time history method, the following methods are acceptable:

- A. The time histories corresponding to the envelopes of the ISRS for all attachment points in each of the three directions are applied at each attachment point simultaneously.
- B. The time histories corresponding to the envelopes of the ISRS for each attachment point in each of the three directions are applied at each corresponding attachment point simultaneously.

The above time history methods of analysis are performed such that primary (inertial) and secondary (static stresses due to differential displacements) are separated. The inertial forces are used for primary stress calculations. Secondary stresses are first computed for each natural mode of the supporting structures and for each excitation direction. The total secondary stress for triaxial excitation is then computed as the SRSS of the resultant secondary stresses for each excitation direction. The ASME BPVC Code Section III requires that the secondary stresses must be combined with the primary stress.

The inertia (primary) and displacement (secondary) stresses are dynamic in nature and their peak values are not expected to occur at the same time. Hence combination of the peak values of inertia stress and anchor displacement stress using the SRSS method is quite conservative. In addition, anchor movement effects are computed from static analyses in which the displacement are applied to produce the most conservative loads on the components.

Using the response spectrum method, support points response spectra are generated from support point acceleration time histories. In accordance with the requirements of Clause 6.5.2.3 of CSA N289.3, ± 15 percent peak broadening is applied to the spectra to account for the RB support structure modeling uncertainties. In general, using the SRSS method to combine modal responses is conservative since the maximum modal responses due to each component of multi-support excitation do not occur simultaneously. For certain “closely spaced” support with highly correlated support excitations, the SRSS superposition may yield unconservative responses. In this case, the modal responses of the “closely correlated” supports are combined algebraically first. Then, correlated sums are combined with the contributions for uncorrelated supports using the SRSS method.

3.3.1.3.10 Use of Equivalent Vertical Static Factors

Equivalent vertical static factors are used when the requirements for the static coefficient method in Subsection 3.3.1.3.1 are satisfied.

3.3.1.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are considered in the modeling of subsystems as discussed in Subsection 3.3.1.3.3.

3.3.1.3.12 Effects of Differential Building Movements

In most cases, subsystems are anchored and restrained to floors and walls of buildings that may have differential movements during a seismic event.

Differential endpoint or restraint deflections cause forces and moments to be induced in the system. As discussed in Subsection 3.3.1.3.9, the stress thus produced is a secondary stress. It is justifiable to place this stress, which results from restraint of free-end displacement of the system, in the secondary stress category because the stresses are self-limiting and, when the stresses exceed yield strength, minor distortions or deformations within the system satisfy the condition which caused the stress to occur.

Refer to Subsection 3.3.1.2 for the methodology used to obtain differential displacements used in the evaluation of subsystems.

3.3.1.4 Seismic Analysis of Other Subsystems

Seismic demands for the evaluation of other subsystems are developed based on ISRS, ATHs and relative displacements calculated with the Response Level 1 structural damping values in accordance with CNSC REGDOC-2.5.2, Section 7.13.1. The use of models with higher (Response Level 2) damping values can be justified based on the level of stress response as applicable to these structures.

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Per Clause 6.5.2 of CSA N289.3, the seismic input at support points for the dynamic analysis of decoupled subsystems are ISRS or time histories representing the in-structure design translational motion in the two horizontal and the vertical directions due to the three components of the input earthquake motion.

If the in-structure rotations are significant, rotational ISRS and ATHs are developed and used for the design of the decoupled subsystems. Relative displacements between different support points of subsystems with multiple or distributed supports are also considered in the evaluation.

3.3.1.5 Seismic Instrumentation

In accordance with the requirements in CNSC REGDOC-1.1.2 (Reference 3.3-39), Section 4.5.6, and CNSC REGDOC-2.5.2, Section 7.13.1, seismic instrumentation is used to monitor the seismic activity at the site for the lifecycle of the reactor facility, starting from commissioning, including outages, until fully decommissioned.

The design of BWRX-300 seismic instrumentation satisfies the more stringent requirements for large reactors in CSA N289.5, Clause 5 in addition to Clauses 1 to 3 and 8 to 10.

The handling of seismic instrumentation system data records is in accordance with requirements of Clause 10 of CSA N289.5. When required, the seismic instrumentation requirements of CSA N289.5 are augmented by the requirements of U.S. NRC RG 1.12 (Reference 3.3-40).

The required actions after an earthquake follow the provisions of CSA N289.1.

3.3.1.5.1 Location and Description of Instrumentation

Free-Field Instrumentation

In accordance with the requirements of Clause 5.2.2 of CSA N289.5, at least two triaxial accelerometers are installed outside of the structure-ground interaction influence of the Power Block, but as close as practicable to the reactor to monitor the free-field ground motion at the BWRX-300 site at the plant grade and close to the RB bottom elevations.

In accordance with U.S. NRC RG 1.12, Section C.1.2, because the deeply embedded RB is founded at a depth more than 12 m below finished grade elevation, installation of a second free-field downhole accelerometer is considered at the bottom of the RB foundation, below the free-field accelerometer at finished grade level.

Structure and Equipment Instrumentation

In accordance with the requirements of Clause 5.2.3.1.2 of CSA N289.5 and Section C.1.2 of U.S. NRC RG 1.12, triaxial accelerometers are installed at several locations inside the RB including:

- One at the top of the mat foundation
- One on the containment internal structure close to the reactor vessel
- One close to the top of the containment internal structure
- One close to the top of the containment structure
- One at the operating floor elevation

Also, in accordance with Clause 5.2.3.1.3 of CSA N289.5, three additional triaxial accelerometers are installed outside of the RB, either at locations of seismically qualified SSC or at other locations that are deemed important.

The specific locations for instrumentation are determined to obtain the most pertinent information consistent with the selected key locations in the RB model to enable easy comparison between

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the measured and calculated in-structure responses. The sensors are installed such that occupational radiation exposures associated with their location, installation, and maintenance are maintained as low as is reasonably achievable.

Structure and equipment instrumentation stations recording are configured to be accessible for maintenance during full-power operation in compliance with the guidance of U.S. NRC RG 1.12. For sensors installed in inaccessible areas, provisions for data recording and an external remote alarm indicating actuation are provided.

Recording and Playback Equipment

Recording and playback units are provided for multiple channel recording and playback of the triaxial accelerometer signals. Characteristics and installation requirements of the recording and playback equipment follow the guidelines in U.S. NRC RG 1.12.

Accelerometers can measure acceleration amplitudes of at least 2g in accordance with Clause 5.1.6.1 of CSA N289.5.

Power Sources

In accordance with Clause 5.1.7.2 of CSA N289.5, a dedicated standby power source is provided for the seismic instrumentation. This backup power source can provide a minimum of 6 hours of continuous operation of any accelerometer or a minimum of 24 hours of continuous operation of any accelerograph in the event of failure of all external power sources.

The central unit of the seismic instrumentation system incorporates a self-contained seismically qualified standby power source dedicated for providing the system a minimum of 6 hours of continuous operation in the event of failure of all external power sources.

3.3.1.5.2 Design and Installation

In accordance with the requirements of Clause 8 of CSA N289.5, all components of the seismic instrumentation system and their supports are designed and installed to maintain their structural integrity, and to remain operational during and following a DBE. Accessibility for servicing and recalibration, anchorage and protection from adverse conditions that can affect their performance are also considered in the design.

Prior to the installation, the operational reliability of the seismic monitoring instrumentation is demonstrated, in accordance with Section C.4.7 of RG 1.12, by using prototype, environmental, vibratory, or historical test results.

3.3.1.5.3 Maintenance and Testing

Maintenance and testing of seismic instrumentation are defined in accordance with the requirements in Clause 9 of CSA N289.5, documented before the first facility startup, and updated as necessary following any modification to the system. All components of the seismic instrumentation system are maintained and tested to ensure that a maximum number of instruments are kept in-service during plant operation and shutdown.

The operability of each of the seismic instrumentations is demonstrated by performing channel checks every two weeks for the first three months of service after startup. After the initial three-month period and three consecutive successful checks, the channel checks are performed on a monthly basis. The channel calibrations are performed every 24 months or during each refueling outage. The channel functional test is performed every 6 months. At least once a year, the system is operated continuously on the standby power source to verify the required backup power availability per CSA N289.5, Clause 9.2.2.

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The guidance of Appendix A to U.S. NRC RG 1.166 (Reference 3.3-41) is followed for instrumentation found to be out of service during an earthquake.

3.3.1.5.4 Arrangements for Control Room Operator Notification

In accordance with the guidance of U.S. NRC RG 1.12, Section C.4.13, the triaxial accelerograph system is triggered whenever a threshold free-field acceleration of not more than 0.01 g is exceeded for any of the three axes. A higher threshold value can be used if 0.01 g is impracticable due to the site geological or geotechnical conditions or the ambient noise at instrument locations.

Activation of the seismic trigger causes an audible and visual annunciation in the control rooms to alert the plant operator that a felt earthquake has occurred in accordance with Clause 5.1.3 of CSA N289.5. Authorities having jurisdiction as well as the local and regional emergency response agencies are advised of the plant status if an earthquake exceeds the threshold acceleration per Clause 6.5.4 of CSA N289.1.

3.3.1.5.5 Comparison of Measured and Predicted Responses

The appropriate response after a felt seismic event is determined by the level of shaking. In accordance with Clause 6.5.1 of CSA N289.1, the BWRX-300 post-seismic plant operation manual defines the response associated with each level of shaking. The required operator actions after a felt earthquake are in accordance with Clause 6.5.7 of CSA N289.1.

Per Clause 6.5.5 of CSA N289.5, an immediate shutdown of the plant is not mandatory if during and following an earthquake the plant continues successful operation. The plant is shut down if it is determined that the earthquake intensity exceeded the DBE or if there is evidence of damage impacting the safety systems.

In the event of a plant trip, all records pertaining to fuel and reactor internals systems are compared to the data that are recorded during a normal shutdown and/or previous plant trips. The intensity of the earthquake and any evidence of damage will dictate if a detailed inspection is required or if a restart is allowed. Prior to startup, the availability of all safety class SSC is confirmed to ensure they can perform their intended functions.

Immediate Response Following a Seismic Event

If the plant remains online following a seismic event, the immediate response is to stabilize the plant in accordance with Clause 6.5.7.1.1 of CSA N289.1 by:

- Testing all systems required to perform nuclear safety functions
- Initiating inspections performed in accordance with the provisions of ANSI/ANS-2.23 (Reference 3.3-42) to assess the intensity of the seismic events and the effects on essential systems

Recorded earthquake data from the seismic instrumentation, coupled with information obtained from a plant walkdown, are used to make the initial determination of whether the plant should be shut down, if it has not already been shut down by operational perturbations resulting from the seismic event.

Seismic Design Basis Exceedance

Following a seismic event, records of free-field ground motion and in-structure responses are reviewed in accordance with Clause 6.5.6.1 of CSA N289.1.

Cumulative absolute velocity calculated in accordance with Section 6.4.1 of ANSI/ANS-2.23 and peak ground velocity are generated from all free-field ground motion to be used as damage

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indicators. Damage criteria for Heavy industrial SSC in Section 6.5.6.2.1 of CSA N289.1 are also considered to help determine seismic design basis exceedance.

The DBE is considered exceeded when the measured free-field motion in any of the three directions (two horizontal and one vertical) exceeds the following limits:

1. Response spectrum limit that is exceeded if:
 - a. At frequencies between 2 and 10 Hz, the recorded response spectral accelerations of 5% damping exceed the corresponding DBE design acceleration response spectrum or 0.2 g, whichever is greater or
 - b. At frequencies between 1 and 2 Hz, the recorded response spectral velocities of 5% damping exceed DBE velocity response spectrum or 152 mm/sec, whichever is greater
2. Cumulative absolute velocity limit that is exceeded if the cumulative absolute velocity value calculated in accordance with Clause 6.5.6.1 of CSA N289.1 is greater than 0.16 g-s, or the peak ground velocity is greater than 50 mm/s.

The DBE exceedance is checked for measurements taken from the free-field plant grade accelerometers and downhole accelerometers using the corresponding design response spectra defining the DBE ground motion at the plant grade and RB foundation bottom elevations.

In addition to the criteria above, the following is also used to determine DBE exceedance:

- The inspection of the seismically qualified SSC shows evidence of overstressing, large displacement, yielded supports, etc.
- If the data collected from the monitoring instruments installed at different elevations in the plant exceed the DBE response parameters at the corresponding locations

Required Pre-Shutdown Earthquake Actions

Prior to the shutdown, the availability of safety class systems required for shutdown and the availability and integrity of the containment system are confirmed by performing pre-shutdown checks in accordance with the provisions of CSA N289.1, Clause 6.5.7.2.

Post-Shutdown Earthquake Response Actions

While the plant is shut down, a detailed inspection and evaluations are performed to assess the state of the plant in accordance with the provisions of CSA N289.1, Clause 6.5.7.3.

Post-shutdown actions include:

- Focused inspections of a preselected set of SSC that are representative of a broad cross section of equipment and structures in nuclear and conventional power plants
- Expanded inspections if damage is found in focused inspections
- Further graded inspections, tests, and analyses that are guided by the damage and earthquake levels

Focused inspections include detailed, visual inspections and tests of a preselected sample of representative structures and equipment, selected to sample all types of safety class and SCN SSC that are considered most likely to be damaged due to earthquake shaking. SCN SSC that experience has shown to be of low seismic capacity to serve as earthquake damage indicators are also included in the focused inspections.

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Expanded inspections and tests are performed if significant physical or functional damage is found during the focused post-shutdown inspections. The expanded inspections include all accessible safety class equipment and structures as well as non-safety-class balance-of plant equipment that is important to safe operation of the plant. Expanded inspections and tests may not be performed if the damage observed as part of the focused inspections is isolated to a specific class of SSC and if the cause of the damage is attributable to a specific design or installation deficiency, such as lack of equipment anchorage, improper installation of expansion bolts, etc. In this case, the design or installation deficiency is corrected for all SSC in the classes involved, and inspections of other undamaged classes may not need to be expanded.

If damage to safety class SSC is observed, the reactor vessel is opened, and reactor vessel internals and fuel are inspected using methods normally employed for in-service inspections.

If the DBE is reached, the plant restart is only allowed after ensuring that the allowable design stresses of seismically qualified SSC are not exceeded.

Results of post-shutdown inspections and tests are documented and reported to the authorities having jurisdiction. Results of inspections are compared with results of previous baseline inspections.

3.3.2 Extreme Weather Conditions

This section presents the design basis weather conditions considered in the design of the BWRX-300 SSC for the bounding extreme meteorological hazards identified in Chapter 2, Section 2.6.

3.3.2.1 Temperature and Humidity

The extreme temperatures and humidity levels specified in Chapter 2, Table 2.6-1 are considered in the BWRX-300 design in accordance with CNSC REGDOC-2.5.2, Sections 7.4.2 and 7.15.1. Conservative safety margins are considered in the evaluations and design of SSC to ensure their availability and efficiency under extreme temperature and humidity conditions.

3.3.2.2 Rain

Rain load is considered in the design of the BWRX-300 building structures.

The RB roof is designed to minimize or eliminate rain loading in accordance with U.S. NRC RG 1.102 (Reference 3.3-43), regulatory position 3, considering rain intensity and duration (PMP) values listed in Chapter 2, Table 2.6-1.

Design for rain loading on the RWB roof is performed in accordance with CSA N291 Clause 6.2, considering PMP values specified in Chapter 2, Table 2.6-1.

The design of the remaining Power Block roofs to minimize and evaluate the potential of ponding follows the guidance in the NBC, Section 4.1.6.4.

3.3.2.3 Snow and Ice

The RB structure is designed using ground snow loads for normal and extreme winter precipitation events of 2.5 kPa and 5.0 kPa, respectively. These loads envelop those used in the design of the nearby Darlington Nuclear Generating Station listed in Chapter 2, Subsection 2.6.9. For the RB structure, ground snow loads are converted to roof snow loading in accordance with the methodology specified in the ASCE/SEI 7 (Reference 3.3-44) referenced in U.S. NRC DC/COL-ISG-7 (Reference 3.3-45).

For the RB structure, the normal roof snow load is considered as a normal live load for all normal operating load combinations considered in the design. The extreme roof snow load is considered as an extreme load for the extreme environmental combinations (See Chapter 9B, Table 9B-4), without concurrent seismic or tornado loads.

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For the RWB design, snow load (including snow drifting conditions, as applicable) is computed in accordance with the methodology specified in CSA N291, Clause 6.3 and NBC, and based on 100 years occurrence specified in Chapter 2, Table 2.6-1.

For the design of other Non-Seismic Category Power Block structures, the design snow load is determined in accordance with the methodology specified in NBC considering 50 years recurrence. The Importance Factor for Snow, I_s , assigned to these structures is based on Table 4.1.6.2-A of NBC for Post-Disaster importance category.

3.3.2.4 Wind

In accordance with REGODOC-2.5.2, Section 7.15.1, wind loads are considered in the design of the BWRX-300 building structures and components.

Site-specific wind speeds for the RB structure are translated into structural loading in accordance with the methodology specified in ANSI/AISC N690. The RB is designed as an ASCE/SEI 7 (referenced in ANSI/AISC N690), Risk Category IV structure (3000-year return period), for severe wind load of 257.5 km/h with 3-second gust basic wind speed that is bounding the site-specific design basis wind speed values in Chapter 2, Table 2.6-1.

Wind loads for the design of the RWB are determined in accordance with the methodology specified in CSA N291, Clause 6.3 and NBC, and based on 100 years return period wind pressure specified in Chapter 2, Table 2.6-1.

Wind loads for the design of other Non-Seismic Category Power Block structures are determined in accordance with the methodology specified in the NBC, Section 4.1.7. The reference wind speed is based on 50-year return period one-hour mean reference design wind. The Importance Factor for Wind, I_w , assigned to these structures is based on Table 4.1.7.3 of NBC for Post-Disaster importance category.

3.3.2.5 Tornado

In accordance with CNSC REGDOC-2.5.2, Section 7.15.1, tornado loads are considered in the design of BWRX-300 building structures and components based on their pertinent Seismic Category listed in Table 3.3-1.

Tornado loads included in the design of the Seismic Category A RB structure include:

- Tornado wind pressures
- Differential pressure loads due to rapid atmospheric pressure change
- Tornado-generated missile impact

The design input tornado wind parameters and tornado missile spectrum applicable to the Seismic Category A RB structure are provided in Chapter 9B, Table 9B.9-2 and Table 9B.9-3. These parameters are based on Region I values from U.S. NRC RG 1.76 (Reference 3.3-47). These values bound the DNNP site-specific parameters listed in Chapter 2, Table 2.6-5, and Table 2.6-6.

The RW-IIa RWB which houses rooms and equipment for handling, processing, and packaging liquid and solid radioactive wastes is designed for the site-specific tornado wind and missile spectrum modified per the requirements of Table 2 of RG 1.143.

The RWB, CB, TB, and Reactor Auxiliary Bay are evaluated for the design basis tornado wind loads applicable for the RB so that their interaction with the RB does not adversely affect the ability of the Seismic Category A and B SSC to perform their safety functions. The interaction evaluation follows the guidance of NEDO-33914 Revision 2, Section 6.3.

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The structural integrity of the CB is maintained in the event of a design basis tornado missile to allow egress of operators to the Secondary Control Room (SCR) in the RB and to ensure availability of SSC providing post-disaster mitigation functions. For the special hardening provisions considered in the design of the CB, refer to Chapter 9B, Section 9B.3.2.2.

For a discussion of tornado dampers used to protect the Heating, Ventilation and Air Conditioning (HVAC) openings in the RB and CB to improve their survivability under tornado, refer to Chapter 9A, Section 9A.5.

The procedures for transforming tornado wind speed into pressure-induced forces to apply to structures and the distribution across the structures are based on BC-TOP-3-A (Reference 3.3-46). U.S. NRC RG 1.76 provides guidance to determine the pressure drop and rate of pressure drop caused by the passage of a tornado.

Missiles created as a result of components and cladding failing during a tornado wind event are considered enveloped by the design basis missile spectrum considered for the RB.

3.3.2.6 Hurricanes

Hurricanes at the DNNP site are considered bounded by tornado loads discussed in Subsection 3.3.2.5.

3.3.2.7 Lightning

Complying with Section 7.4.2 of CNSC REGDOC-2.5.2, grounding and lightning protection systems are used to protect structures, transformers and equipment against lightning induced surges as described in Chapter 8, Section 8.6.

Protection measures against fires and electromagnetic compatibility issues that could affect the functionality of electrical systems as a result of lightning are addressed in Subsections 3.3.6 and 3.3.7.1.

3.3.2.8 Extreme Wind Interaction

As described in Subsection 3.3.2.5, evaluations are performed to ensure that there is no adverse interaction between the RWB, CB, TB and Reactor Auxiliary Bay and the RB under design basis tornado wind loads applicable for the RB.

3.3.3 Extreme Hydrological Conditions

Potential sources of external floods considered in the BWRX-300 design are discussed in Chapter 2, Subsection 2.5.3.

To conform with Section 7.4.2 of CNSC REGDOC-2.5.2 and in accordance with U.S. NRC RG 1.102, Seismic Category A and RW-IIa structures are designed to include protective features that are used to mitigate or eliminate the adverse consequences of flooding due to external sources.

Conforming with CNSC REGDOC-2.5.2, Section 7.15.1, the integrated RB structure is designed to withstand the maximum external flood and groundwater levels specified in Chapter 2, Section 2.5.3.1.

Protection measures considered for the integrated RB structure against underground water includes the use of:

1. Hydrostatic and hydrodynamic loads to design walls below flood level in conformance with CNSC REGDOC-2.5.2, Section 7.15.1
2. Suitable provisions to ensure water tightness of external surfaces and penetrations below design basis maximum flood and groundwater levels

3. No exterior access openings below grade

In accordance with U.S. NRC RG 1.143, the RWB is designed for one-half of the Probable Maximum Flood (PMF) listed in Chapter 2, Subsection 2.5.3.1.

Because plant grade is above design flood level, the Power Block structures remain accessible during postulated flood events. Thus, no emergency actions are required due to flooding to ensure the safe operation of the BWRX-300 plant.

3.3.3.1 Analysis Procedure

The BWRX-300 RB is analyzed and designed to withstand the effects of the maximum external flood and highest groundwater levels specified for the plant. The maximum flood and highest groundwater levels listed in Chapter 2, Subsection 2.5.3.1 are considered in defining the input design parameters for the structural design to account for flood and groundwater loadings.

Because the flood level at the DNNP site is below the finished grade level, only hydrostatic effects are considered in the analysis and design of structures, while dynamic phenomena associated with a flooding event, such as currents, wind waves, and their hydrodynamic effects are not considered. The hydrostatic pressure associated with the design flood level or with the design groundwater level is considered as a structural load on the basemat and basement walls for structural design in accordance with CNSC REGDOC-2.5.2, Sections 7.4.2 and 7.15.1. Uplift or floating of structures is considered and the total buoyancy force is based on the hydrostatic pressure due to the design flood level, excluding wave action, or the design groundwater level. The lateral, overturning and upward hydrostatic pressures acting on the side walls and on the foundation slab, respectively, are also considered in the structural design of these elements.

3.3.4 Aircraft Crash

This section discusses non-malevolent, general aviation crashes in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.4.2. For robustness against malevolent acts, including aircraft crashes, refer to Subsection 3.3.7.4.

Small aircraft crashes are considered in the BWRX-300 design but are screened out per Chapter 2, Subsection 2.2.3.1. The design considers these aircraft crashes as missiles bounded by the design basis tornado missiles discussed in Subsection 3.3.2.5.

To mitigate their potential of equipment damage and fire impacts, the design of the BWRX-300 Seismic Category A structures addresses penetration resistance of buildings and considers physical separation of redundant or backup equipment, where applicable.

3.3.5 Missiles

3.3.5.1 Missiles Generated by Extreme Winds

Refer to Subsection 3.3.2.5 for details.

3.3.5.2 Site Proximity Missiles (Except Aircraft)

The design considers site proximity missiles to be bounded by the design basis tornado missiles discussed in Subsection 3.3.2.5.

Due to the distance between the sites, the maximum turbine missile from the existing Darlington site does not impact the DNNP site.

3.3.5.3 Structures, Systems and Components to be Protected from Externally Generated Missiles

Seismic Category A, RW-IIa, and portions of the TB and CB structures are designed to withstand the effects of externally generated missiles. For Seismic Category A SSC, the tornado wind

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characteristics and tornado missile spectra considered in the design are listed in Chapter 2, Table 2.6-3 and Table 2.6-4. Tornado wind and tornado missile spectra design input values considered in the design of the RWB are listed in Table 2 of RG 1.143.

The response determination methodology due to missile impact loading on the RB structure, consisting of Steel Bricks™ modules, is in accordance with ANSI/AISC N690, Appendix N9.1, Section 6c.

The response determination methodology due to missile impact loading on the RWB and portions of the TB and CB is in accordance with CSA N291, Annex A.

3.3.5.4 Barrier Design Procedures

In accordance with CSA N291, Clause A.5, barrier design for impact loads satisfies the criteria for local and overall effect. The procedures for designing barriers to withstand the effects of missile impacts are per U.S. NUREG-0800, SRP 3.5.3.

3.3.5.4.1 Local Damage Prediction

The prediction for local damage in the impact area depends on the basic material of construction of the barrier.

Concrete Barriers

Sufficient thickness of concrete is provided to prevent perforation, spalling, or scabbing of the barriers in the event of missile impact.

Per CSA N291, Clause A.5.2.3, empirical formulas are applicable over a limited range of missile and target parameters.

Required concrete barrier thicknesses are determined in accordance with U.S. NUREG-0800, SRP 3.5.3 and are in no case less than those of Region I listed in Table 1 of U.S. NUREG-0800, SRP 3.5.3. In accordance with CSA N291, Clause A.5.2.4, the required barrier or wall thickness to prevent perforation is at least 20% greater than the calculated thickness from the applicable empirical formulas. Also, the required barrier or wall thickness to mitigate missile penetration is at least 50% greater than the calculated thickness from the applicable empirical formula.

Steel Barriers

Steel barrier thicknesses are determined using the Stanford equation (Reference 3.3-48) in accordance with the regulatory guidance of U.S. NUREG-0800, SRP 3.5.3.

Composite Sections

Composite section barriers are utilized in the BWRX-300 for missile protection when the residual velocity of the missile perforating the first element is considered as the striking velocity for the next element for prediction of local damage in accordance with the regulatory guidance of U.S. NUREG-0800, SRP 3.5.3.

3.3.5.4.2 Overall Damage Prediction

The BWRX-300 design for impactive loads satisfies the criteria for the overall effect of Clause A.5.3 of CSA N291. Dynamic effects of impactive loads are evaluated by dynamic analysis in accordance with Clause A.4.1.1 of CSA N291 or the equivalent static load approach mentioned in Clause A.4.1.2 of CSA N291.

3.3.5.4.3 External Doors

The RB external doors are designed to resist tornado missiles unless shielded by external stair towers or elevator shafts. External stair towers or elevator shafts credited for shielding are evaluated for tornado missiles.

3.3.6 External Fires, Explosions and Toxic Gases

In line with requirements of CNSC REGDOC-2.5.2, Section 7.4.2, damages due to fires, explosions, and release of toxic gases as a result of transportation and industrial accidents at or near the DNNP site are considered in the BWRX-300 design. The following subsections provide information on measures considered to protect and mitigate the effects of:

- External fires – Subsection 3.3.6.1
- Explosions – Subsection 3.3.6.2
- Release of toxic gases – Subsection 3.3.6.3

3.3.6.1 External Fires

Per Chapter 2, Subsections 2.2.3, 2.2.4, 2.4.1 and 2.6.10, sources of external fires at the DNNP site include fireballs as a result of a rail transportation accident, forest fires, lightning and accidental fires in on-site storage areas of hydrogen, liquid waste or fuel oil. As mentioned in Chapter 2, Subsection 2.2.4, the risk of fire due to pipeline ruptures close to the DNNP site is negligible and is therefore not considered in the design.

Chapter 9A, Section 9A.6 describes the BWRX-300 fire protection systems implemented to resist and mitigate the effects of external fires. Buildings and structures within the protected area are supplied fire water from redundant loops by two fire water storage tanks (See Chapter 9A, Section 9A.6.6) providing suction to fire pumps located in a Fire Pump Enclosure structure (See Chapter 9B, Section 9B.3.6).

Figure A1.4-1 in Appendix A of Chapter 1 shows the location of the fire water storage tanks and Fire Pump Enclosure at the DNNP site.

Protection measures against the release of toxic gases as a result of external fires are discussed in Subsection 3.3.6.3.

3.3.6.2 Explosions

The RB structure is designed to withstand impulsive and impactive loads as discussed in Subsection 3.5.5.4.

3.3.6.3 Release of Toxic Gases

On-site activities that could result in release of toxic gases that could impact the safe operation of the BWRX-300 DNNP are summarized in Chapter 2, Section 2.4. External sources of toxic gases and chemicals are discussed in Chapter 2, Section 2.2.

Mitigation measures considered in the design of MCR/SCR are referenced in Chapter 6, Section 6.4.

3.3.7 Other External Hazards

3.3.7.1 Electromagnetic Interference

Protection against electromagnetic interference caused by lightning, high-voltage transmission lines at DNGS and telecommunication towers (See Chapter 2, Subsection 2.2.9) is provided through the use of appropriate shielding and qualification of equipment.

Safety Class SSC are protected against electromagnetic interference to enable them to perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform.

For a description of plant grounding, lightning protection and electromagnetic compatibility systems and their design requirements, refer to Chapter 8, Section 8.6.

3.3.7.2 Biological Phenomena

In accordance with CNSC REGDOC-2.5.2, Section 7.4.2, the Pumphouse/forebay structure is designed to prevent clogging by algae and exceptional quantities of fish and to stop them from entering the cooling systems. Measures considered to mitigate the effects of such clogging include locating the intake tunnel and lakebed intake structure at an adequate depth in the lake and the installation of traveling water screens to prevent intake of biofouling material as described in Chapter 9B, Subsection 9B.3.5.

As shown in Chapter 1, Appendix A, Figure A1.4-1, the BWRX-300 protected area is fenced which, in turn, prevents entry of large animals into the plant.

Screens or equivalent engineered features are also provided to prevent blockage of outside air intakes by non-human biota.

3.3.7.3 Collisions of Floating Bodies and Frazil Ice with Water Intakes

To satisfy requirements in CNSC REGDOC-2.5.2, Section 7.4.2, the design of the intake structure includes measures to mitigate the potential risk of blockage by frazil ice accumulations and physical damages as a result of a marine accident.

Measures considered to preclude blockage by frazil ice include a proper design of the Circulating Water System (CWS) recirculation line to prevent the formation of frazil ice in the forebay. Refer to Chapter 10, Section 10.8 for information related to the CWS.

To prevent marine transportation accidents, a restricted zone is established around the BWRX-300 lakebed intake structure and discharge diffusers to stop commercial ships from approaching offshore structures as stated in Chapter 2, Subsection 2.2.3.4.

3.3.7.4 Robustness Against Malevolent Acts

The BWRX-300 design provides robust physical features for the protection against malevolent actions found in the Design Basis Threats (DBTs) and Beyond Design Basis Threats (BDBTs). This results in the following fundamental capabilities remaining available after malevolent actions intended to cause substantial radiological releases:

- Ability to shut down the reactor and maintain sub-criticality
- Ability to cool irradiated fuel, both in the core and in the fuel pool
- Ability to limit or prevent the release of radioactivity affecting public health and safety

The ultimate gauge of success of the above three key functions is the prevention of radioactive releases that impact the health and safety of the public.

The BWRX-300 development has included a security by design approach from the early stages of design that uses sound engineering principles to demonstrate that, within an acceptable margin of confidence, sufficient capabilities are available to perform the above functions over a wide range of threats. This approach focuses on protecting the passive plant features and other key reactor components from hostile action by creating a robust perimeter.

The following are examples of features that enhance protection against malevolent actions:

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- Much of the RB structure, including the portion housing the RPV, is embedded underground, thereby naturally limiting access pathways.
- The number of entrances to the RB are minimized while maintaining emergency exits for personnel safety.

The BWRX-300 Security Annex further describes structures and features to detect, assess, impede, and delay threats up to and including the design basis threat for radiological sabotage in compliance with CNSC REGDOC-2.5.2, Section 7.22.1.

3.3.8 References

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- 3.3-2 CSA N291, "Requirements for Safety-Related Structures for Nuclear Power Plants," CSA Group.
- 3.3-3 CSA N289.1, "General Requirements for Seismic Design and Qualification of Nuclear Power Plants," CSA Group.
- 3.3-4 USNRC Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion."
- 3.3-5 ASCE/SEI 43, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," American Society of Civil Engineers.
- 3.3-6 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.3-7 IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design" International Atomic Energy Agency.
- 3.3-8 USNRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."
- 3.3-9 Canadian Commission on Building and Fire Codes, "National Building Code of Canada," National Resource Council of Canada.
- 3.3-10 10 CFR 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."
- 3.3-11 CSA N289.5, "Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities," CSA Group.
- 3.3-12 NK054-REP-01210-00098 R000, Geotechnical Data Report – R2, Darlington New Nuclear Project Geotechnical Investigation, EXP Services Inc. Project No. BRM-00025482-A0," Ontario Power Generation. 2013 (Reference 2.7-37)
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- 3.3-18 Campbell, K.W., et. al, Reference-Rock Site Conditions for Central and Eastern North America: Part II – Attenuation (Kappa) Definition, Pacific Engineering Research Center, PEER Report No. 2014/12. 2014 (Reference 2.7-42).
- 3.3-19 EPRI TR-102293-V5, "Guidelines for Determining Design Basis Ground Motions," Electric Power Research Institute. 1993 (Reference 2.7-44)
- 3.3-20 Silva, W.J., N. Abrahamson, G. Toro and C. Costantino. "Description and validation of the stochastic ground motion model." Brookhaven National Laboratory, Associated Universities, Inc. Upton, New York,. 1996 (Reference 2.7-45)
- 3.3-21 CSA N289.2-10, "Ground Motion Determination for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.3-22 Toro G. R., "Probabilistic models of site velocity profiles for generic and site-specific ground motion amplification studies." Technical Report 779574, Brookhaven National Laboratory, Upton, New York.
- 3.3-23 Darendeli, M.B., "Development of a new family of normalized modulus reduction and material damping curves." PhD thesis, The University of Texas, Austin.
- 3.3-24 ASCE/SEI 4, "Seismic Analysis of Safety-Related Nuclear Structures," American Society of Civil Engineers.
- 3.3-25 USNRC NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines." Reference 2.7-24
- 3.3-26 NK38-REP-03611-10041 (Reference 2.7-10)
- 3.3-27 USNRC DC/COL-ISG-017, "Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses."
- 3.3-28 NK38-CORR-03611-0847339, "Disposition to the CNSC's Comments on the Submission of an Update to the Probabilistic Seismic Hazard Assessment for Darlington NGS," August 19, 2020.)
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- 3.3-37 NEDC-33926P, "Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.3-38 BC-TOP-4A, "Seismic Analyses of Structures and Equipment for Nuclear Power Plants," Bechtel Power Corporation.
- 3.3-39 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 3.3-40 USNRC Regulatory Guide 1.12, "Nuclear Power Plant Instrumentation for Earthquakes."
- 3.3-41 USNRC Regulatory Guide 1.166, "Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake."
- 3.3-42 ANSI/ANS-2.23, "Nuclear Power Plant Response to an Earthquake," American National Standards Institute, Inc./American Nuclear Society.
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- 3.3-45 USNRC DC/COL-ISG-7, "Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures."
- 3.3-46 BC-TOP-3-A, "Tornado and Extreme Wind Design Criteria for Nuclear Power Plants," Bechtel Power Corporation.
- 3.3-47 USNRC Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants."
- 3.3-48 ORNL-NSIC-5, "U.S. Reactor Containment Technology. A Compilation of Current Practice in Analysis, Design, Construction, Test, and Operation," Oak Ridge National Laboratory.

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Table 3.3-1: Seismic Categories and Design Basis of BWRX-300 Structures

Structure	Safety Class	Seismic Category /Evaluation	Design/Evaluation Basis	Design Basis Earthquake ⁽¹⁾	Limit State ⁽²⁾
SCCV and Containment Steel Structures	SC1	Seismic Category A	CSA N289 series ASCE/SEI 43 and ASCE/SEI 4 ASME BPVC (see NEDC-33926P)	DBE	LS-D
Containment Internal Structures	SC1	Seismic Category A	CSA N289 series and N291 ASCE/SEI 43 and ASCE/SEI 4 ANSI/AISC N690	DBE	LS-D
RB SC and Steel Structures	SC1	Seismic Category A			
RWB Structure	SC3 ⁽³⁾	Seismic Category RW-IIa	CSA N289 Series and N291 RG 1.143 ASCE/SEI 43 and ASCE/SEI 4	½ DBE	LS-D
		Seismic Interaction Evaluation		DBE	LS-C
CB Structure	SC2	Non-Seismic Category	NBC		
		Seismic Interaction Evaluation	CSA N291 CSA N289 series ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
TB Structure	SC2	Non-Seismic Category	NBC		
		Seismic Interaction Evaluation	CSA N291 and CSA N289 series ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
Reactor Auxiliary Bay Structure	SC2	Non-Seismic Category	NBC		
		Seismic Interaction Evaluation	CSA N291 and CSA N289 series ASCE/SEI 43 and ASCE/SEI 4	DBE	LS-C
Other Structures	SC3/S CN	Non-Seismic Category	NBC		

1. DBE is defined in Subsection 3.3.1

2. Limit States per ASCE/SEI 43:

- LS-D Essentially elastic response
- LS-C response with limited permanent deformations

3. The RWB is designed in accordance with the radioactive waste management requirements for Category RW-IIa from U.S. NRC RG 1.143

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Table 3.3-2: Base Case Rock Dynamic Properties

Bedrock Formation	Total Unit Weight (kN/m ³)	Base Case Shear Wave			Poisson's Ratio
		V _s (m/s)	$\sigma_{\mu \ln}$	$\sigma_{\mu \ln V_s}$	
Blue Mountain (Whitby)	26.4	2,203	0.10	0.15	0.30
Lindsay1	26.6	2,708	0.10	0.15	0.31
Lindsay2	26.6	2,591	0.10	0.15	0.31
Lindsay3	26.6	2,881	0.10	0.15	0.31
Verulam1	26.4	2,185	0.10	0.15	0.33
Verulam2	26.4	2,500	0.10	0.15	0.31
Verulam3	26.4	2,623	0.10	0.15	0.31
Verulam4	26.4	2,761	0.10	0.15	0.31
Bobcaygeon	26.3	2,906	0.10	0.15	0.31
Gull River	26.5	3,139	0.10	0.15	0.32
Shadow Lake	25.7	2,706	0.10	0.15	0.30
Gneiss	27.3	3,128	0.10	0.15	0.28

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Table 3.3-3: Base Case Engineered Fill and In-situ Soil Dynamic

Layer	Shear Wave Velocity (m/s)			Poisson's Ratio
	Base Case V_s	$\sigma_{\mu \text{ In}}$	$\sigma_{\mu \text{ In} V_s}$	Average
Fill 1	207	0.40	0.25	0.35
Fill 2	235	0.40	0.25	0.35
Fill 3	254	0.40	0.25	0.35
Fill 4	271	0.40	0.25	0.35/0.40
Fill 5	287	0.40	0.25	0.35/0.40
Fill 6	300	0.40	0.25	0.35/0.40
Fill 7	314	0.40	0.25	0.35/0.40
Upper till	513	0.40	0.25	0.35/0.40
Intermediate glacio-lacustrine (Sandy)	506	0.40	0.15	0.40
Intermediate glacio-lacustrine (Silty)	480	0.40	0.15	0.40
Lower till	524	0.40	0.15	0.40

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Table 3.3-4: Rock Layers Kappa Values

Case	Bedrock Kappa (κ_0, ref; sec)	Rock Layer Kappa (κ_r; sec)	Total Kappa at Top of Rock (sec)
Base Case	0.006	0.002	0.008
Lower Realization	0.006	0	0.006
Upper Realization	0.006	0.006	0.012

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Table 3.3-5: Selected Time History Records

Record	NUREG/CR-6728 Database Bin	Component	Scaling Factor	HP (Hz)	LP (Hz)	Peak Ground Acceleration (G)	Duration (seconds)
HWA056	Rock M7+ R 10-50 km	H1 (North)	1.46	0.03	50	0.203	86.000
		H2 (West)	1.46	0.02	50	0.207	86.000
		Vertical	1.50	0.02	50	0.120	86.000
TCU047	Rock M7+ R 10-50 km	H1 (North)	0.40	0.03	50	1.168	89.995
		H2 (West)	0.44	0.02	50	0.700	89.995
		Vertical	0.50	0.02	50	0.556	89.995
ILA063	Rock M7+ 50-100 km	H1 (North)	1.31	0.02	50	0.221	78.990
		H2 (West)	1.37	0.02	50	0.226	78.990
		Vertical	2.26	0.04	50	0.122	78.990
HWA026	Rock M7+ 50-100 km	H1 (North)	2.10	0.03	50	0.135	89.995
		H2 (West)	1.45	0.02	50	0.202	89.995
		Vertical	2.40	0.02	50	0.110	89.995
TAP075	Rock M7+ 100-200 km	H1 (North)	1.69	0.02	50	0.171	91.999
		H2 (West)	1.45	0.01	30	0.205	91.999
		Vertical	2.57	0.03	30	0.110	91.999

Table 3.3-6: Seismic Damping Values for BWRX-300 Structures

Material	Response Level 1	Response Level 2
Steel-plate composite structures	3	5
Welded and Friction-bolted steel structures	2	4
Bearing-bolted steel structures	4	7
Reinforced concrete structures	4	7

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Table 3.3-7: Seismic Damping Values for Primary Subsystems

Component	Level 1 Damping		Level 2 Damping	
	Horizontal	Vertical	Horizontal	Vertical
Reactor Vessel	2	2	4.0	4.0
Vessel Support Skirt	2	2	4.0	4.0
Shroud	2	2	4.0	4.0
Shroud Support Spring	2	2		
Shroud Head & Separator	2	2	4.0	4.0
Fuel	4	4	6.0	6.0
CRD Guide Tubes	1	1	2.0	2.0
CRD Housing	1	1	2.0	2.0
CRD Restraint Springs	-	2		
Stabilizer and Bellows	-	2		
Welded Steel			4.0	4.0
Bolted Steel			7.0	7.0

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Table 3.3-8: Preliminary Dynamic Loading α , β – Damping

Loading	Shell Model	Total Model Damping at A & B Freq	A Freq (Hz)	B Freq (Hz)	α	β
LOCA	52	6%	10	60	6.527	.000257
		6%	1.8	12.7	1.2083	.0011637
	110	4%	10	60	4.3731	.0001655
		4%	1.8	12.7	.8121	.0007246

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Table 3.3-9: Seismic Damping Values for Piping and Equipment

Structure or Component	Level 1 Damping	Level 2 Damping
Equipment and large-diameter piping system, pipe diameter greater than 12 in.	3	5
Small-diameter piping systems, diameter equal to or less than 12 in.	2	5
Welded steel structures	3	4
Bolted steel structures	4	7

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Table 3.3-10: Piping and Equipment Damping Values for All Other Non-Seismic Loadings

Structure or Component	When considered by itself and/or combined with other load and designated as normal, upset and emergency	When considered by itself and/or combined with other load and designated as faulted
Equipment and large-diameter piping system, pipe diameter greater than 12 in.	2	3
Small-diameter piping systems, diameter equal to or less than 12 in.	1	2
Welded steel structures	2	4
Bolted steel structures	4	7

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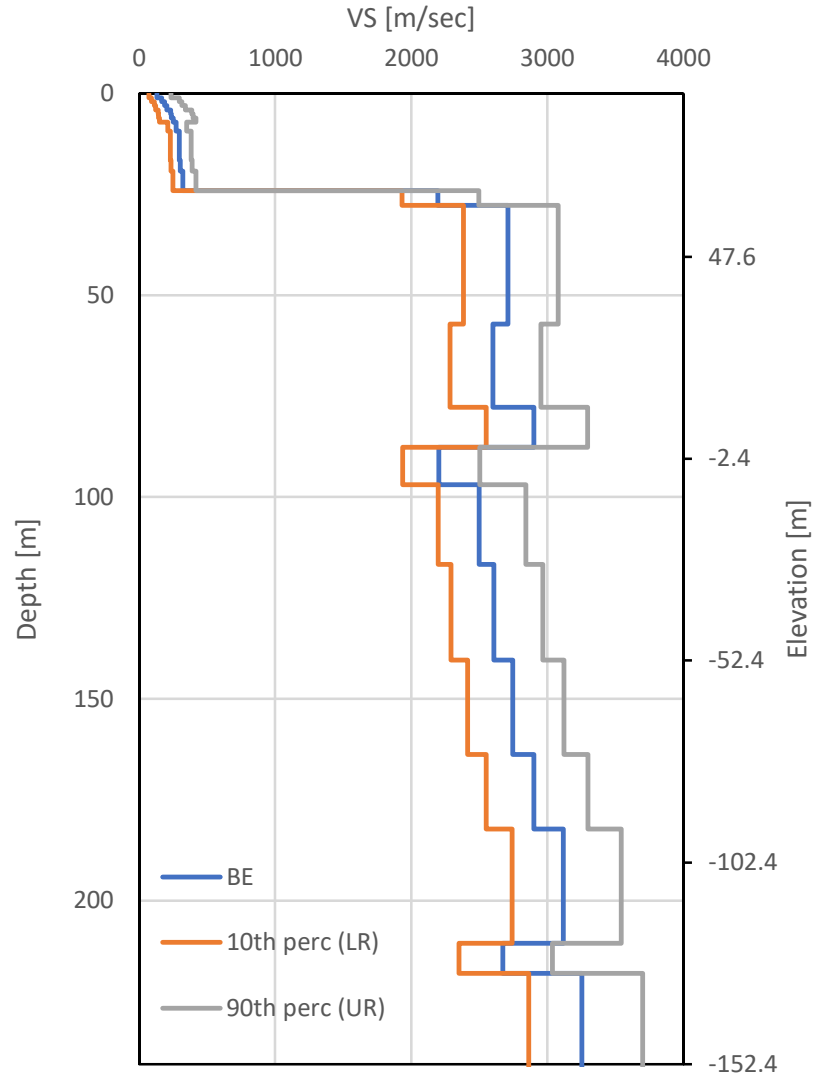


Figure 3.3-1: Shear Wave Velocities for the Bounding In-situ Profile

<u>Shear-wave velocity cases (weight)</u>	<u>Kappa cases (weight)</u>	<u>Total weight</u>
BE (0.4)	BE (0.4)	0.16
	10 th perc. (0.3)	0.12
	90 th perc. (0.3)	0.12
10 th perc. (0.3)	BE (0.4)	0.12
	10 th perc. (0.3)	0.09
	90 th perc. (0.3)	0.09
90 th perc. (0.3)	BE (0.4)	0.12
	10 th perc. (0.3)	0.09
	90 th perc. (0.3)	0.09
		$\Sigma = 1.0$

Figure 3.3-2: Cases Considered for Explicit Considerations of Epistemic Uncertainties

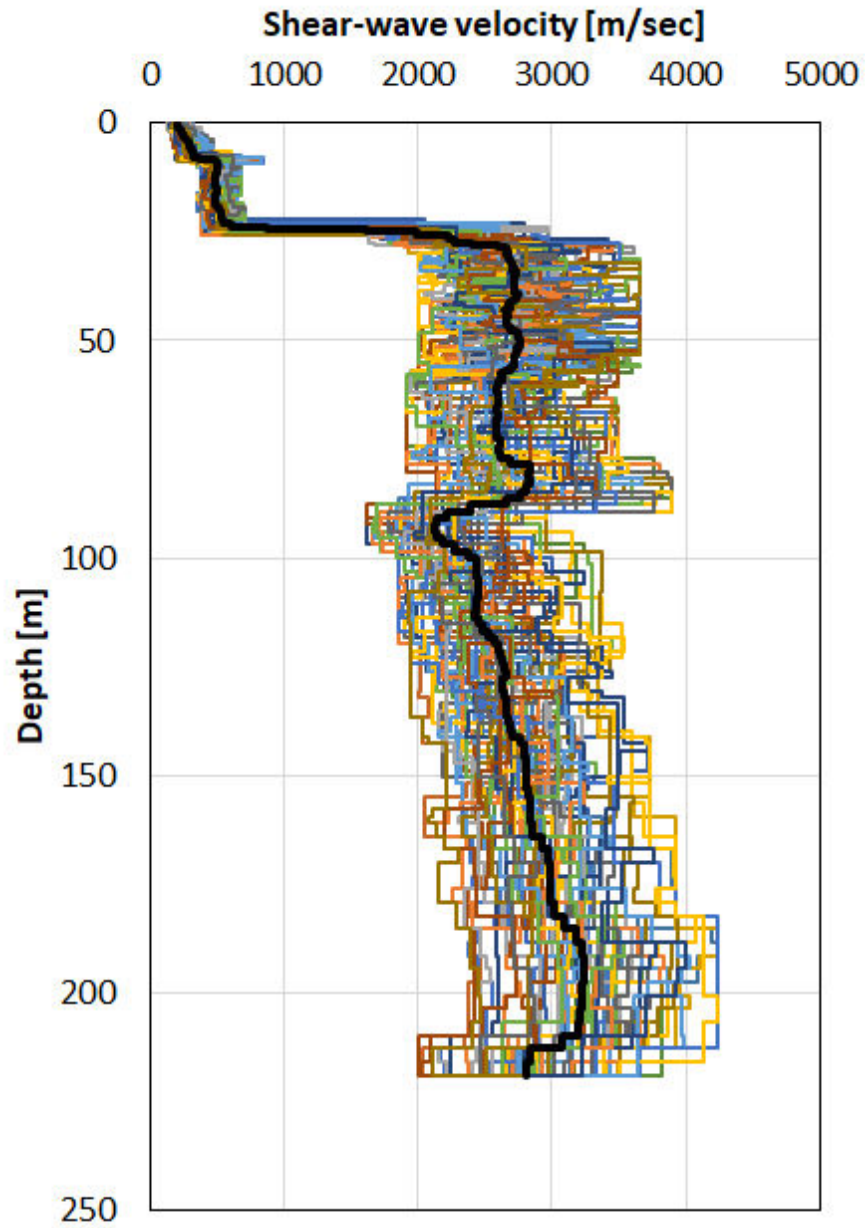


Figure 3.3-3: Shear Wave Velocity and Layer Thicknesses Randomization – BE-BE Case

Note: The Black line designates the resulting mean curve

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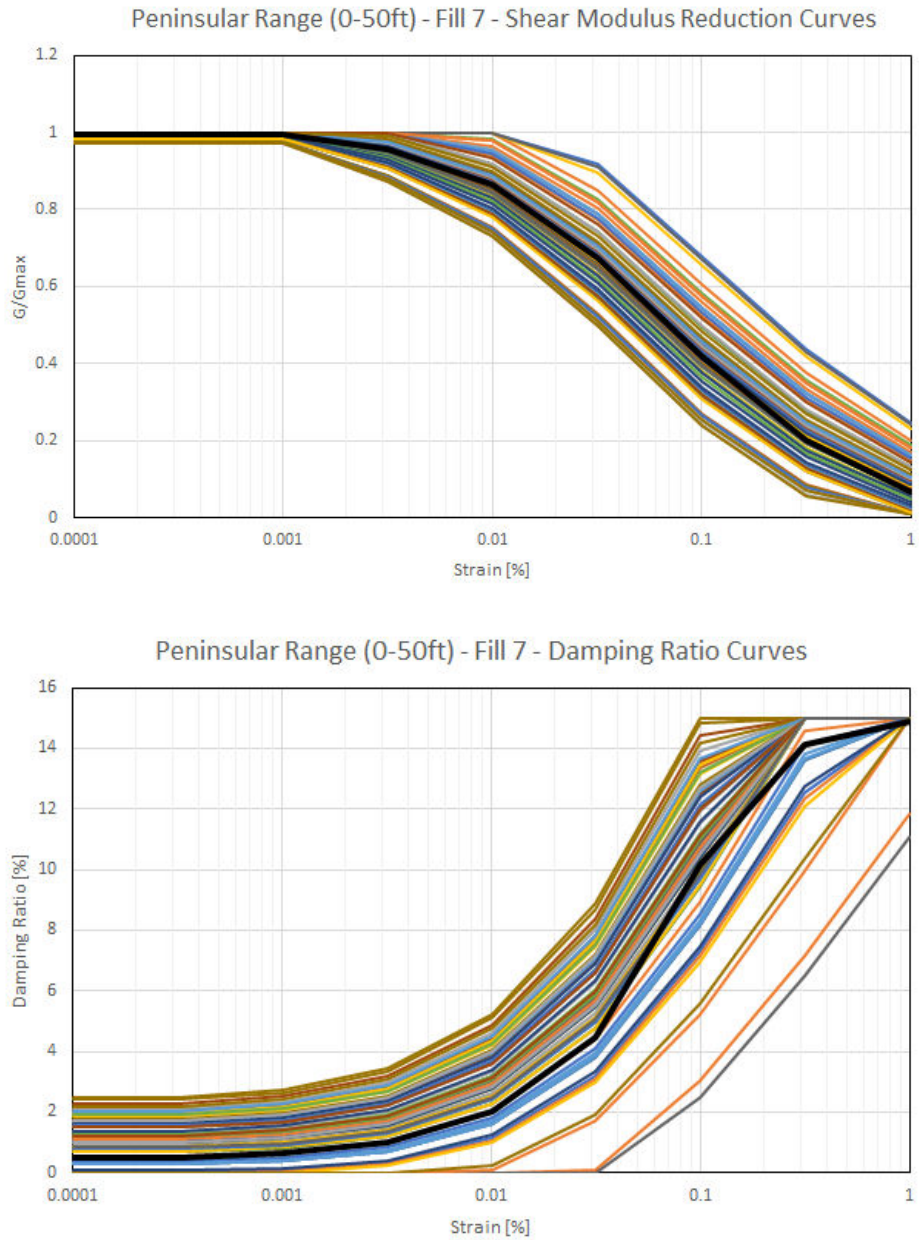


Figure 3.3-4: Soil Degradation Curves Randomization

Note: The Black line designates the resulting mean curve

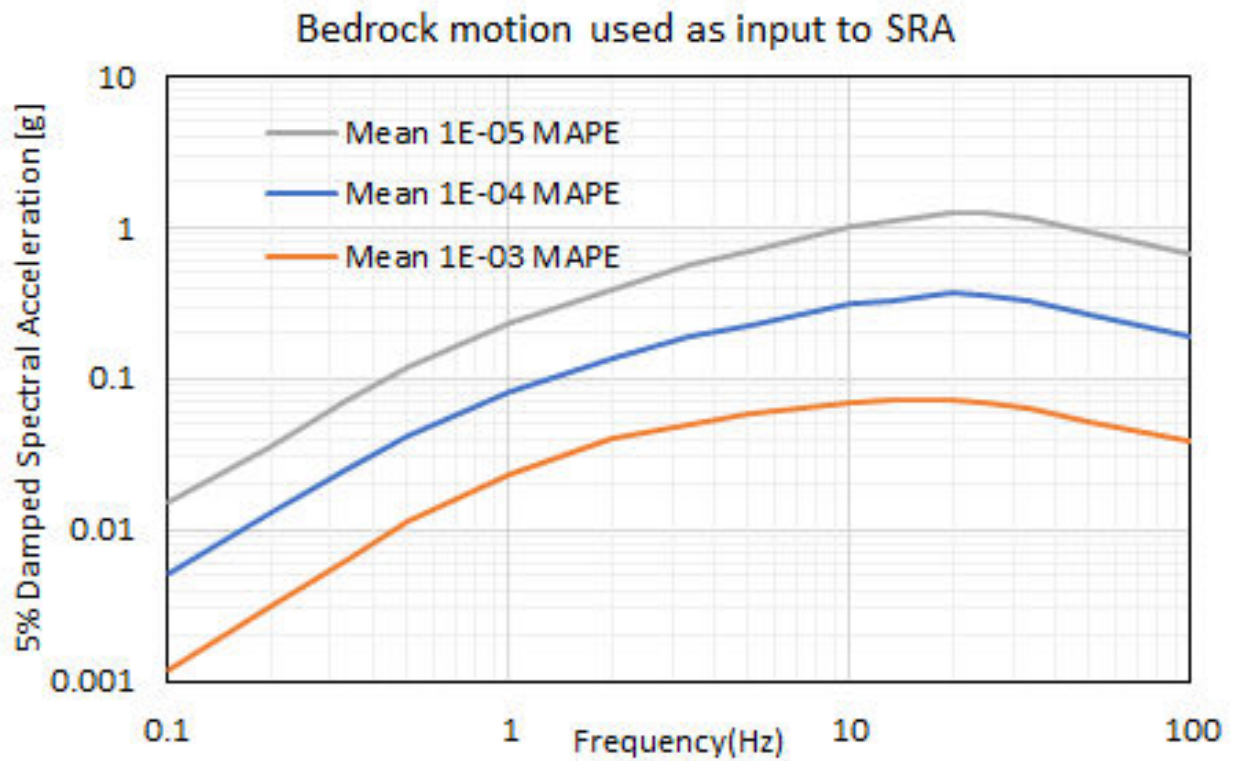


Figure 3.3-5: Uniform Hazard Response Spectra at Bedrock Elevation

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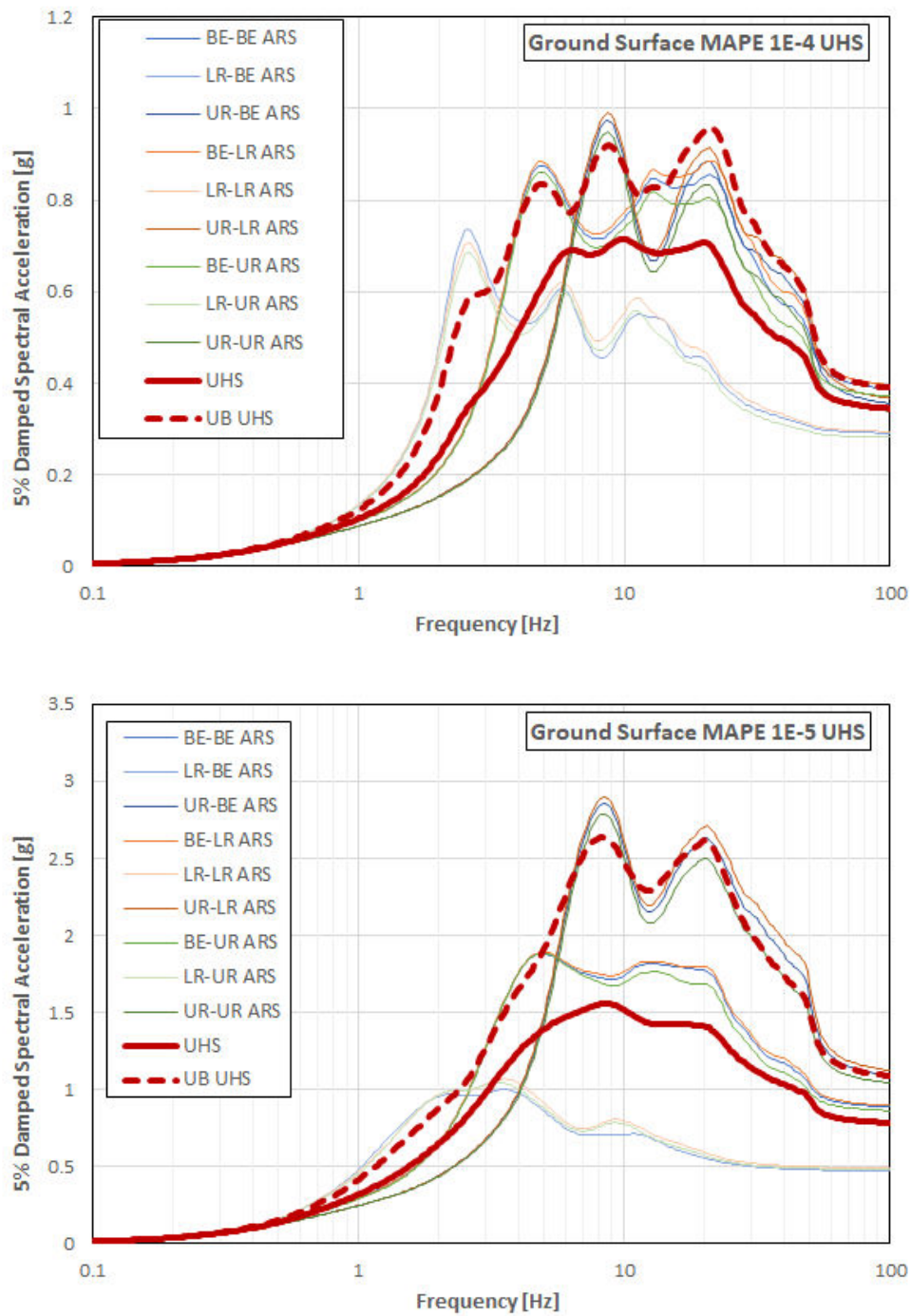


Figure 3.3-6: Ground Surface Uniform Hazard Response Spectra

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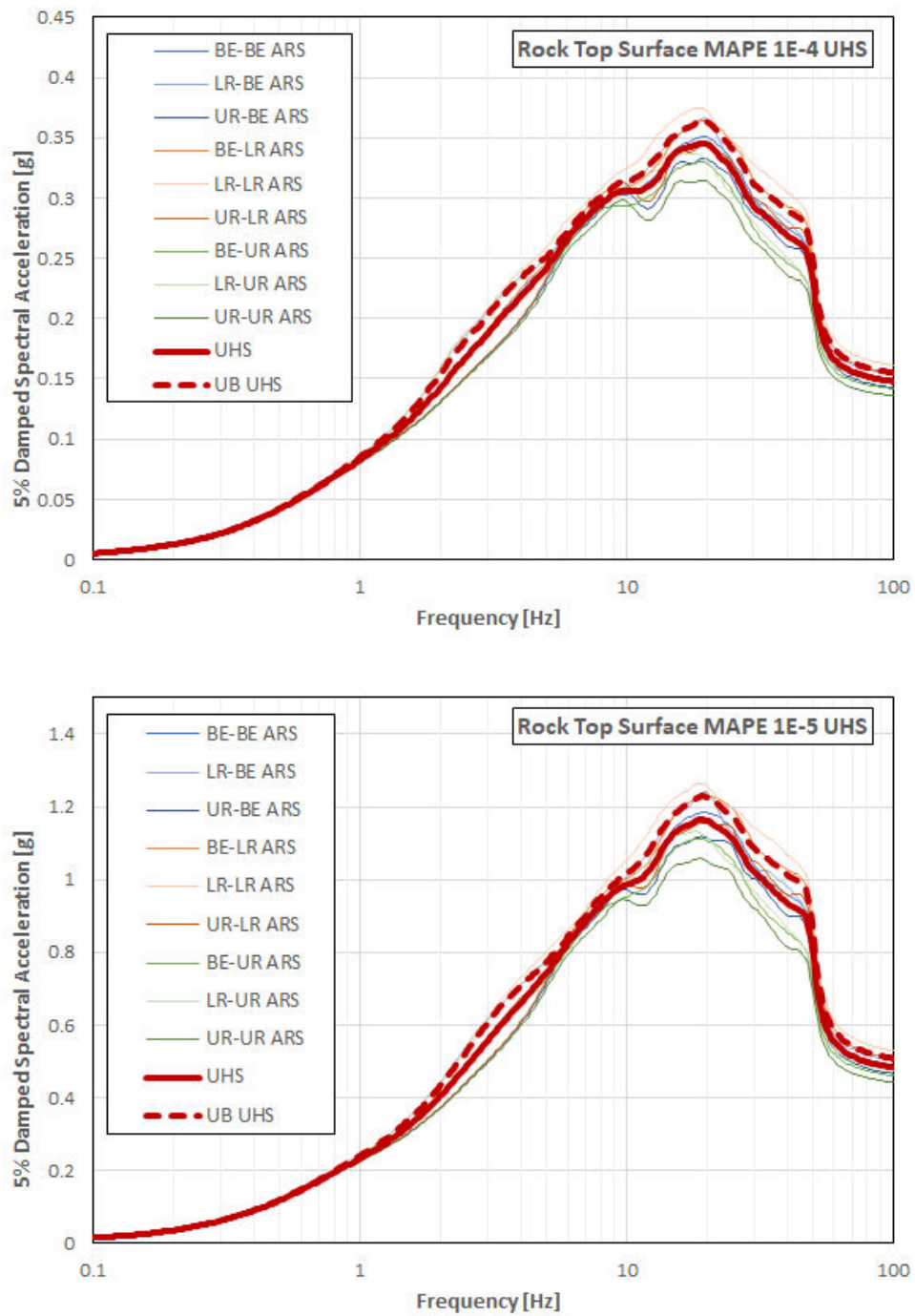


Figure 3.3-7: Rock Top Surface Uniform Hazard Response Spectra

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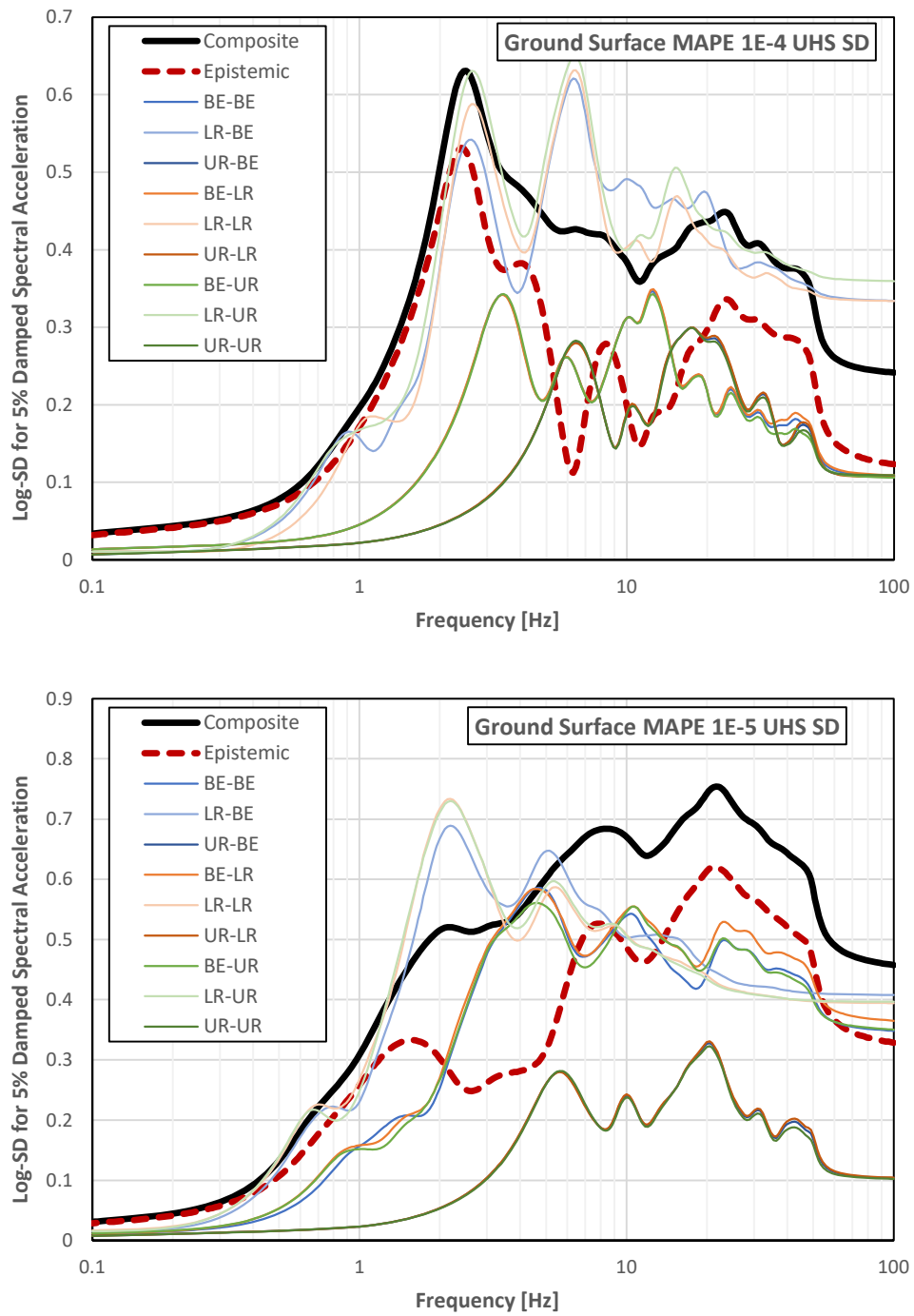


Figure 3.3-8: Ground Surface Composite and Epistemic Log-Normal Standard Deviations

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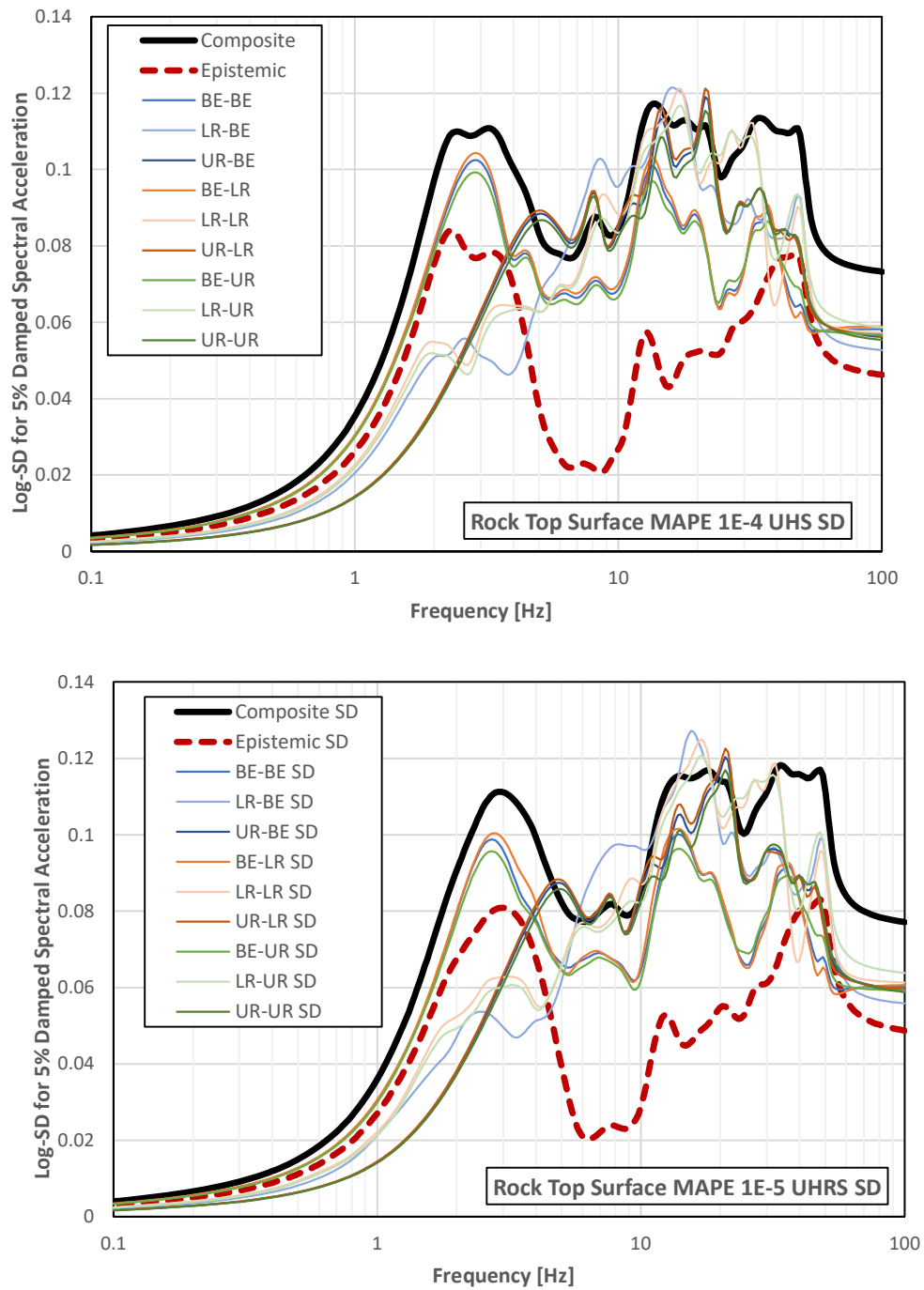


Figure 3.3-9: Rock Top Surface Composite and Epistemic Log-Normal Standard Deviations

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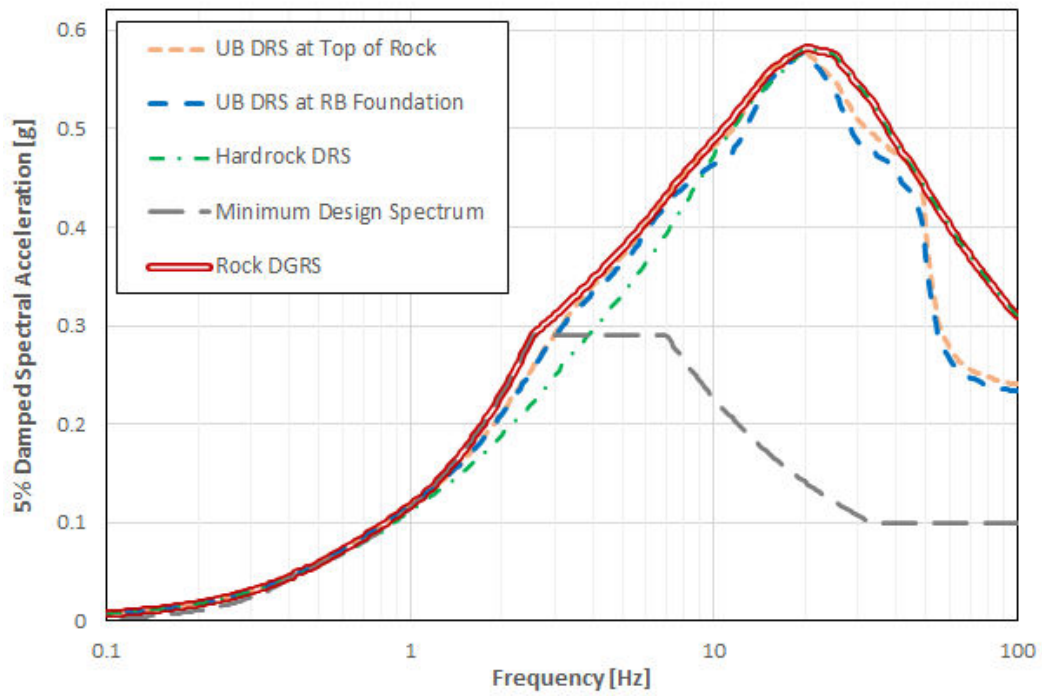


Figure 3.3-10: Bounding Horizontal Rock Ground Motion Response Spectra

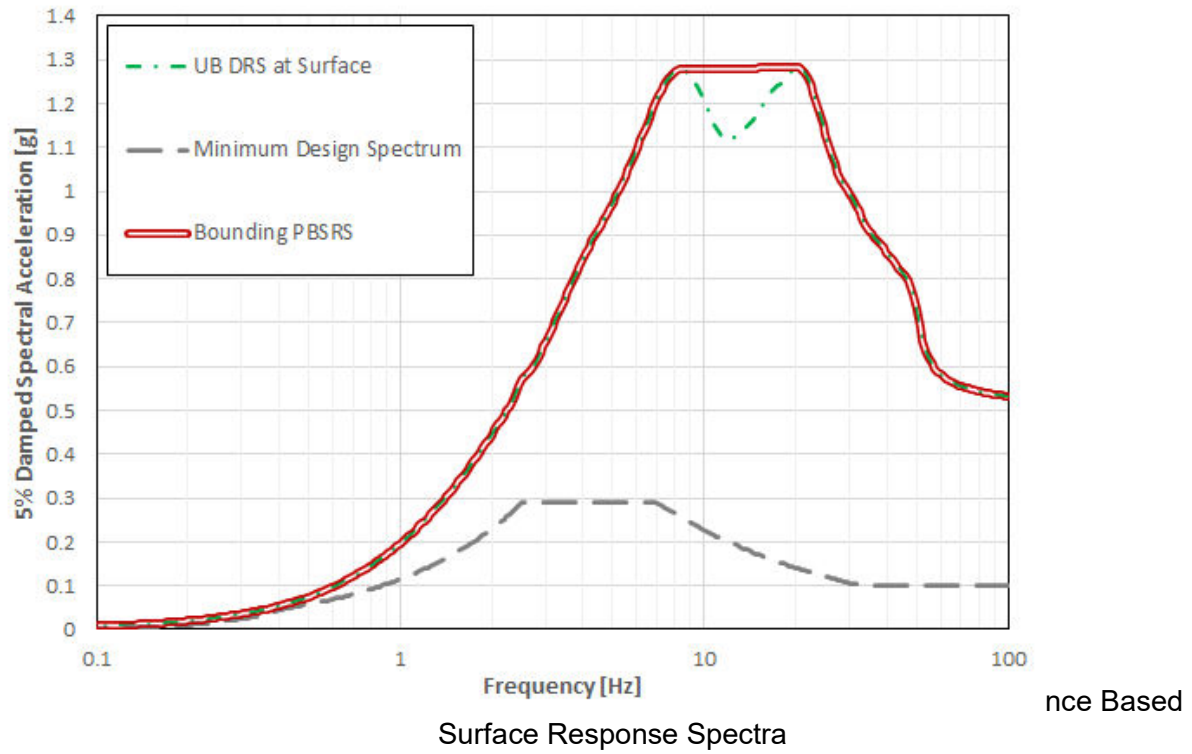


Figure 3.3-11: Bounding Horizontal Performance Based Surface Response Spectra

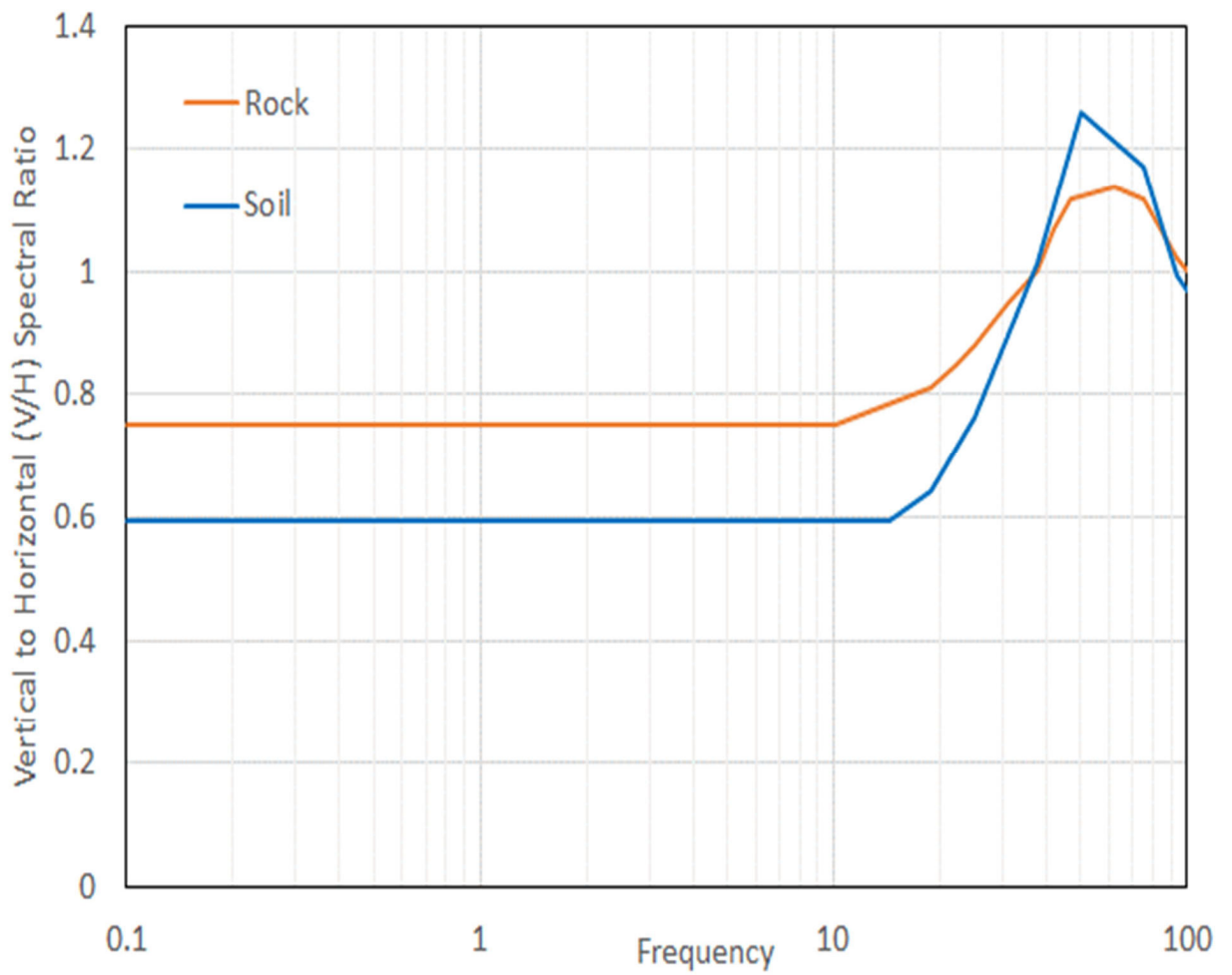


Figure 3.3-12: Vertical to Horizontal (V/H) Spectral Ratios

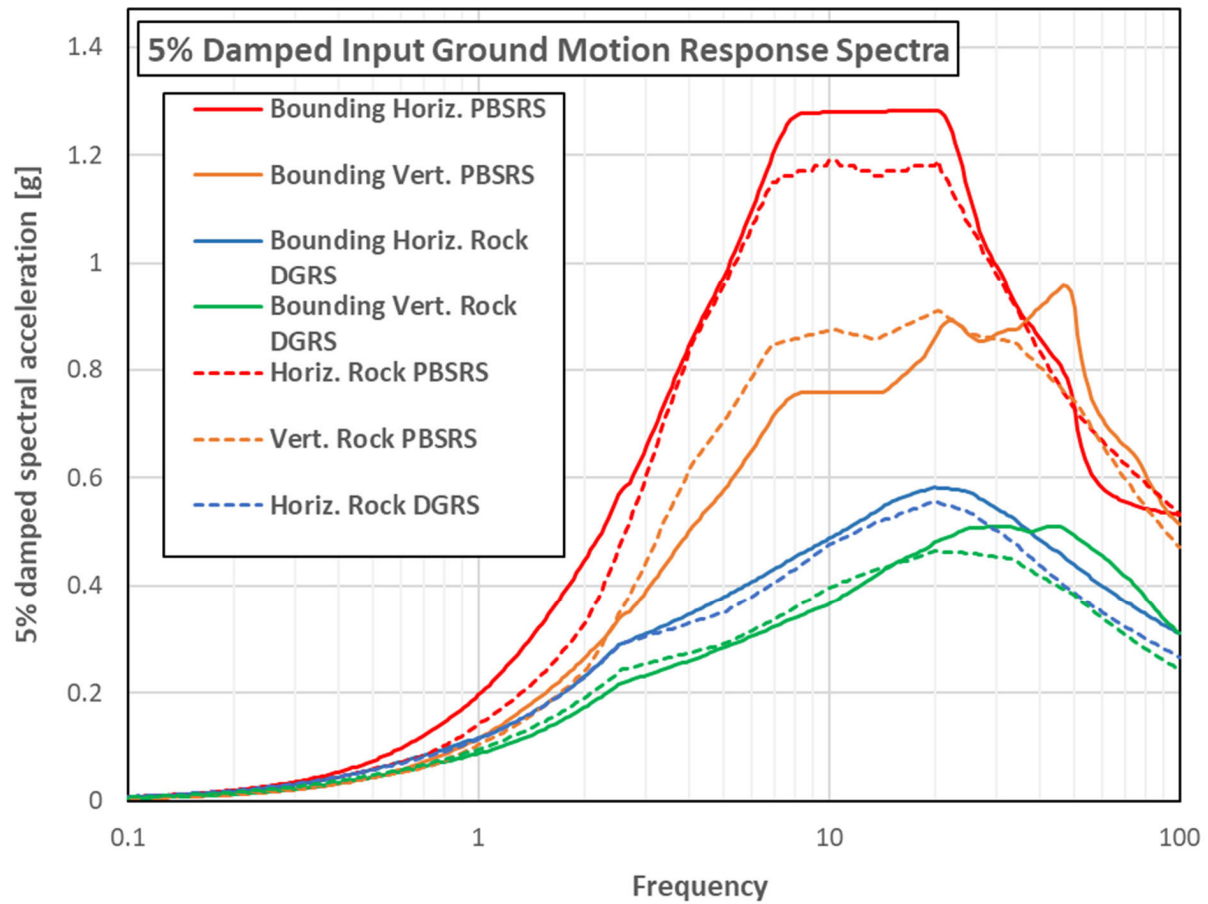


Figure 3.3-13: Comparison of Bounding to Updated Ground Motion Design Response Spectra

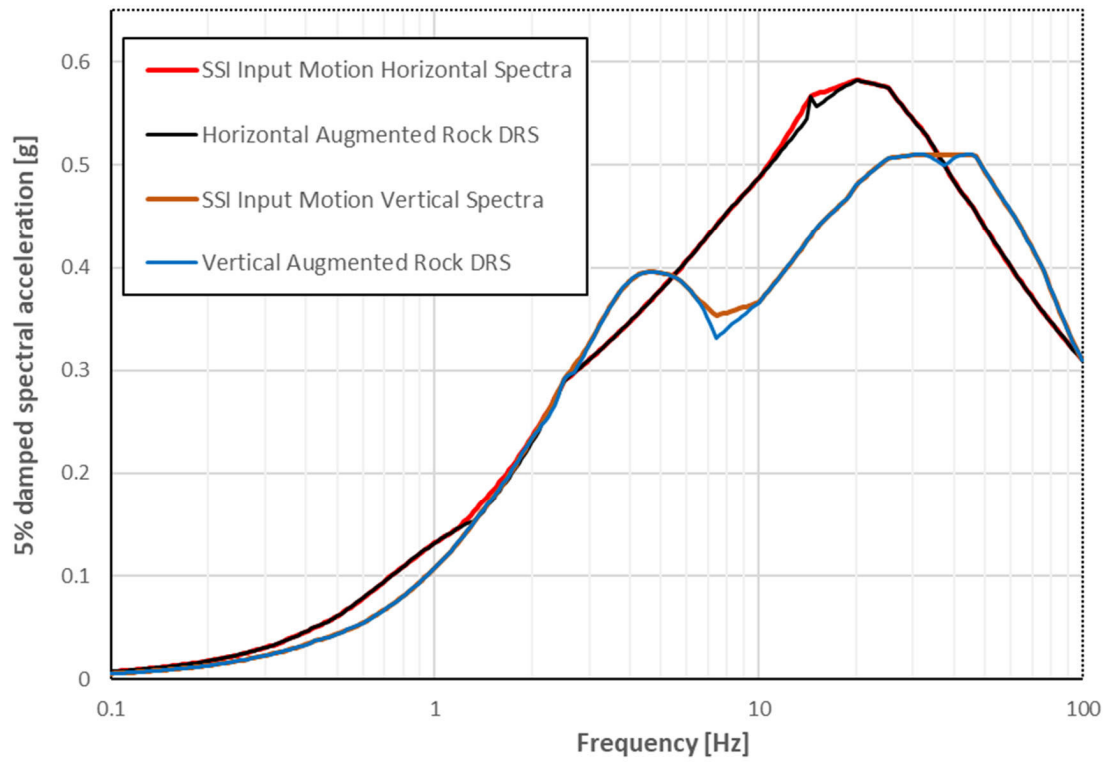


Figure 3.3-14: Augmented and Smoothed Horizontal and Vertical Rock Design Ground Response Spectra

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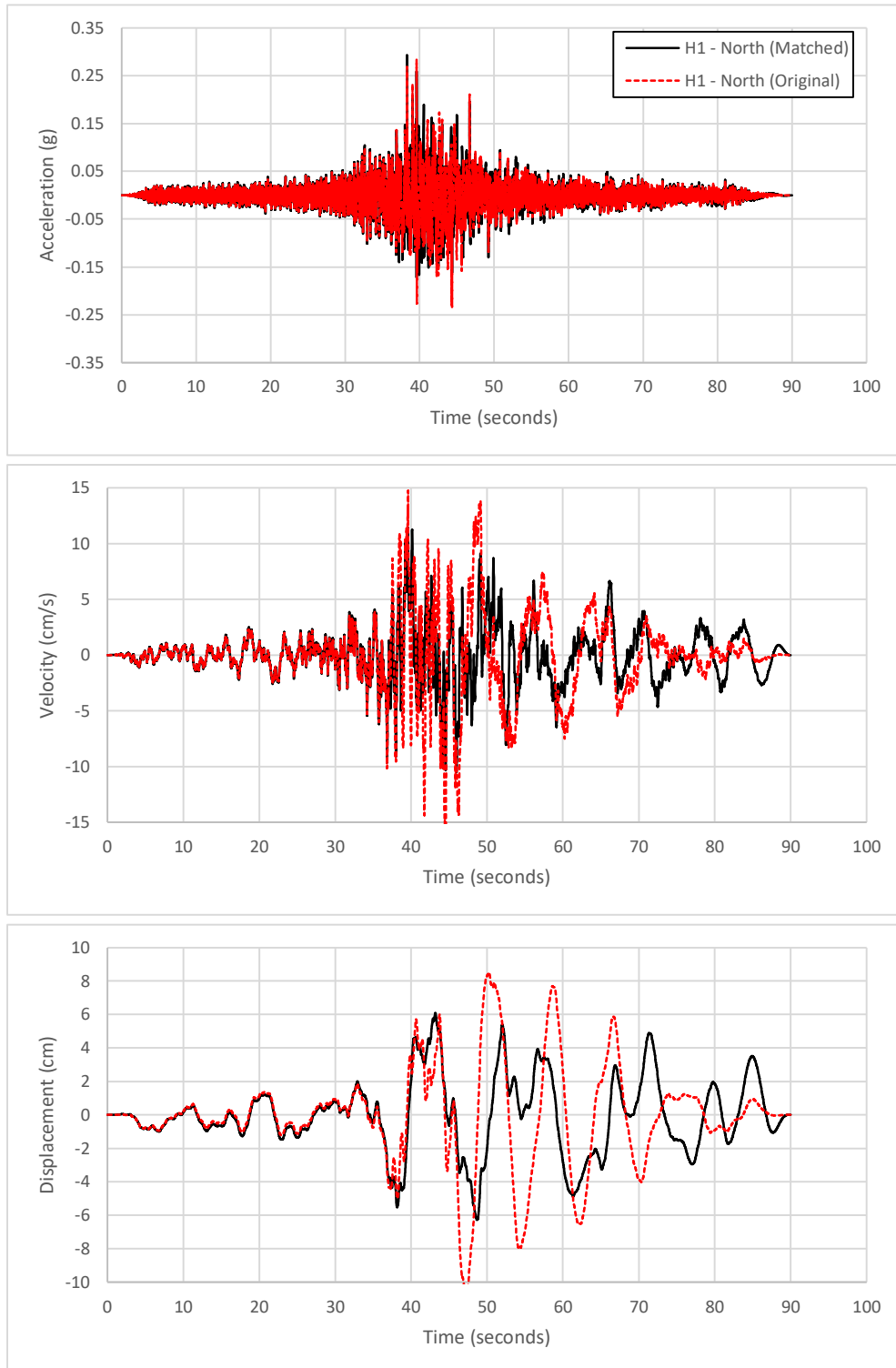


Figure 3.3-15: Horizontal Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 North

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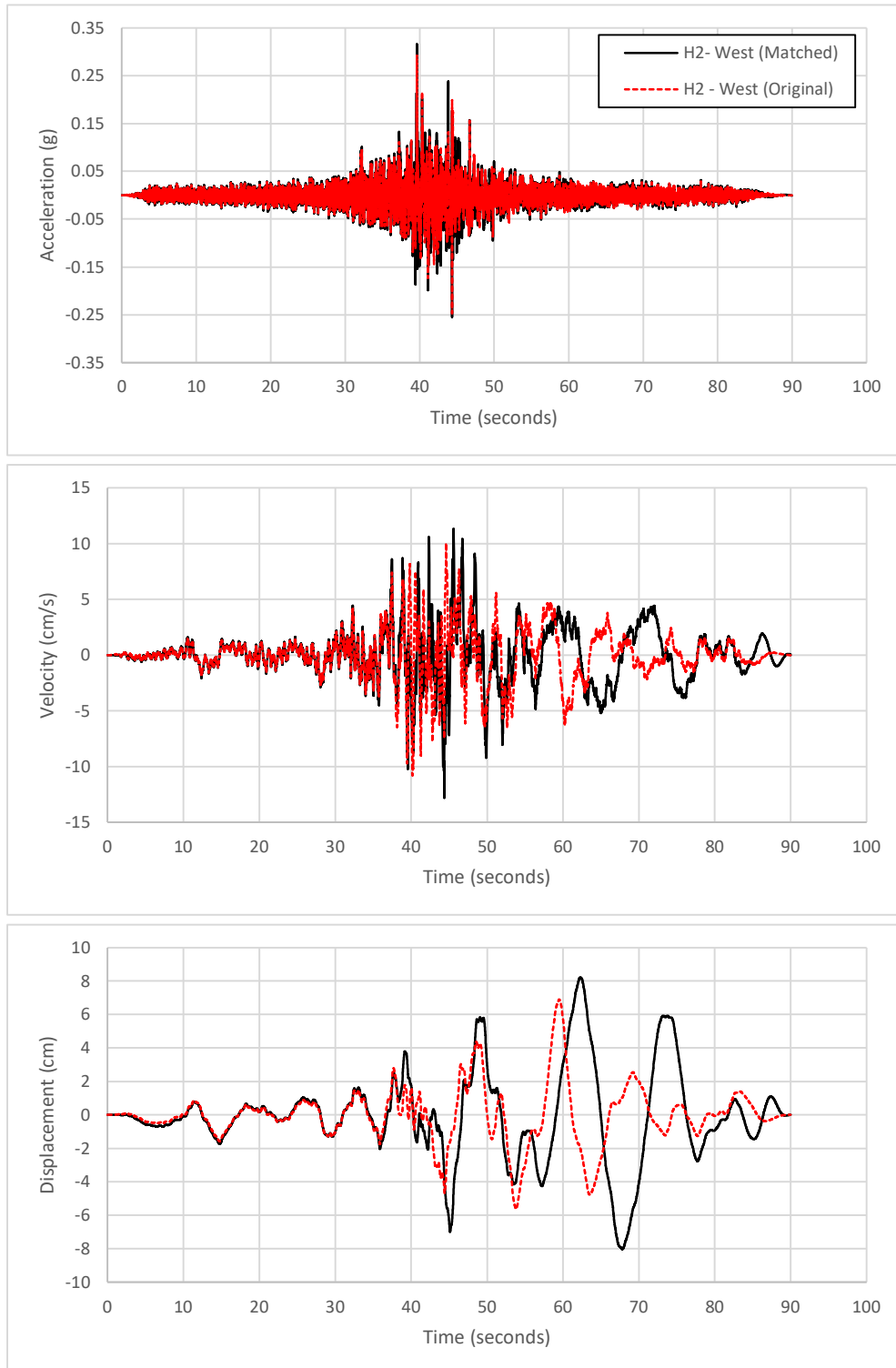


Figure 3.3-16: Horizontal Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 West

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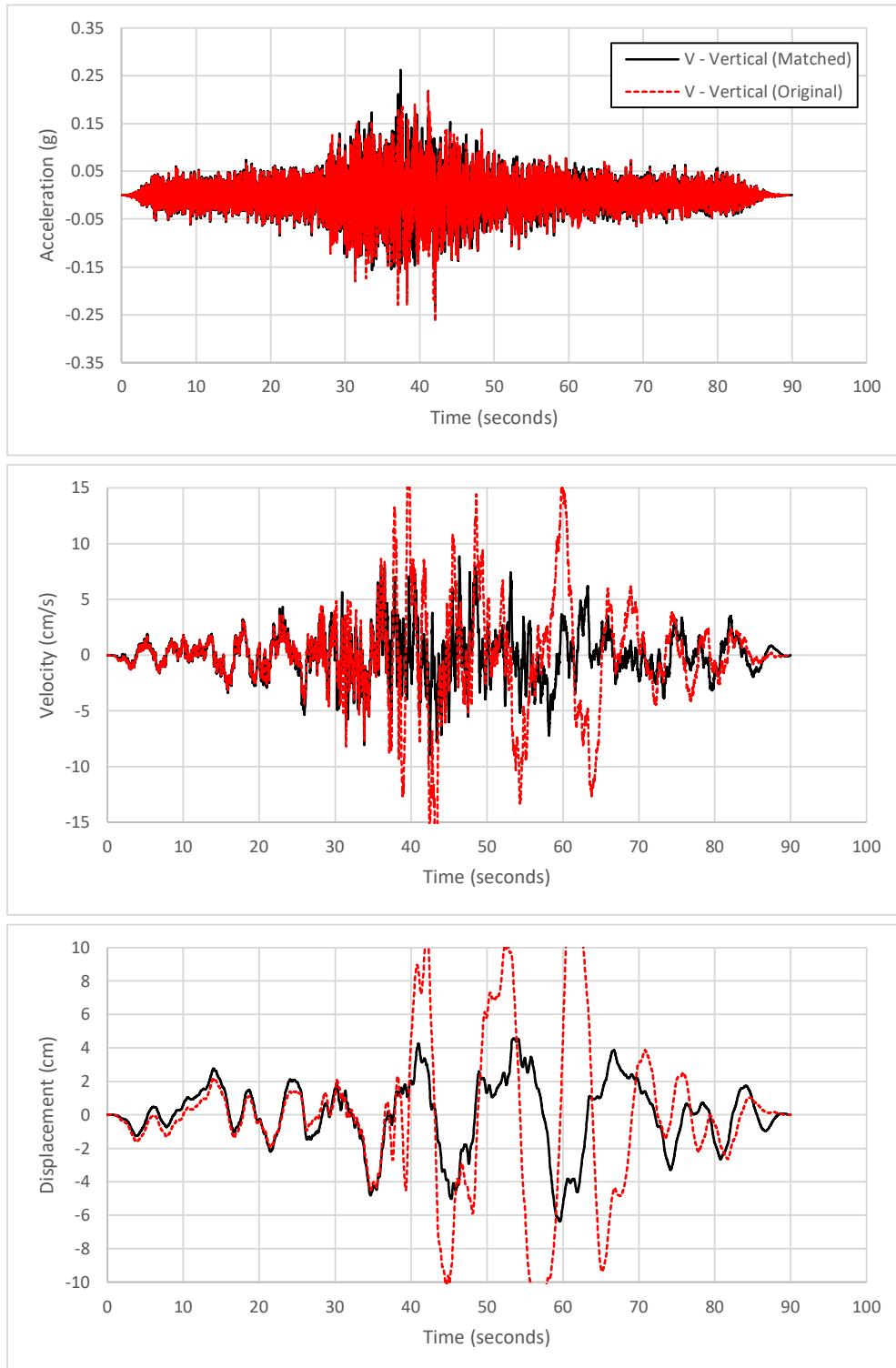


Figure 3.3-17: Vertical Response Spectrum Matched Acceleration, Velocity and Displacement Time Histories from Seed Record HWA026 Vertical

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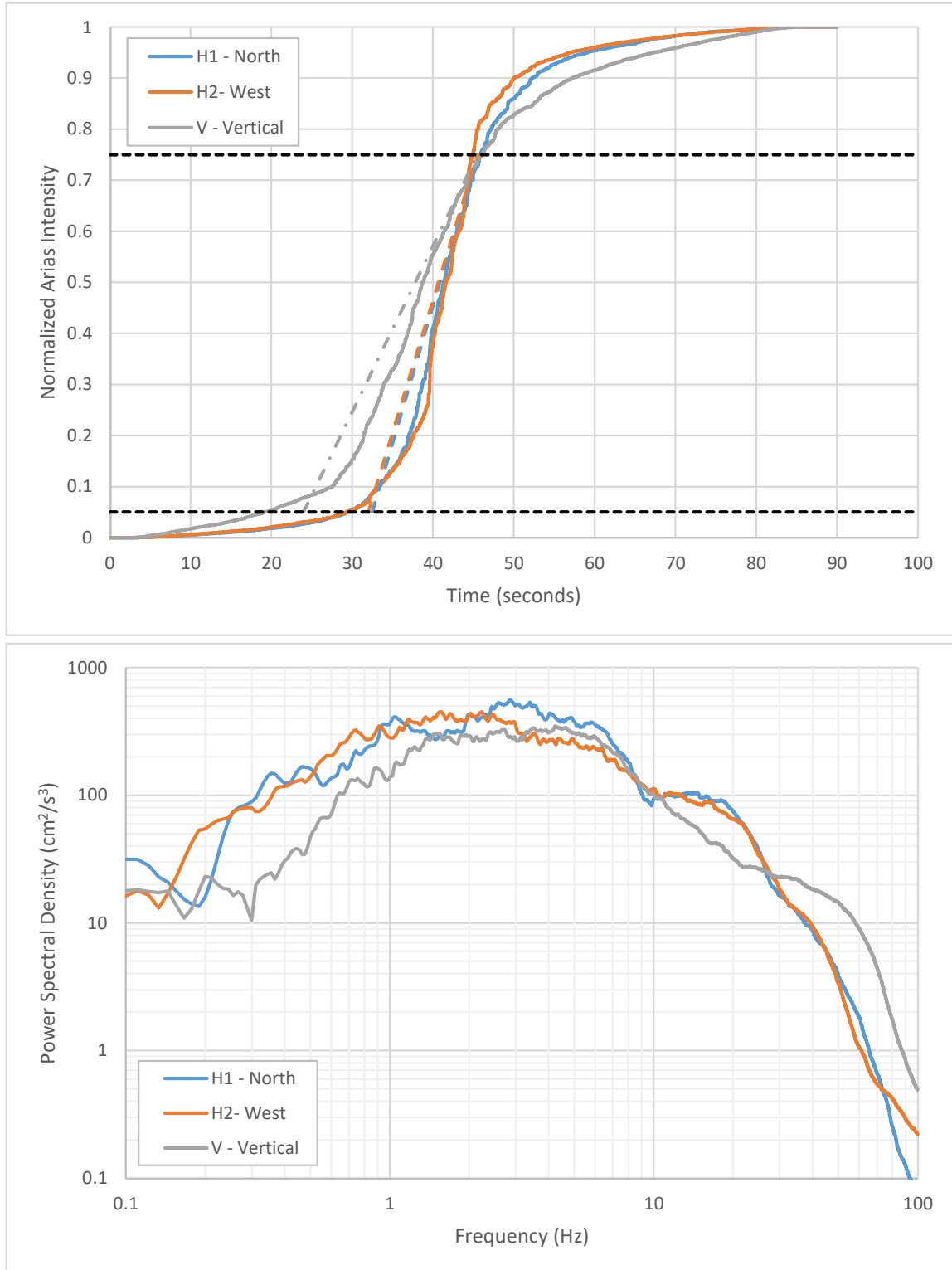


Figure 3.3-18: Normalized Arias Intensity and Power Spectral Density Function for Response Spectrum Matched HWA026 Acceleration Time Histories

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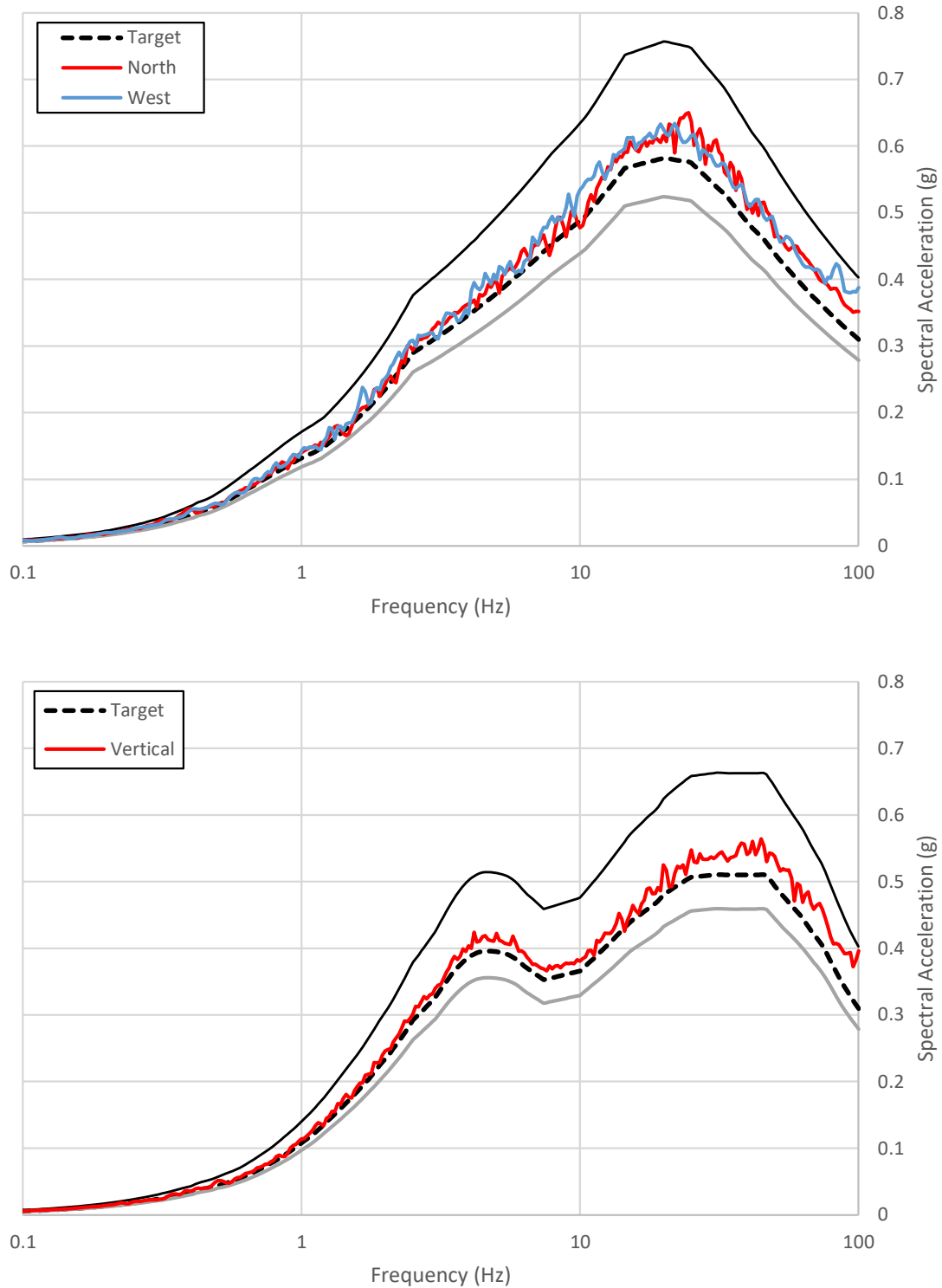
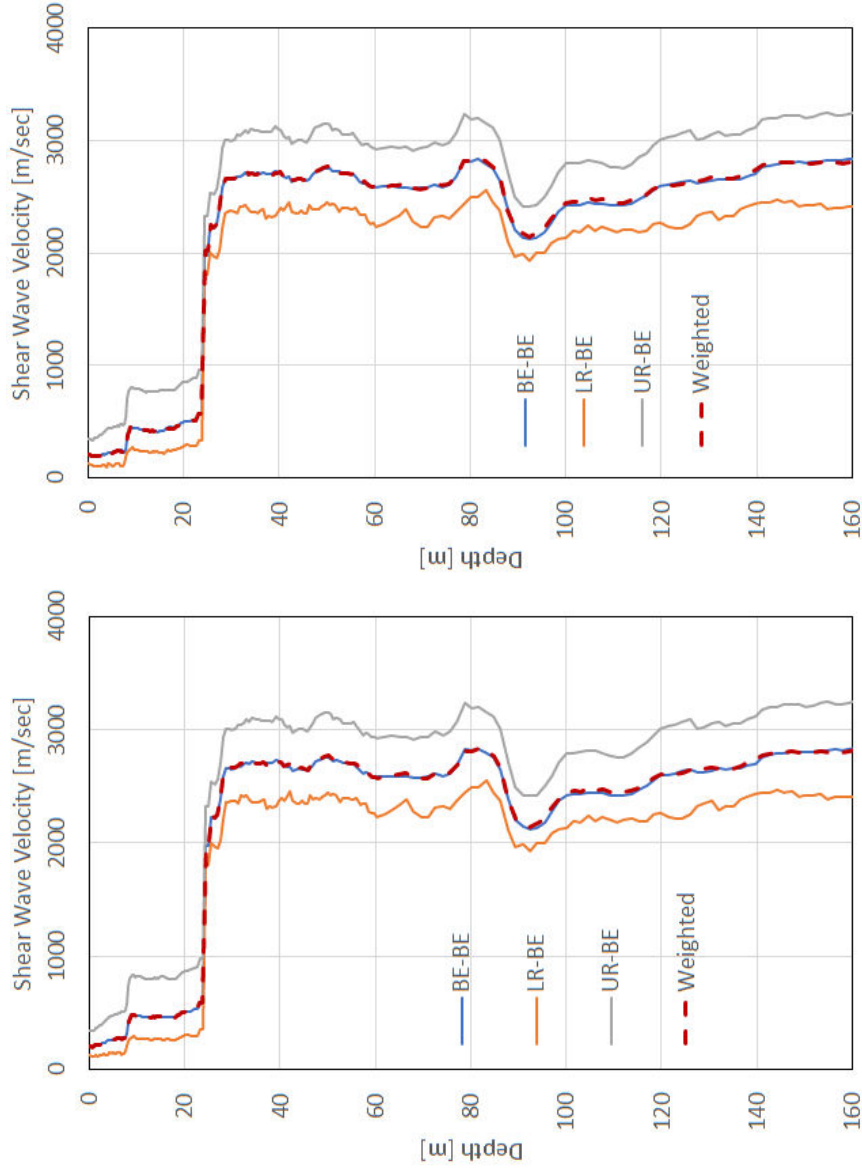


Figure 3.3-19: 5% Damped Response Spectra for Response Spectrum Matched HWA026 Acceleration Time Histories



A) 1 E-04 MAPE

B) 1 E-05 MAPE

Figure 3.3-20: Logarithmic Mean of Strain-Compatible Shear Wave Velocities

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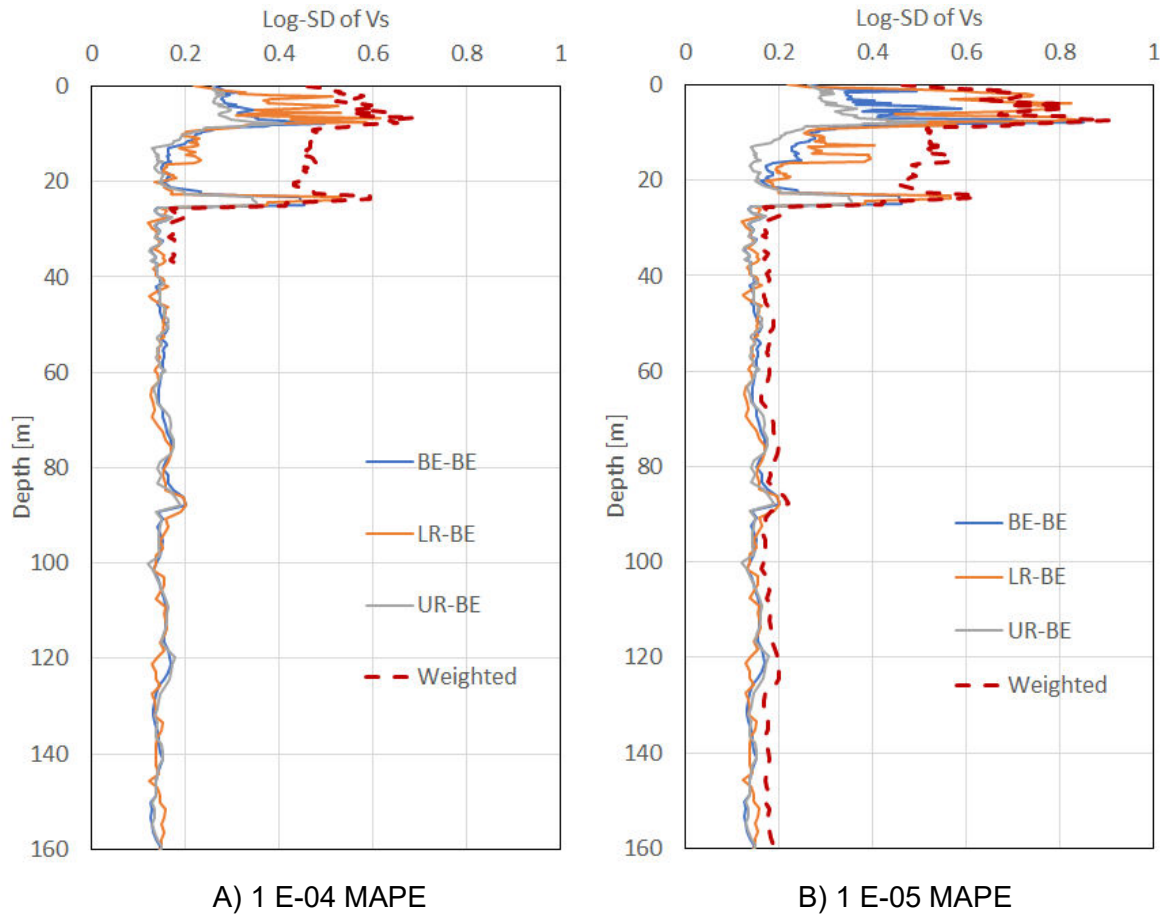


Figure 3.3-21: Logarithmic Standard Deviation of Strain-Compatible Shear Wave Velocities

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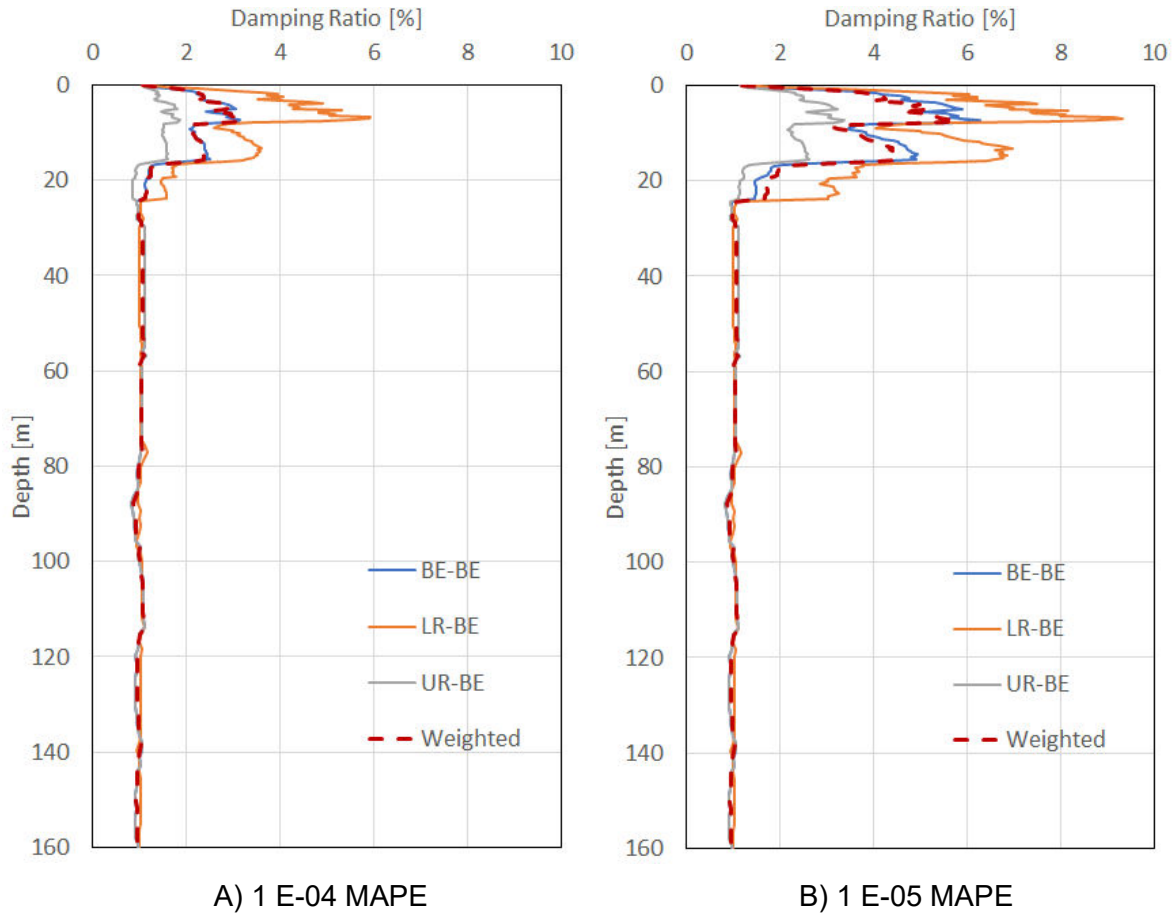


Figure 3.3-22: Logarithmic Mean of Strain-Compatible Damping Ratios

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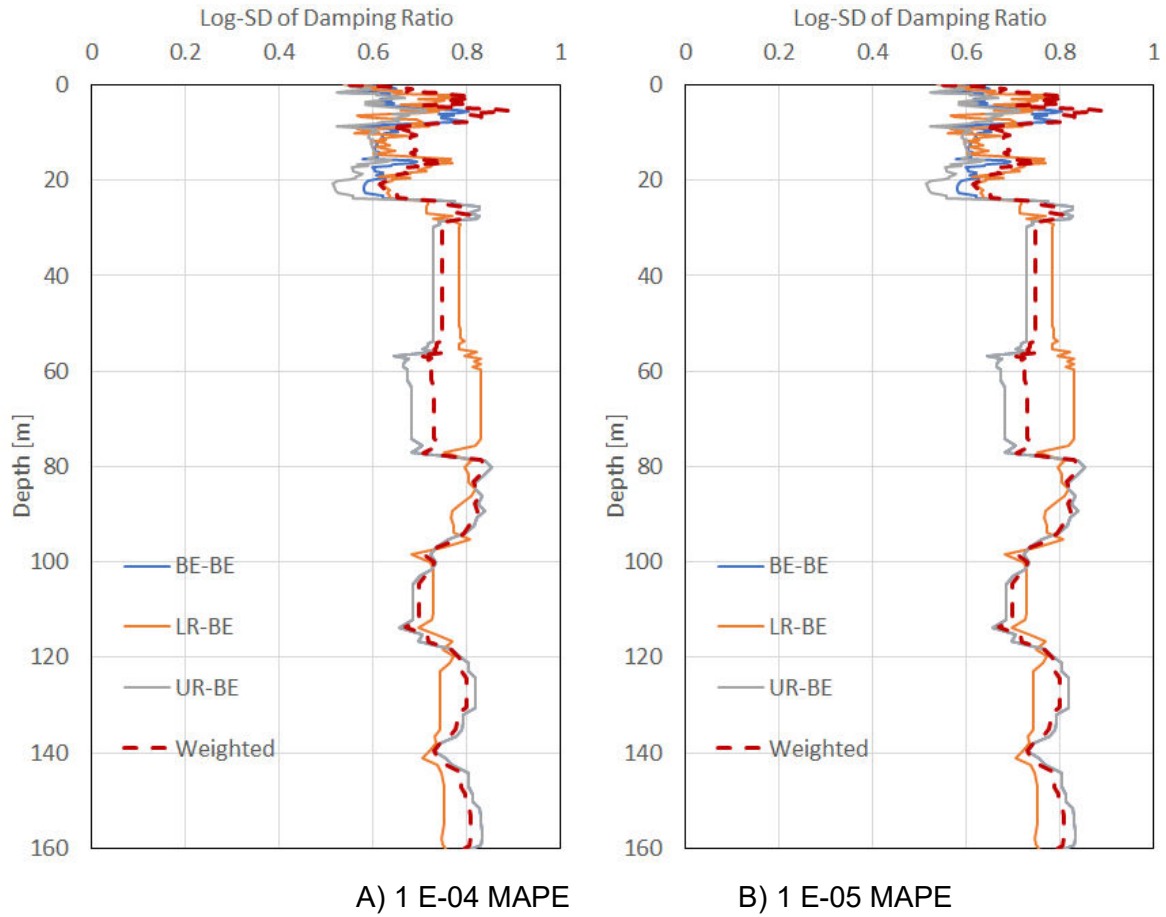
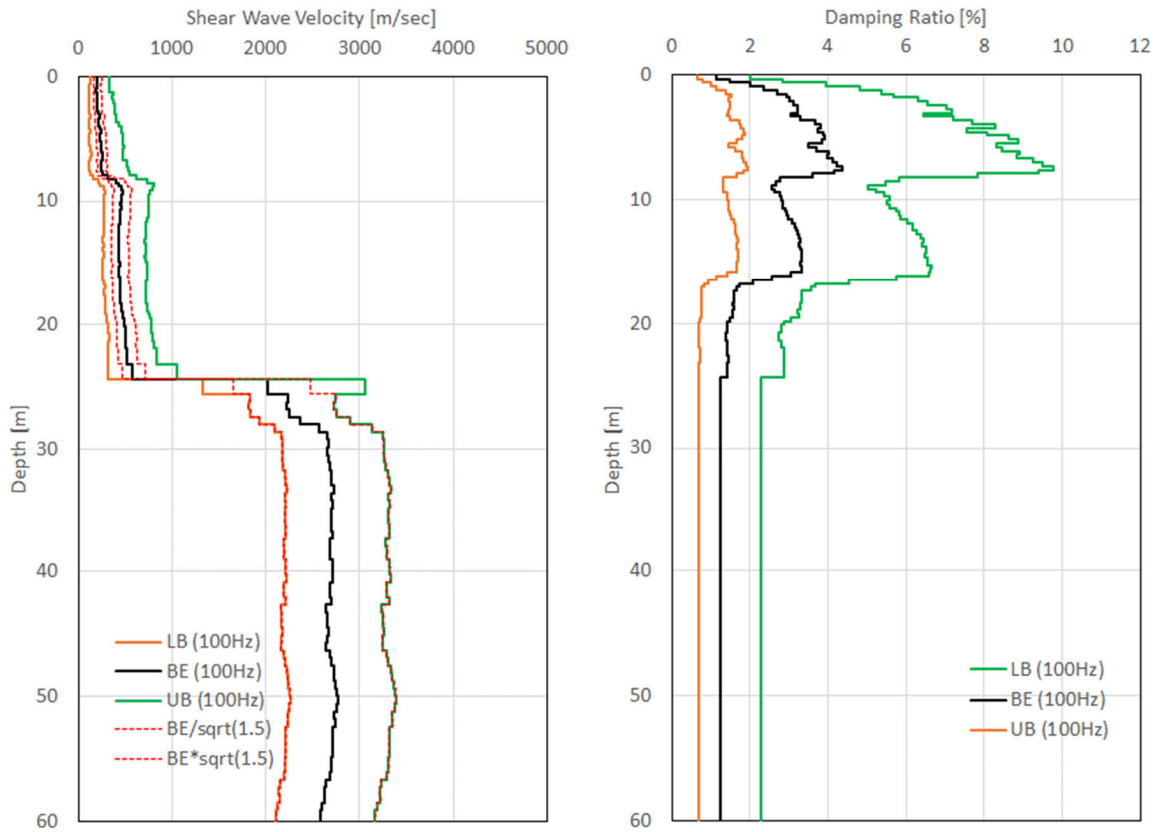


Figure 3.3-23: Logarithmic Standard Deviation of Strain-Compatible Damping Ratios

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a) Shear Wave Velocity

b) Shear Wave Damping Ratio

**Figure 3.3-24: Strain-Compatible Shear Wave Velocity and Damping
Using 100 Hz Interpolation**

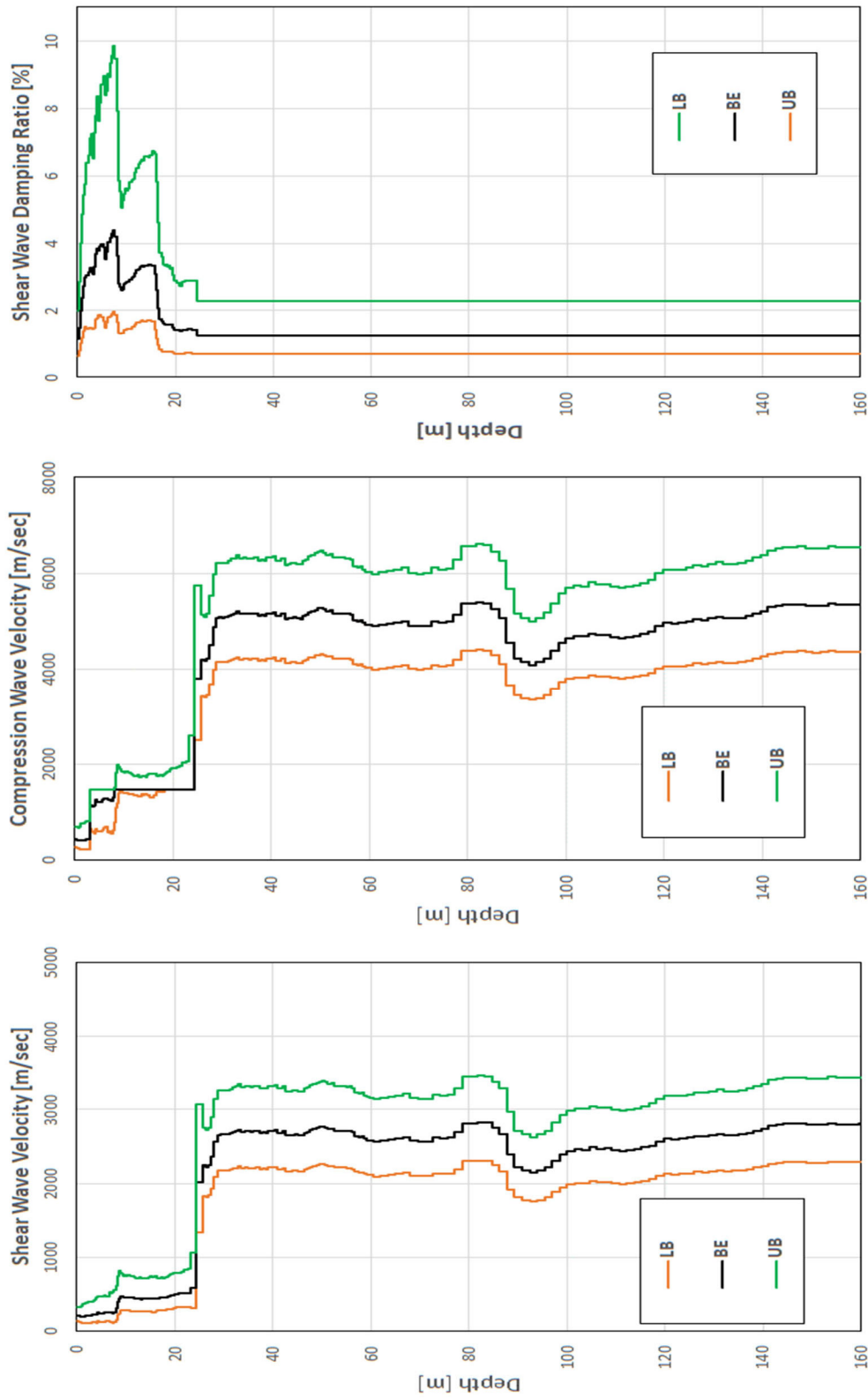


Figure 3.3-25: Subgrade Profiles for Bounding BWRX-300 Seismic Analyses

3.4 Protection Against Internal Hazards

This section discusses design basis internal hazards that could compromise the safety functions of SC1 SSC and preventive, and mitigation measures implemented in the design to eliminate their adverse effects. SC2/SC3 SSC credited in the fault evaluation with mitigating fault sequences initiated by internal hazards are also protected against internal hazards. For BDBA internal hazards, refer to Chapter 15, Sections 15.5 and 15.6.

The list of internal hazards considered in the BWRX-300 design is generated from the industry guidelines and the specifics of the BWRX-300 technology. These hazards are in accordance with CNSC REGDOC-2.5.2 (Reference 3.4-1), Section 7.4.1 supplemented by IAEA SSG-64 (Reference 3.4-2), which supersedes IAEA NS-G-1.11 (Reference 3.4-3) referenced in CNSC REGDOC-2.5.2. Screening methodology of internal hazards for safety analysis purposes and ultimately confirmation of adequacy of protection measures is identical to that of the external hazards presented in Section 3.3.

Protection and mitigation methods considered in the design are in line with the design safety objectives and D-in-D concept discussed in Subsections 3.1.1 and 3.1.6, respectively. They include the use of separation, barriers/shielding and monitoring programs as described in Subsection 3.1.5 to preclude unacceptable radiation releases following accidents due to internal hazards.

When applicable, loads generated by internal hazards are considered in the BWRX-300 design in compliance with requirements in Section 7.15.1 of CNSC REGDOC-2.5.2 and CSA N291 (Reference 3.4-4). Combination of loads from randomly occurring individual internal hazards is also considered in the design to ensure structure are adequately protected against internal hazards.

3.4.1 Internal Fires, Explosions and Toxic Gases

Protection and mitigation measures considered in the BWRX-300 design against internal fires, explosions, and toxic gases to comply with CNSC REGDOC-2.5.2, Section 7.4.1 are discussed in Subsections 3.4.1.1 through 3.4.1.3.

3.4.1.1 Internal Fires

Protection against internal fires is provided by:

1. A fire protection system to detect, notify, and suppress internal fires and the implementation of a comprehensive fire protection program.
2. Designing, locating, and compartmentalizing SSC to minimize the probability and effect of fires and explosions. Separation is provided between defense lines to the extent that defense lines are credited in the fault evaluation to mitigate the same event. Separation is provided using passive fire barriers to subdivide the plant into separate areas. Separation also confines the effects of fires to a single compartment or area minimizing the potential for adverse effects from fires on redundant SSC.

The fire protection system comprises fire alarms, automatic fire suppression, smoke removal, yard fire main with hydrants, building standpipe and hose stations, fire pumps, water supply and fire extinguishers. Details including design features and parameters of the fire protection system are provided in Chapter 9A, Section 9A.6.

The comprehensive fire protection program covers administrative controls, procedures, periodic inspections, maintenance, testing and training of personnel to ensure a safe shutdown of the plant and the health and safety of plant operators and the public. This program ensures the following life safety performance objectives are met during all operational modes and plant configurations:

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- Fire hazard controls are included in design and operational stages
- Fire notification means are provided
- Safe egress and/or areas of refuge are provided for occupants for use in the event of a fire
- A safe environment and other required support are provided for essential staff so they can perform all necessary plant control functions during and following a fire
- Protection for personnel performing emergency services is provided both during and following a fire
- Access and emergency lighting are provided for all areas where manual firefighting, evacuations, or operation field actions are expected

The fire safety assessments form a key element in the fire protection program. The fire safety assessments document a systematic review of the fire hazards at DNNP and the potential consequences of design basis fire events.

To satisfy requirements in CSA N293 (Reference 3.4-5) and CSA N293S1 (Reference 3.4-6), a fire hazard assessment is performed as discussed in Chapter 9A, Subsection 9A.6.10 to identify the specific fire hazards and fire protection capabilities for the plant. Chapter 9A, Subsection 9A.6.10 also discusses the fire safe shutdown analysis that evaluates fire effects on the safe shutdown systems to demonstrate compliance to the related requirements of the CSA N293 standard. Methodology for these evaluations is illustrated in Chapter 9A, Figures 9A.6.10-1 and 9A.6.10-2.

The BWRX-300 fire protection design satisfies requirements in CSA N293, CSA N293S1 and the applicable clauses of the NBC (Reference 3.4-7). The D-in-D principle discussed in Subsection 3.1.6 is used to achieve a high degree of fire protection by providing redundancy, diversity and balance in the fire protection measures included in the design to prevent, detect, suppress, and limit the effects of fires. A summary of fire protection measures for the Power Block buildings is provided in Subsections 3.4.1.1.1 and 3.4.1.1.2. Fire protection design features are discussed in Chapter 9A, Section 9A.6 and Chapter 9B, Sections 9B.2 and 9B.3.

3.4.1.1.1 General Protection Measures for Power Block Building Structures

The Power Block buildings are generally steel frame construction except for the RWB and the TB portion enclosing the main steam line which are of reinforced concrete construction, and the RB which is constructed using Steel Bricks™. To satisfy requirements in Section 7.12.1 of CNSC REGDOC-2.5.2, the walls, floors, and ceilings are designed to have 3-hour fire resistance ratings where required based on high combustible loadings (lubrication oil tank, for example) in the room or where an adjacent room contains equipment or systems from a different safety class division.

Corridors, stair enclosures and elevator hoistways that do not communicate between areas of different safety class divisions may have walls with a 2-hour minimum fire rating. Non-concrete interior walls are constructed of metal studs and gypsum wallboard to the required fire resistance rating.

Doors, including frames and hardware, penetrating rated fire barriers comply with the NBC or equivalent National Fire Protection Association (NFPA) ratings for that barrier.

The fireproofing of structural steel members where required by calculation based on combustible loading, is accomplished by application of an Underwriters Laboratory (UL) of Canada or equivalent UL - listed or Factory Mutual approved cementitious or ablative material, or by UL -

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listed or Factory Mutual approved boxing design. The required fire rating determines the fireproofing material thickness.

To satisfy requirements in Section 6.8.1.4 of CSA N293, wall and ceiling surface finishes are specified to meet flame spread index of 0-25 and smoke-developed index of 0-100 in accordance with CAN/ULS-S102 (Reference 3.4-8). Floor finishes have a flame spread rating of 0-300 and a smoke development classification less than 450 when tested in accordance with ASTM E648 (Reference 3.4-9) and ASTM E662 (Reference 3.4-10).

Suspended ceilings, including the lighting fixtures are of non-combustible construction in accordance with Section 5.7.1.1 of CSA N293.

To prevent the spread of spilled flammable and combustible liquids, including contaminated firefighting water, diking, draining or a combination of both is used to contain and control the volume of liquids in the buildings. Spill control measures are also included in the design to contain the contents of any above grade oil-filled vessel or tank larger than 208 liters and all tanks containing chemicals used in water/wastewater treatment or quality control.

3.4.1.1.2 General Protection Measures for Systems and Components

Complying with Section 6.8.4.1 of CSA N293, the BWRX-300 design minimizes the use of plastics, wood and other combustible materials in electrical equipment, cable raceways and wiring racks. Non-combustible and heat-resistant materials are used wherever practical throughout the unit.

Electrical cable in open tray raceways is limited to low voltage cable and meets IEEE 383 standards (Reference 3.4-11) in accordance with Section 6.8.4.4 of CSA N293. Vertical cables have a maximum vertical char of 1.5m when tested in accordance with the vertical flame tray test (Method 2-FT4) test in CSA C22.2 No. 2556 (Reference 3.4-12). Circuitry over 1000 volts is in conduit.

Certain areas of the plant have cable trays in stacked array. Where stacking of trays occurs, power cable, which is the most susceptible to internally generated fires, is routed in the uppermost tray to the greatest extent possible to provide isolation from other trays in the stack. A vertical separation is provided between horizontal cable trays. Groups of stacked trays for redundant SCN cables are separated horizontally.

Piping and cable tray penetrations are provided with fire-stops when penetrating fire rated barriers in accordance with Section 6.5.2.1 of CSA N293. Electrical cable fire-stops are tested to demonstrate a fire rating equal to the rating of the barrier they penetrate in accordance with Section 6.5.2.1 of CSA N293. As a minimum the penetrations meet the requirements of NUREG-1552 (Reference 3.4-13), including Supplement 1 of CSA C22.2 No 0.3 (Reference 3.4-14). The tests are performed or witnessed by a representative of a qualified, independent testing laboratory. The documented test results for the acceptable fire-stops are made a part of the plant design records.

To satisfy requirements in Section 6.3.1.1 of CSA N293, control, power, or instrument cables and equipment of redundant systems used for achieving and maintaining safe shutdown, are separated from each other by three hour rated fire barriers, except within inerted containment. Where the equipment of more than one division is required to be located within a single fire area (Control Room), cables are within conduit or a floor trench.

Fire separations are required to separate redundant fire safe shutdown systems and separate safe shutdown systems from other hazards.

Suitable design of the ventilation systems limits the consequences of a fire by preventing the spread of the products of combustion to other fire areas. Means are provided to ventilate,

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exhaust, or isolate the fire area as required, with consideration given to the consequences of ventilation system failure caused by the fire, resulting in a loss of control for ventilating, exhausting, or isolating a given fire area.

Filter media (excluding charcoal filters and High Efficiency Particulate Air (HEPA) filters) used in air handling systems meet the combustibility requirements of Class I in accordance with CAN/ULC-S111(Reference 3.4-15).

HVAC penetrations through 2-hour or 3-hour rated fire barriers are provided with fire/smoke dampers compatible with the rating of the fire barrier.

In accordance with Section 6.8.4.2 of CSA N293, electrical cabinets are designed to limit flame spread across cabinets.

3.4.1.2 Internal Explosions

The BWRX-300 fire hazard assessment evaluates the combustible loading along with the associated suppression requirements for each of the Power Block significant rooms and document the findings on the room data sheets.

Potential explosions of the following components are considered in the design:

- Batteries
- Diesel generators
- Switchgear
- Hydrogen tanks
- Miscellaneous hydrogen fires
- Offgas/hydrogen recombiners
- Transformers
- Transient combustibles
- Turbine auxiliaries

To satisfy requirements of CNSC REGDOC-2.5.2, Section 7.4.1, separation is provided between defense lines to the extent that defense lines are credited in the fault evaluation to mitigate the same event. Design measures considered include the use of fire barriers and blowout doors where flammable and combustible materials are located, and redundancy to enhance the reliability of systems.

Non-combustible and heat-resistant materials are also used, wherever practical throughout the Power Block, particularly in locations such as the containment and control rooms to reduce the risk of fires and explosions.

Administrative controls are also implemented to ensure stored chemicals and combustibles cannot ignite or react in sufficient quantities to impact nuclear safety. Collapse of structures, pipe whip, jet effects, and internal flooding as a result of internal explosions is also considered in the design.

3.4.1.3 Release of Internal Hazardous (Toxic) Gases

Plant personnel are protected from the adverse effects due to uncontrolled release of hazardous substances as a result of fires or internal explosions in compliance with CNSC REGDOC-2.5.2, Sections 7.4.1, 7.12.1 and 7.12.2.

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Preventive and mitigation measures against the release of hazardous and toxic gases include a proper design of ventilation systems to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire. Refer to Chapter 9A, Sections 9A.5 and 9A.6 for details of the BWRX-300 HVAC and fire protection systems, respectively.

Complying with CNSC REGDOC-2.5.2, Sections 8.10.1 and 8.10.2, the habitability of the MCR and SCR is ensured by designing the HVAC systems in these rooms to detect and limit the introduction of airborne radioactivity, toxic gas or smoke into the rooms as described in Chapter 6, Section 6.4. As stated in Chapter 6, Section 6.4.2.1, habitability requirements in the control rooms are maintained without credit for any breathing apparatus or protective clothing.

HVAC systems also supply outside air into the SCCV via the containment inerting system and exhaust inerting gases to provide a habitable environment for maintenance personnel during outage and maintenance periods.

3.4.2 Internal Flooding

SC1 SSC and SC2/SC3 SSC credited with flood event mitigation in the fault evaluation are protected against internal flooding in compliance with CNSC REGDOC-2.5.2, Sections 7.4.1 and 7.15.1.

Appropriate means are included in the design to prevent failure of SSC that are not designed to be submerged or exposed to spray as a result of flooding. They include the use of redundant system trains or divisions, structural barriers or compartments, curbs and elevated thresholds, and a leak detection system.

The design of the integrated RB structures considers the loads associated with the post-accident internal flooding of the containment following a DBA. The hydrostatic loads from the maximum possible water level are applied as pressures to the affected walls and mat foundation and applicable loads are also used for design of containment metal components.

The BWRX-300 internal flooding analysis identifies flooding sources, equipment in each area, and maximum internal flood levels in each area. The sources of internal flooding hazards include:

- Leaks and breaks in pressure retaining components
- High-energy piping breaks and cracks
- Moderate-energy piping through-wall cracks
- Pump mechanical seal failures
- Failure of isolating devices
- Storage tank ruptures
- Actuation of fire protection system
- Flow from upper elevations and nearby areas

The flood level in each internal area is determined by evaluating the inflow due to internal flooding sources, outflow from area compartment, and accumulation in each compartment area due to net flow.

3.4.3 Internal Missiles

Complying with CNSC REGDOC-2.5.2, Sections 7.4.1 and 7.15.1, the BWRX-300 design includes preventive and mitigation measures against internal missiles. The methodology used to

determine internal missiles is discussed in Subsection 3.4.3.1, while Subsection 3.4.3.2 provides the general preventive and mitigation measures considered in the design.

3.4.3.1 Sources of Internal Hazards

Potential missiles inside and outside containment and turbine missiles are identified, and their statistical significance determined. A statistically significant missile is defined as a missile that could cause unacceptable plant consequences or exceedance of radiological release limits. Criteria for determining statistically significant missiles are obtained from applicable portions of U.S. NUREG-0800 (Reference 3.4-16), SRP 3.5.1.1 through 3.5.1.3.

These missile sources could result from in-plant component overspeed failures or high-pressure system ruptures in compliance with CNSC REGDOC-2.5.2, Section 7.4.1. Rotating equipment failures include evaluations of pumps, fans, blowers, diesel generators, compressors, and turbines. Potential missiles from failure of pressurized components include valve bonnets, valve stems, pressure vessels, thermowells, retaining bolts, and blowout panels.

3.4.3.2 Protection from Internal Missile Hazards

Preventive and mitigative measures considered in the BWRX-300 design against internal missiles include the following:

- Locating the system or component in an individual missile-proof structure
- Physically separating redundant systems or components of the system from the missile trajectory path or calculated range
- Providing localized protection shields or barriers for systems or components
- Designing the particular structure or component to withstand the impact of the most damaging missile
- Providing design features on the potential missile source to prevent missile generation
- Orienting the potential missile source to prevent unacceptable consequences caused by missile generation

Refer to Subsection 3.3.5.4 for barrier design procedures for impactive loads, including internal missiles.

3.4.4 Pipe Breaks

BWRX-300 SC1 SSC and SC2/SC3 SSC credited with event mitigations in the fault evaluation are adequately protected from the consequences associated with a postulated rupture of high-energy piping and crack of moderate-energy piping inside and outside containment in compliance with Sections 7.4.1 and 7.7 of CNSC REGDOC-2.5.2 and IAEA SSG-64. Design bases and measures used to protect these SSC, referred to in the following subsections as essential SSC, are discussed in Subsections 3.4.4.1 and 3.4.4.2.

Effects that may result from a postulated rupture of high-energy piping include (1) pipe whipping, (2) pipe break reaction forces, (3) jet impingement forces, (4) blast waves, (5) sub-compartment pressurization, (6) decompression waves, (7) Missile generation, (8) environmental effects and (9) Flooding.

In the BWRX-300 design, a whipping pipe may hit a target and cause secondary failure in the target object depending on the thrust force, materials and sizes of the pipe/target. Severance in the target may occur and form a missile. A pipe whipping about a plastic hinge is not assumed to cause severance at the plastic hinge. Therefore, a break cannot cause the whipping pipe to act

as a missile. Criteria related to the evaluation of and protection against missiles, including those resulting from jet impingement or a whipping pipe, are provided in Subsection 3.4.3.

Protection against flooding and environmental effects as a result of high-energy pipe breaks are discussed in Subsections 3.4.2 and 3.9.4, respectively.

3.4.4.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Inside and Outside Containment

3.4.4.1.1 Design Basis

In addition to meeting requirements in CNSC REGDOC-2.5.2 and IAEA SSG-64, the BWRX-300 pipe break event protection also conforms to 10 CFR 50 Appendix A (Reference 3.4-17), General Design Criterion 4. To supplement the guidance provided in IAEA SSG-64, the design bases for this protection are in compliance with NRC Branch Technical Position (BTP) 3-3 (Reference 3.4-18) and BTP 3-4 (Reference 3.4-19) included in Subsections 3.6.1 and 3.6.2, respectively, of U.S. NUREG 0800. BTP 3-4 describes an acceptable basis for selecting the design locations and orientations of postulated breaks and cracks in fluid systems piping. Standard Review Plan Subsections 3.6.1 and 3.6.2 describe acceptable measures that could be taken for protection against the breaks and cracks and for restraint against pipe whip that may result from breaks.

Protection against pipe break event dynamic effects is provided to fulfill the following objectives:

1. Assure that the reactor can be shut down safely and maintained in a safe shutdown condition and that the consequences of the postulated piping failure are mitigated to acceptable limits with Loss of Preferred Power (LOPP).
2. Assure that containment integrity and leak tightness are maintained.

3.4.4.1.2 Design Evaluation

An analysis of pipe break events is performed to identify those essential systems, components, and equipment that provide protective actions required to mitigate, to acceptable limits, the consequences of the pipe break event.

Pipe break events involving high-energy fluid systems are evaluated for the effects of pipe whip, jet impingement, flooding, sub-compartment pressurization, and other environmental effects. Pipe break events involving moderate-energy fluid systems are evaluated for wetting from spray, flooding, and other environmental effects.

Adequate protection is provided against the effects of pipe break events for essential SSC to an extent that their ability to shut down the plant safely or mitigate the consequences of the postulated pipe failure is not impaired. This is accomplished by means of design features such as physical separation, jet shields and pipe whip restraints or by designing the SSC to accommodate applicable loads due to postulated pipe failure.

3.4.4.1.3 General Protection Measures

The direct effects associated with a particular postulated break or crack are mechanistically consistent with the failure. Thus, actual pipe dimensions, piping layouts, material properties, and equipment arrangements are considered in defining the following specific measures for protection against actual pipe movement and other associated consequences of postulated failures:

1. Protection against the dynamic effects of pipe failures is provided in the form of pipe whip restraints, equipment shields, and physical separation of piping, equipment, and instrumentation.

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2. As an alternative to protective measures, SSC identified as essential targets under postulated pipe breaks are analyzed to show that the essential functionality remains available under all applicable loading conditions resulting from the pipe break.
3. The precise method chosen depends largely upon limitations placed on the designer such as accessibility, maintenance, and proximity to other pipes.
4. Protection of SCN systems and components from the effects of postulated pipe breaks is considered where a resulting failure of the SCN system or component could lead to failure of an essential SSC. This includes consideration of coatings and insulation materials which could result in debris generation

Separation

To meet requirements in CNSC REGDOC-2.5.2, Section 7.6.1.1, the plant layout arrangement provides physical separation and segregation of essential SSC to the extent practicable to provide sufficient distance such that the effects of the failure cannot impair their essential functionality.

Physical separation between redundant safety class systems supporting Defense Line 3 (DL3) with their related auxiliary supporting features is another basic protective measure incorporated in the design to protect against the dynamic effects of postulated pipe failures.

Pipe Whip Restraints

Pipe whip restraints are used where pipe break protection requirements could not be satisfied using spatial separation, barriers, shields, analysis of the SSC or enclosures alone, and when it is necessary to limit the piping movement (pipe whip) following a postulated break. Restraints are located based on the specific postulated break locations determined in accordance with Subsection 3.4.4.2. After the restraints are placed, the piping and essential SSC are evaluated for jet impingement and pipe whip. For those cases where unacceptable jet impingement damage could still occur, barriers, shields, or enclosures are utilized in conjunction with pipe whip restraints.

The design criteria for restraints are given in Subsection 3.4.4.2.

Barriers, Shields, and Enclosures

Protection requirements are met through the protection afforded by the walls, floors, columns, abutments, and foundations in many cases. Where adequate protection is not already present because of spatial separation or existing plant features, additional barriers, deflectors, shields, or guard pipes are provided as necessary to meet the functional protection requirements of essential targets.

Structures acting as barriers, shields, or enclosures are designed to withstand the consequences of postulated pipe failures (i.e., pipe whip, jet impingement, pressurization of compartments, water spray, and flooding, as appropriate) in combination with other internal hazards such as missiles and loadings associated with the DBE within their respective design load limits. Procedures used to design these structures are provided in Subsection 3.3.5.4.

The BWRX-300 barrier design ensures a resistance to impulsive loads that is at least 20% greater than the steady-state magnitude of the impulsive load in accordance with regulatory guidance of U.S. NRC RG 1.243 (Reference 3.4-20), Regulatory Position 11.1.2 and provisions of CSA N291, Clause A.3.5.1.

3.4.4.1.4 Protective Features and Operator Actions

All available systems are considered for mitigating the consequences of a failure. In judging the availability of systems, account is taken of the postulated failure and its direct consequences such

as unit trip and LOPP, and of the assumed single active component failure and its direct consequences.

As stated in Chapter 15, Section 15.5, no operator actions are required to mitigate the effects of high-energy pipe breaks.

3.4.4.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

This section discusses the location criteria and methods of analysis needed to evaluate the dynamic effects associated with postulated breaks and cracks in high and moderate - energy fluid system piping inside and outside of the primary containment. This information provides the design basis for the requirements for protection of essential SSC.

3.4.4.2.1 Criteria Used to Define Break and Crack Location and configuration

The following subsections establish the criteria for the location and configuration of postulated breaks and cracks.

Definition of High-Energy Fluid Systems

High-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.1.4), are either in operation or are maintained pressurized under conditions where either or both of the following are met:

- Maximum operating temperature exceeds 93.3°C; and
- Maximum operating pressure exceeds 1.9 MPaG.

Definition of Moderate-Energy Fluid Systems

Moderate-energy fluid systems are defined to be those systems or portions of systems that, during normal plant conditions (as defined in Subsection 3.1.4), are either in operation or are maintained pressurized (above atmospheric pressure) under conditions where either or both of the following are met:

- Maximum operating temperature is 93.3°C or less; and
- Maximum operating pressure is 1.9 MPaG or less.

Piping systems are classified as moderate-energy systems when they operate as high-energy piping for only short operational periods in performing their system function but, for the major operational period, qualify as moderate-energy fluid systems. An operational period is considered short if the total fraction of time that the system operates within the pressure-temperature conditions specified for high-energy fluid systems is less than 2% of the total time that the system operates as a moderate-energy fluid system.

Postulated Pipe Breaks and Cracks

A postulated pipe break is defined as a sudden gross failure of the pressure boundary either in the form of a complete circumferential severance (guillotine break) or a sudden longitudinal split without pipe severance and is postulated for high-energy fluid systems only. For moderate-energy fluid systems, pipe failures are limited to postulation of cracks in piping and branch runs; these cracks affect the surrounding environmental conditions only and do not result in whipping of the cracked pipe. High-energy fluid systems are also postulated to have cracks for conservative environmental conditions in a confined area where high and moderate-energy fluid systems are located.

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The following high-energy piping systems are considered as potential candidates for a postulated pipe break during normal plant conditions and are analyzed for potential damage resulting from damage effects:

- Main Steam
- Isolation Condenser System
- Control Rod Drive System
- Reactor Water Cleanup System
- Condensate Feedwater System
- Condenser Offgas System (in TB)

Moderate-Energy piping systems considered as potential candidates for a postulated pipe crack include the following:

- Boron Injection
- IC Pool Cooling
- Shutdown Cooling
- Fuel Pool Cooling
- Passive Containment Cooling
- Containment Inerting

3.4.4.2.2 Location of Postulated Pipe Breaks

Postulated pipe breaks are selected as follows:

Piping in Containment Penetration Areas

Regions of high energy piping associated with reactor containment penetrations will consider analytical concepts to eliminate the need to consider postulated breaks. .

ASME Code Section III Class 1 High-Energy Piping in Areas Other Than Containment Penetration

With the exception of those portions of piping identified above as containment penetration areas, breaks in ASME Code, Section III, Class 1 piping (Reference 3.4-21) are postulated at the following locations in each piping and branch run:

- At terminal ends
- At intermediate locations where the maximum stress range or fatigue usage values exceed the limits specified in BTP 3-4

ASME Code Section III Class 2 and 3 High-Energy Piping in Areas Other Than Containment Penetration

With the exception of those portions of piping identified above as containment penetration areas, breaks in ASME Code, Section III, Class 2 and 3 piping (Reference 3.4-22) are postulated at the following locations in those portions of each piping and branch run:

- At terminal ends
- At intermediate locations where the maximum stress values exceed the limits specified in BTP 3-4

Non-ASME High-Energy Piping

Breaks in seismically analyzed non-ASME high-energy piping systems are postulated according to the same criteria as for ASME Code Section III, Class 2 and 3 high-energy piping systems.

Breaks in non-seismically analyzed, non-ASME high-energy piping systems are postulated at each terminal end and at each intermediate location of potential high stress or fatigue, such as pipe fittings, valves, flanges, and welded-on attachments

3.4.4.2.3 Location of Postulated Pipe Cracks

Postulated pipe crack locations are selected as follows:

Piping in Containment Penetration Areas

Regions of high energy piping associated with reactor containment penetrations will consider analytical concepts to eliminate the need to consider postulated cracks.

High-Energy Piping in Areas Other Than Containment Penetrations

With the exception of those portions of piping identified above as containment penetration areas, cracks in high-energy piping are postulated as follows:

1. For ASME BPVC Code, Section III Class 1 piping, at axial locations where the calculated stress range values exceed the limits specified in BTP 3-4.
2. For ASME BPVC Code, Section III Class 2 and 3 or non-ASME class piping, at axial locations where the calculated stress values exceed the limits specified in BTP 3-4.
3. For piping which has not been evaluated to obtain stress information, through-wall cracks are postulated at axial locations that produce the most severe environmental effects.

Moderate-Energy Piping in Areas Other Than Containment Penetrations

With the exception of those portions of piping identified above as containment penetration areas, through-wall cracks in moderate-energy piping adjacent to safety class SSC are postulated except where:

1. For ASME BPVC Code, Section III, Class 1 piping the calculated stress range values are less than the limits specified in BTP 3-4.
2. For ASME BPVC Code, Section III, Class 2 or 3 and non-ASME class piping, the calculated stress values are less than the limits specified in BTP 3-4.

Through-wall cracks, unless the piping system is exempted above, are postulated at axial and circumferential locations that result in the most severe environmental consequences.

Through-wall cracks are postulated in fluid system piping designed to non-seismic standards as necessary to assure that essential system and component functionality is maintained following a piping failure assuming a concurrent single active failure.

Moderate-Energy Piping in Proximity to High-Energy Piping

In cases where both high-energy and moderate-energy piping systems exist in a confined area, cracks are postulated in the piping system which leads to the more conservative environmental conditions.

3.4.4.2.4 *Types of Breaks and Cracks to be Postulated*

Pipe Breaks

The following criteria are used to postulate breaks in high-energy fluid system piping at the identified locations:

1. For the purposes of considering dynamic effects, circumferential breaks are postulated only in piping having a nominal diameter greater than 25 mm.
2. Longitudinal breaks are postulated only in piping having a nominal diameter equal to or greater than 100 mm.
3. Longitudinal breaks are not postulated at terminal ends.
4. Circumferential breaks are assumed at all terminal ends.
5. At each of the intermediate postulated break locations identified to exceed the stress and usage factor limits of the criteria in Subsection 3.4.4.2.2, consideration is given to the occurrence of either a longitudinal or circumferential break. Examination of the state of stress in the vicinity of the postulated break location is used to identify the most probable type of break based on the BTP 3-4 rules.
6. Where breaks are postulated to occur at each intermediate pipe fitting, weld attachment, or valve without the benefit of stress calculations, only circumferential breaks are postulated.
7. For a circumferential break, the dynamic force of the jet discharged at the break location is based upon the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient.
8. For longitudinal breaks, the dynamic force of the fluid jet discharge is based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location.

Pipe Cracks

The following criteria are used to postulate through-wall leakage cracks in high- or moderate-energy fluid system piping at the identified locations:

1. Leakage cracks are only postulated in piping having a nominal diameter greater than 25 mm.
2. The postulated cracks are oriented circumferentially to result in the most severe environmental consequences.
3. Crack openings are assumed as a circular orifice of area equal to that of a rectangle having dimensions one-half-pipe-diameter in length and one-half-pipe-wall thickness in width.
4. The flow from the crack opening is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments, based on a conservatively estimated time period to effect corrective actions.

3.4.4.2.5 Analysis Methods to Define Blowdown Forcing Functions and Response Models

Analytic Methods to Define Blowdown Forcing Functions

Analytical methods used to establish pipe rupture blowdown and jet thrust forcing forces are in accordance with ANSI/ANS 58.2 (Reference 3.4-23), Section 6.2.

The rupture of a pressurized pipe causes the flow characteristics of the system to change, creating reaction forces that can dynamically excite the piping system. The reaction forces are a function of time and space and depend upon fluid state within the pipe prior to rupture, break flow area, frictional losses, plant system characteristics, piping system, and other factors.

Criteria used for calculation of fluid blowdown forcing functions include the following:

1. Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed under loading).
2. For a circumferential break, the dynamic force of the jet discharge at the break location is based on the cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs are taken into account, as applicable, in the reduction of jet discharge.
3. All breaks are assumed to attain full size within one millisecond after break initiation.

Pipe Whip Dynamic Response Analysis Criteria

Dynamic forces are assumed to cause pipe whip reaction whenever moments cause excessive plastic deformation and the formation of a plastic hinge. Significant motion occurs only when the thrust force acts through an arm of sufficient length to induce a plastic hinge. This length is called the plastic hinge length. When the stiffness of a piping system is such that a plastic hinge cannot form, the pipe lateral displacement is assumed to be equal to the pipe diameter.

Pipe whip restraints are used to prevent piping from deforming plastically by forming hinges. They absorb blowdown force energy and limit jet impingement's zone of influence.

The prediction of time dependent and steady thrust reaction loads caused by blowdown of subcooled, saturated, and two-phase fluid from ruptured pipe is used as an input to evaluate the pipe whip dynamic response.

Pipe motion following circumferential breaks are assumed in the plane defined by the initial axis of the jet thrust force and rotation about a plastic hinge point, or at an intermediate point, such as the second change in direction, where the moment resisting capacity is less than straight pipe, provided the distance to this point is not significantly less than the plastic hinge length. The arc of the whipping pipe for planar motion is assumed to be limited to 180 degrees due to crimping at the plastic hinge and the pipe folding back against itself. Where a system consisting of piping, restraints and supporting structures is so complex that the assumption of planar motion is neither conservative nor realistic, the whip zone of influence can be conservatively enlarged to a region approaching a sphere with a radius equal to the distance between the break point and the first restraint. In lieu of this assumption, a more detailed elastoplastic analysis may be performed.

Longitudinal breaks in the form of axial split without pipe severance are postulated in the centre of the piping at two diametrically opposed points (but not concurrently) located so that the reaction force is perpendicular to the plane of the piping configuration and produces out-of-plane bending.

Alternatively, a single split is assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

For restrained longitudinal breaks or those breaks for which it can be shown that the pipe resists bending elastically, the zone of whip influence is taken to be all points within a distance of one pipe diameter from the axis of the pipe, unless physically limited by piping restraints, structural members or piping stiffness. For unrestrained longitudinal breaks in elbow fittings, the out-of-plane forces are assumed to cause whipping through a zone of influence described by the rotation of the fitting through 360 degrees about an axis which connects the two plastic hinges formed in the attached legs of piping.

A whipping pipe is considered capable of rupturing impacted pipes of smaller nominal pipe diameter, and of developing through-wall cracks in impacted pipes of equal or larger nominal pipe sizes with thinner wall thickness.

If a whipping pipe contains a large in-line mass (such as a valve), or if there is a change in the pipe shape (e.g., an elbow) near the end of the pipe, rupture of target pipes which are equal to or larger than the whipping pipe is considered.

Pipe Whip Dynamic Response Methods

Analytical models used to evaluate pipe whip dynamic response adequately represent the mass, inertia and stiffness properties of the piping system accounting for interaction effects of both the piping and pipe whip restraint.

Analytical methods used for piping response are based on those defined in ANSI 58.2, Section 6.3 and include complete system dynamic analysis, simplified dynamic analysis, quasi-dynamic analysis, energy balance analysis, and static analysis.

In cases where it is necessary to calculate stresses at locations which are far away from the break (e.g., in containment penetration break exclusion area), a more extensive model of the ruptured piping, supports, and pipe whip restraints is necessary.

If the snubbers or other seismic restraints are included in the piping model, they are modeled with the same stiffness used in the seismic analysis of the pipe. However, credit for seismic restraints cannot be taken if the applied load exceeds the ASME BPVC Code Section III (Reference 3.4-21, Reference 3.4-22 and Reference 3.4-24) Service Level D rating.

Pipe Whip Analysis Material Properties

Strain rate effects and other material property variations are considered in the pipe whip analysis of piping and pipe whip restraints.

Material properties and design limits consistent with those stated in ANSI/ANS 58.2, Sections 6.6.2 and 6.6.3 are applied for plastic deformation design of piping and pipe whip restraint design under dynamic and steady-state loading conditions.

3.4.4.2.6 *Dynamic Analysis Methods to Verify Integrity and Operability*

Jet Impingement Analyses and Effects on Essential Components

For each postulated circumferential and longitudinal break, an evaluation of jet impingement effects on essential targets including jet impinging force, thermal energy, and moisture is completed in accordance with the methodology criteria in this section.

In the case of circumferential breaks, jets are assumed to be oriented axially with respect to the pipe. In the case of longitudinal breaks, jets are assumed to be oriented radially.

Potential targets, or portions of targets adjacent to the jet boundary, are assumed to be impinged upon when reasonable variations in jet geometry or pipe movement are considered.

In evaluating the potential for jet impingement on specific targets, consideration is given to the movement of the jet centreline due to pipe whip, including pipe-restraint interaction.

Thermal and moisture effects on essential targets are determined in accordance ANSI/ANS 58.2, Section 7.4 and 7.5.

Modeling of the jet geometry and determination of the jet impingement force acting on a target is calculated according to ANSI/ANS 58.2, Sections 7.2, 7.3, and Appendices C and D, with modifications applied as identified in NUREG/CR-7275 (Reference 3.4-25).

Pipe Whip Effects on Essential Structures, Systems and Components

This section provides the criteria and methods used to evaluate the effects of pipe displacements on essential SSC following a postulated pipe rupture.

Pipe whip (displacement) effects on essential SSC can be placed in two categories: (1) pipe displacement effects on components (nozzles, valves, tees, etc.) which are in the same piping run that the break occurs in; and (2) pipe whip or controlled displacements onto external components such as building structure, other piping systems, cable trays and conduits.

(1) Pipe Displacement Effects on Components in the Same Piping Run

Essential components located in the same run as the postulated break meet the applicable ASME Code class limits for Service Level D and limits to ensure required operability.

(2) Pipe Displacement Effects on Essential Structures, Systems, and Components

The criteria and methods used to calculate the effects of pipe whip on external components consist of the following:

1. The effects on barriers, shields, or enclosures credited for protecting essential SSC are evaluated in accordance with the barrier design procedures given in Subsection 3.3.5.4.
2. If the whipping pipe impacts an essential system or component, mitigating measures are established to ensure essential functionality is not lost for the postulated break scenario.

Loading Combinations and Design Criteria for Pipe Whip Restraint

Pipe whip restraints are non-ASME code class components. As a result, other methods (i.e., testing) such as the use a reliable database may be used instead of the rules applied to ASME code class components for their design and sizing.

Pipe whip restraints are designed for both the thrust force at the pipe rupture location and the impact force of the pipe. The magnitude of these forces is a function of the pipe size, fluid temperature, and operating pressure.

Pipe whip restraints, as differentiated from piping supports, are typically designed only to function, and carry loads for an extremely low probability gross failure in a piping system carrying high-energy fluid. They are also required to remain functional following an earthquake up to and including the design basis DBE.

Pipe whip restraints are designed with sufficient clearances to prevent an increase in the pipe stresses by their presence during any normal mode of reactor operation or condition and are designed to allow for in-service inspection of the process piping with minimal obstruction.

3.4.4.2.7 Analytic Methods to Define Blast Wave Interaction to SSC

Sub-compartment pressurization due to postulated pipe breaks is considered where applicable.

3.4.4.2.8 Sub-compartment Pressurization

As discussed in Chapter 6, Subsection 6.3.2.2, the BWRX-300 containment sub-compartments do not contain large high-energy pipes and are, therefore, not subject to sub-compartment pressurization loads. For breaks outside the containment, mass and energy releases into the sub-compartments are calculated as described in Chapter 15, Subsection 15.5.9.2. Pressurization of the sub-compartments of the reactor building is calculated using the GOTHIC code described in Chapter 15, Subsection 15.5.1.2. The GOTHIC model of the RB includes all sub-compartments of the RB as lumped parameter volumes, including all flow passages between the rooms. This includes all doors and blowout panels which may be closed normally but may open if a pressure differential develops between the sub-compartments.

3.4.4.2.9 Decompression Waves

3-D thermal hydraulic code TRACG (See Chapter 15, Subsection 15.5.1.2) generates pressure time history in the annular region between chimney/shroud and RPV due to acoustic decompression wave as a result of a pipe break. Generated time history is part of the inputs to RPV primary structural FE model along with jet impingement, jet reaction and pipe whip restraint loads inputs to determine dynamic effects on RPV components, RPV internals and nozzles/pipings attached to RPV.

3.4.5 Other Internal Hazards

3.4.5.1 Hard Object Impact

Complying with CNSC REGDOC-2.5.2, Section 7.15.3 and IAEA SSG-64, the BWRX-300 design considers hard object impact loads resulting from the drop of heavy loads lifted and handled in areas where SSC required for safe shutdown of the plant are located.

Drops considered are those most likely to occur during the handling of plant equipment for maintenance or during spent fuel transfer operations. Other drops considered are drops as secondary effects of other internal hazards or external hazards discussed in Section 3.3.

In accordance with U.S. NRC RG 1.244 (Reference 3.4-26), the BWRX-300 heavy load is defined per the provisions of U.S. NUREG-0612 (Reference 3.4-27) as any load, carried in a given area after a plant becomes operational, that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool.

Critical heavy load handling evolutions considered are those where inadvertent operations or equipment malfunctions, separately or in combination, could:

- Cause a release of radioactivity
- Cause a criticality accident
- Cause the inability to cool fuel within the reactor vessel or within the Fuel Pool
- Prevent a safe shutdown of the reactor

Measures considered to reduce the potential of heavy load drops in the RB meet the D-in-D guidelines in U.S. NRC RG 1.244 and Section 5.1 of US NUREG-0612. They include a proper plant arrangement, the implementation of a heavy loads program as part of the plant procedures and effective means of lifting and transporting heavy loads designed to satisfy the single failure proof guidelines of Section 5.1.6 of US NUREG-0612.

Chapter 9A, Subsection 9A.8.1 provides an overview of the BWRX-300 heavy load program which identifies all heavy loads lifted during operation of the plant and the safe travel paths determined for their lifting. This program also manages the safe execution of heavy load evolutions.

Chapter 9A, Subsection 9A.8.1 describes the various cranes and hoists used to lift and transport heavy loads and applicable guides and standards used for their design. The RB polar crane main and auxiliary hoists meet the requirements of single failure proof systems in accordance with ASME NOG-1 (Reference 3.4-28). The refueling platform main hoist meets the requirements of a single failure proof hoist. Periodic inspection and maintenance of cranes are also planned to ensure their safe functioning.

3.4.5.2 Failure of Non-Structural Element

The failure of non-structural elements is considered in the BWRX-300 design.

Staircases and elevator shafts are evaluated and designed for interaction with plant Seismic Category A or B SSC in the event of DBE.

Architectural components and shielding blocks whose failure or dislocation could affect the safe operation of any Seismic Category A or B SSC are also evaluated for seismic interaction.

Scaffolding and other temporary structures considered a temporary alteration in support of maintenance are evaluated for seismic interaction as well, following the plant temporary structures procedure.

3.4.5.3 Electromagnetic Interference

Internal electromagnetic interference is caused by induction or radiation from installed equipment.

Complying with CNSC REGDOC-2.5.2, Section 7.5, safety class SSC are protected against electromagnetic interference to enable them to perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform.

Qualification requirements for protection against electromagnetic interference are presented in Subsection 3.9.5.

Plant grounding, lightning protection and electromagnetic compatibility systems and their design requirements are discussed in Chapter 8, Section 8.6.

3.4.6 References

- 3.4-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.4-2 IAEA Safety Standards Series No. SSG-64, "Protection against Internal Hazards in the Design of Nuclear Power Plants," International Atomic Energy Agency.
- 3.4-3 IAEA NS-G-1.11, "Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants," International Atomic Energy Agency.
- 3.4-4 CSA N291, "Requirements for Safety-Related Structures for Nuclear Power Plants," CSA Group.
- 3.4-5 CSA N293, "Fire Protection for Nuclear Power Plants," CSA Group.
- 3.4-6 CSA N293S1, "Supplement #1 to N293-12, Fire Protection for Nuclear Power Plants (Application to Small Modular Reactors)," CSA Group.
- 3.4-7 Canadian Commission on Building and Fire Codes, "National Building Code of Canada," National Resource Council of Canada.
- 3.4-8 CAN/ULC-S102, "Method of Test for Surface Burning Characteristics of Building Materials and Assemblies," Underwriters' Laboratories of Canada.

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- 3.4-9 ASTM E648, "Standard Test Method for Critical Radiant Flux of Floor-Covering Systems Using a Radiant Heat Energy Source," American Society for Testing and Materials.
- 3.4-10 ASTM E662, "Standard Test Method for Specific Optical Density of Smoke Generated by Solid Materials," American Society for Testing and Materials.
- 3.4-11 IEEE 383-2015, "IEEE Standard for Qualifying Electrical Cables and Splices for Nuclear Facilities," Institute of Electrical and Electronic Engineers.
- 3.4-12 CAN/CSA C22.2 No. 2556, "Wire and Cable Test Methods," CSA Group.
- 3.4-13 USNRC NUREG-1552, "Fire Barrier Penetration Seals in Nuclear Power Plants."
- 3.4-14 CAN/CSA C22.2 No 0.3-09, "Test Methods for Electrical Wires and Cables," CSA Group.
- 3.4-15 CAN/ULC-S111-13, "Standard Methods of Fire Tests for Air Filter Units," Underwriters' Laboratories of Canada.
- 3.4-16 USNRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition,"
- 3.4-17 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants."
- 3.4-18 USNRC BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."
- 3.4-19 USNRC BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment."
- 3.4-20 USNRC Regulatory Guide 1.243, "Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments."
- 3.4-21 ASME BPVC-III NB, "Section III - Rules for Construction of Nuclear Facility Components, Subsection NB: Class 1 Components," American Society of Mechanical Engineers.
- 3.4-22 ASME BPVC-III NCD, "BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 1-Subsection NCD-Class 2 and Class 3 Components," American Society of Mechanical Engineers.
- 3.4-23 ANSI/ANS 58.2-1988, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," American National Standards Institute/American Nuclear Society.
- 3.4-24 ASME BPVC-III NE-2021, "BPVC Section III - Rules for Construction of Nuclear Facility Components-Division 1 - Subsection NE – Class MC Components," American Society of Mechanical Engineers.
- 3.4-25 USNRC NUREG/CR-7275, "Jet Impingement in High-Energy Piping Systems."
- 3.4-26 USNRC Regulatory Guide 1.244, "Control of Heavy Loads at Nuclear Facilities."
- 3.4-27 USNRC NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."
- 3.4-28 ASME NOG-1, "Cranes, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," American Society of Mechanical Engineers.

3.5 General Design Aspect for Civil Engineering Works of Seismic Category Buildings and Civil Engineering Structures

This Section presents the design principles, design basis requirements, criteria and applicable codes and standards used in the design of the BWRX-300 civil structures, including their foundations in compliance with requirements in CNSC REGDOC-1.1.2 (Reference 3.5-1), Section 4.5.5.

Below are the key PSAR sections that impact the BWRX-300 Civil/structural design that should be reviewed along with this section:

- Chapter 1 which provides the DNNP general site and facility layout, a description of the BWRX-300 buildings, plant operational modes, principles of safety management and applicable codes & standards utilized in the design
- Chapter 2 which described the characteristics of the DNNP site on which the BWRX-300 facility is constructed
- Chapter 3, Section 3.1, which provides the general design aspects and D-in-D safety framework utilized in the BWRX-300 design
- Chapter 3, Section 3.2, which provides the general classification of BWRX-300 SSC and the approach used to establish these classifications
- Chapter 3, Sections 3.3 and 3.4, which provide methodology and general design requirements for protection against the effects of external and internal hazards
- Chapter 9B which provides specific information on compliance with the design rules for civil engineering works and structures

From the site layout presented in Chapter 1, Appendix A, Figure A1.4-1, the primary buildings in the BWRX-300 Power Block consist of the Reactor Building (RB) which houses the containment, Radwaste Building (RWB), Control Building (CB), Turbine Building (TB), and Reactor Auxiliary Bay. In the following sections, reference to the integrated RB structure is inclusive of the RB, containment, and containment internal structures, whereas RB is used to refer to the part of the integrated structure located outside of containment.

The seismic categorization of these structures is provided in Table 3.3-1. Per Subsection 3.2.3 and Table 3.3-1, the Seismic Category A integrated RB housing SC1 SSC has the utmost importance to safety and is credited for the safety analysis of the BWRX-300. RWB structures that support and protect equipment and components for storage and processing of highly radioactive gas, liquids and solid materials are categorized as RW-IIa. The CB, TB and Reactor Auxiliary Bay categorized as Non-Seismic structures are not credited in the safety analysis but are relied upon for their D-in-D function since they house and protect SC2 or SC3 systems and components. The RWB, CB, TB, and Reactor Auxiliary Bay can also affect the BWRX-300 safety considering their proximity to and interaction with the integrated RB structure.

Other civil structures for which design basis requirements are provided are the Pumphouse/Forebay structures and tunnels that support the condenser cooling and plant cooling water systems, and the Fire Pump Enclosure. For the location of these structures, refer to Chapter 1, Appendix A, Figure A1.4-1.

In accordance with Section 3.1 of CNSC REGDOC-1.1.5 (Reference 3.5-2) and Section 5.4 of CNSC REGDOC-3.5.3 (Reference 3.5-3), design principles for BWRX-300 structures are provided in a graded manner commensurate to their importance to safety. The primary focus of this Section is for the Seismic Category A integrated RB. Design principles for the RWB, CB, TB,

Reactor Auxiliary Bay, Pumphouse/Forebay and Fire Pump Enclosure structures are provided in Chapter 9B, Section 9B.3.

Remaining plant structures shown in Chapter 1, Appendix A, Figure A1.4-1 are not covered since they are not credited in the safety analysis.

3.5.1 General Design Principles for Seismic Category A Structures

The BWRX-300 Seismic Category A integrated RB structure is designed to meet the serviceability, strength, and stability requirements for all possible load combinations under the categories of normal operation, Anticipated Operational Occurrence (AOO) and DBA in compliance with requirements in CNSC REGDOC-2.5.2 (Reference 3.5-4), Sections 7.15.1 and 7.7. The robustness of the design to prevent potential release of radioactivity to the public and environment under Design Extension Condition (DEC) is considered in compliance with requirements in CNSC REGDOC-2.5.2, Sections 7.7 and 7.15.1 and is discussed in Subsection 3.5.6.

The integrated RB structure and its common foundation are primarily constructed using an advanced steel-plate composite system called Steel Bricks™. The Steel Bricks™ system has a configuration similar to the typical steel-plate composite system except that the tie-rods in the typical steel-plate composite system are replaced by diaphragm plates created by bending the plates that facilitates the fabrication process. The Steel Bricks™ modules used to construct the integrated RB comprise of a pair of steel faceplates, shear connectors, diaphragm plates, and concrete fill. The faceplates and concrete fill act as the composite system to provide strength and stability to the Steel Bricks™ system. The shear connectors facilitate the composite action between the faceplates and concrete fill, and the diaphragm plates act as shear reinforcement besides holding the system together. The design of the structures serving as the containment pressure boundary is performed in accordance with the provisions of ASME Boiler and Pressure Vessel Code (BPVC) as described in NEDC-33926P (Reference 3.5-5). The Steel-plate Composite Containment Vessel (SCCV) is designed in accordance with NEDC-33926P, as described in Subsection 3.5.3.1.

Similarly, the Class MC containment metal components are designed in accordance with the provisions of ASME BPVC, Section III, Division 1, Subsection NE (Reference 3.5-6).

ANSI/AISC N690 (Reference 3.5-7) that has been endorsed by U.S. NRC RG 1.243 (Reference 3.5-8), along with NEDC-33926P provide the specifications for the design, fabrication, construction, examination, and inspection of RB Steel Bricks™ and steel structures that do not provide the containment pressure boundary and for the containment internal structures.

These U.S. codes and standards are adopted for the BWRX-300 steel-plate composite structures (Steel Bricks™) since there are no equivalent standards or regulatory guidance in Canada.

Clause 6.1.2 of CSA N291 (Reference 3.5-9) permits the use of alternate design methods for design of nuclear structures and concrete containments in Canada. Requirements for design, fabrication, construction, examination, and testing of containment, containment internal structures, RB, and their foundations presented in Subsections 3.5.2 through 3.5.5 ensure compliance to the regulatory requirements in CNSC REGDOC-2.5.2 and meet the intent and ensure a level of safety and performance commensurate with the applicable Canadian standards.

3.5.1.1 Structural Analysis Criteria for Seismic Category A Structures

In accordance with requirements in CNSC REGDOC-1.1.2, Section 4.5.5 and CNSC REGDOC-2.5.2, Sections 7.13.1, 7.15.1, 7.22 and 8.6, the RB, containment and the containment internal structures are analyzed as one integrated structure, using ANSYS and ACS SASSI computer programs, to determine structural design demands resulting from various design loads and design

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load combinations. Evidence of qualification of these computer programs, including a description of the programs and extent of use, is presented in Appendix 3B.

The following Finite Element (FE) analyses are performed to obtain stress demands for the design of the BWRX-300 RB, containment, and containment internal structures:

- 1-g static SSI analyses
- Static and quasi-static analyses
- Thermal stress analyses
- Seismic SSI analyses

Static analyses provide design demands on the RB integrated structures from dead loads, live loads, earth pressure loads, hydrostatic and hydrodynamic loads, severe and extreme environmental loads, plant operating loads during normal operation, testing and abnormal plant conditions. Thermal analyses provide stress demands due to normal operating and accidental load conditions. Design Basis Earthquake (DBE) seismic demands are obtained directly from the results of one-step approach SSI seismic analyses discussed in Subsection 3.3.1.2.

The effect of interaction with the surrounding subgrade is incorporated in the analyses of the deeply embedded integrated RB by considering the surrounding soil and rock as a layered half-space continuum. The geotechnical design parameters used as input for the static and thermal analyses are developed as described in Subsection 3.5.2.2.

3.5.1.1.1 FE Model of Integrated RB Structure

To determine internal forces resulting from various loads and loading combinations, a detailed structural model is developed for the integrated RB, containment, and containment internal structures, including their foundations, penetrations, and openings, following the general FE modeling guidelines for the integrated RB structure discussed in Subsection 3.3.1.2 and NEDO-33914 Revision 2 (Reference 3.5-10), Section 5.1.1. The integrated structural FE model adequately represents the RB structural configuration for all main structural members and meets the mesh refinement and quality attributes required for calculation of structural stress demands. The use of the common model enables the FE results obtained from the different analyses to be directly combined in design load combinations per governing design codes.

Materials properties assigned to the integrated RB model depend on the analyzed loads and resulting stress responses. Unit weight properties are assigned to the models used for the 1-g static SSI analyses to adequately simulate gravity and earth pressure loads. The dynamic model of the integrated RB used for the seismic SSI analyses is assigned seismic mass inertia properties as discussed in Subsection 3.3.1.2.

As discussed in Subsection 3.3.1.2, stiffness properties are assigned to the SCCV and RB to reflect effective stiffness for load combinations without accidental thermal load. For load combinations with accidental thermal load, reduced stiffness is considered to account for the cracking effects on the redistribution of forces and moments. Spring elements are also used in the integrated FE element model to represent the stiffness of the connections between the different structural members that are designed to relieve stresses due to thermal expansion.

3.5.1.1.2 1-g Static SSI Analyses

Stress demands for the design of the integrated RB structure from dead loads and earth pressure design loads are obtained by applying the Earth gravity (1-g) load in the vertical direction to the SSI model described in Subsection 3.5.1.1. The 1-g static SSI analyses utilize the same substructuring method as the seismic SSI analyses described in Subsection 3.3.1.2. LB equivalent

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linear stiffness properties and UB unit weight properties assigned to the subgrade model used in the analyses are discussed in Subsection 3.5.2.2.

Maximum dynamic responses of the SSI system that are equivalent to its static response under 1-g gravity load are calculated by applying on the 1-g SSI analyses model an equivalent static 1-g excitation in the vertical direction as vertically propagating compression wave. To simulate 1-g excitation, a harmonic acceleration time history is used with:

- A low frequency equal to the analysis frequency increment, and
- An amplitude equal to the Earth's gravity (g).

The 1-g excitation is applied at control point located at the surface of the site free-field model.

Stress demands obtained from the one-step 1-g static SSI analyses include the effects of static earth pressures simulated by the interaction of the integrated RB structural model with the subgrade FE model. Shell elements at the surface of the subgrade are included in the SSI model to simulate the applicable overburden inertia loads from the surrounding Power Block foundations and other surcharge loads.

Contact springs are used at the interfaces of the RB structure with the surrounding subgrade as discussed in Subsection 3.3.1.2. In accordance with the FE modeling guidance in NEDO-33914 Revision 2, Section 5.1.1, the following stiffness properties are assigned to the contact springs in the models used for the 1-g static SSI analyses to provide UB lateral soil pressures on the RB below grade exterior walls:

1. The contact springs in the direction normal to the RB exterior walls are assigned properties representing UB stiffness conditions at the SSI interfaces.
2. The friction at the RB exterior walls is not considered by assigning very low stiffness properties to the contact springs in vertical and tangential direction.

Results obtained from these contact spring elements serve for calculation of earth pressures on the below grade RB shaft exterior wall and mat foundation.

Subgrade Modeling Assumptions for Deeply Embedded RB

Per NEDO-33914, Section 5.1.2, the following assumptions related to the modeling of the subgrade are introduced in the 1-g Static SSI analyses to enable an efficient calculation of stress demands on the RB structure due to pressure loads from soil and rock surrounding and supporting the RB shaft:

1. The properties of the subgrade materials are represented by linear elastic constitutive models
2. The non-linearities at soil-structure interfaces are not considered
3. The rock mass is assumed continuous and the presence of cavities, fracture zones, joints, bedding planes, discontinuities and other weak zones is not considered

The soil and rock strata in the 1-g static SSI models are modeled based on the principles of continuum mechanics using isotropic linear elastic properties. Possible fracture zones, joints, bedding planes, discontinuities and cavities in the rock are not explicitly included in the design SSI analyses models. Bounding properties assigned to the soil and rock materials are discussed in Subsection 3.5.2.2.

The effects of non-linearities at soil-structure interfaces are addressed by using elastic contact spring stiffness properties that provide bounding structural demands.

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Rock with disadvantageous fracture zones, joints, bedding planes and discontinuities is reinforced to create a more self-supporting rock mass. If needed, rock reinforcements are provided as initial ground support. The rock reinforcements and any other support provided during the excavation and construction may degrade and is inaccessible after construction. Therefore, the design addresses the rock loads remaining after the initial ground support degrades by including the potential weight of the rock in the static 1-g SSI analysis or by applying additional pressures on the RB outer shaft wall. Additional horizontal pressure loads are also applied on the model to account for possible residual stresses in the DNNP rock mass.

RB Design Earth Pressure Load Validation

Validations of the earth pressure loads are to be performed following the guidelines in Section 5.1.3 of NEDO-33914 Revision 2 to ensure the 1-g SSI static analysis provides conservative earth pressure design demands on the deeply embedded RB structure.

In accordance with requirements in CNSC REGDOC-2.5.2, Section 7.13.1 and NEDO-33914 Revision 2, Section 4, Foundation Interface Analyses (FIA) are performed on models representative of the non-linear constitutive behavior of soil and rock materials surrounding the RB shaft and employ non-linear interface modeling features capable of capturing the effects of non-linearities at the subgrade structure contact surfaces. The results of the FIA are to be used for validation of the design earth pressures following the guidance of Section 5.1.3 of NEDO-33914 Revision 2.

3.5.1.1.3 Static and Quasi-Static Load Analyses

In accordance with requirements in CNSC REGDOC-2.5.2, Section 7.15.1, the following static and quasi-static analyses are performed on the integrated RB FE model to calculate structural stress demands due to:

- Live loads
- Crane loads
- Structural Integrity Test (SIT) and accident condition containment internal pressure load including differential containment and RB sub-compartment loads
- Horizontal hydrostatic pressure loads on pool walls
- Groundwater pressure loads on the integrated RB common mat foundation and below-ground exterior wall
- Extreme wind and tornado loads on RB roof and exterior wall
- Rain and snow loads
- Seismic water sloshing and breathing mode quasi-static pressure loads on pool walls
- Quasi-static pressure High Energy Line Break (HELB) loads (jet impingement, blast loads)
- Equipment and pipe reaction loads including RPV reaction loads.
- Post-accident internal flooding loads

The analyses of global static and quasi-static loads that can affect the global response of the integrated RB consider the effect of subgrade stiffness. Following the sub-structuring methodology, design demands from these loads are obtained from subgrade stiffness impedance analyses performed on models consisting of two parts:

- Super-element representing LB stiffness of the subgrade surrounding the RB, and

- Integrated FE model of the RB, containment and containment internal structures described in Subsection 3.5.1.1.1.

The super-elements define the stiffness of the subgrade at the nodes of the RB interfaces with the surrounding soil. The stiffness properties of the super-elements are developed using a layered 3-D solid FE model. Subgrade stiffness properties assigned to the super-elements are described in Subsection 3.5.2.2. To adequately simulate half-space boundary conditions, the depth of these models is deeper than three times the largest foundation dimension. The horizontal extent of these models is more than three times the RB shaft diameter.

The nodes of the super-element are coincident with the nodes of the integrated RB FE structural model. The coincident super-element and structural model nodes are connected by contact spring elements as described in Subsection 3.5.1.1.2. LB stiffness properties are assigned to these contact spring elements to yield larger structural deformations and conservative design stress demands. Equivalent linear subgrade stiffness properties assigned for the subgrade stiffness impedance static analyses are discussed in Subsection 3.5.2.2.

Fixed bases analyses are performed for the local loads with smaller magnitudes that do not affect the Integrated RB mat common mat foundation or global response.

Demands due to hydrostatic lateral pressure loads are obtained from static analyses of the integrated RB model with vertical supports applied to all mat foundation nodes. Demands from the upward buoyant pressures on the mat foundation are obtained from a static analysis of the integrated RB structural model with vertical supports at the nodes connecting the RB exterior wall with the mat foundation and horizontal supports established at the central node of the mat. The results from the two groundwater load analyses are enveloped and then combined with the results of the 1-g SSI analysis cases to obtain earth pressure and groundwater load demands for the design of integrated RB structure.

Additional Rock Pressure load analyses are performed to account for possible residual horizontal stresses in the DNNP rock strata. Two boundary conditions are considered for these analyses that result in conservative stress demands:

1. Vertical supports established at all mat foundation nodes and horizontal supports established at the central node of the mat; and
2. Vertical supports at the nodes connecting the RB exterior wall with the mat foundation and horizontal supports established at the central node of the mat.

The results of these two sets of additional static rock pressures analyses are enveloped and then combined with the results of the 1-g SSI analyses to ensure the RB structural design adequately addresses the effects of anisotropic and heterogenous rock behavior and accounts for potentially unstable rock mass loads.

3.5.1.1.4 Thermal Stress Analyses

To calculate structural stress demands due to the normal operating and DBA temperature loads, sub-structuring thermal stress analyses are performed on the integrated RB FE structural model coupled with super-element representing UB stiffness of the subgrade.

Stiffness properties are assigned to the Steel Bricks™ shell elements to account for the stiffness reduction effects under normal operating and DBA temperature loads. The corresponding structural stiffness conditions are used for the analyses for design loads that occur in combination with the normal and accident thermal loads.

For the thermal analyses, UB stiffness properties are assigned to the super-element modeling the subgrade and to the contact elements modeling the soil-structure interfaces resulting in

conservative thermal stress demands for the design of the RB and containment structures. Equivalent linear subgrade stiffness properties assigned for the thermal stress analyses are discussed in Subsection 3.5.2.2.

3.5.2 Foundations

This section presents general design rules for the common Steel Bricks™ mat foundation supporting the integrated RB structure. Design rules for other foundations are discussed in Chapter 9B, Section 9B.3.

3.5.2.1 Applicable Codes, Standards and Other Specifications

Applicable codes, standards and specifications for the containment and RB common Steel Bricks™ foundation are the same as those for the superstructures.

The jurisdictional boundary for the application of the NEDC-33926P to the containment is the portion within the perimeter or exterior surface of the SCCV as shown in Figure 3.5-1.

The jurisdictional boundary for application of the ANSI/AISC N690 to the non-pressure retaining portion of the common foundation is the portion spanning from the exterior surface of the SCCV to the exterior surface of the RB (See Figure 3.5-1).

3.5.2.2 Bounding Subgrade Design Parameters

Bounding subgrade parameters are determined based on data available prior to the completion of the complete characterization of geotechnical and seismic conditions at the DNNP site presented in Chapter 2, Section 2.7. These conservative subgrade property inputs adequately address uncertainties related to the use of incomplete characterizations of the DNNP site geotechnical and seismic conditions.

Based on the information from the available groundwater flow patterns and conditions at the DNNP site provided in NK054-REP-01210-00011 (Reference 3.5-11) and NK054-REP-07730-00005 (Reference 3.5-12), an Upper Bound groundwater level at elevation 85 m CGD corresponding to a depth of 3 m below the plant grade at elevation 88 m CGD is considered a parameter for the bounding design.

The geotechnical and hydrological investigations of the DNNP site have been completed and bounding subgrade design parameters determined (see Chapter 2, Subsection 2.7.5). The data collected from ground water measuring wells at the DNNP site indicate an upper bound nominal water table at a shallower depth of 2 m. The increase of an additional meter in the nominal ground water table elevation results in a 6% higher magnitude of the total force from ground water pressure load than the one calculated using the bounding design ground water table at 3 m depth.

The exterior RB wall is the main structural member resisting the below grade lateral pressures applied on the RB integrated structures. These below grade lateral loads include the static earth pressure, ground water hydrostatic pressure, and additional rock pressure that account for a large majority of the demand on the below grade portion of the exterior RB wall in approximately equal shares. Therefore, the effect of the marginal 6% increase in the ground water pressure, that represents no more than a third of the total structural demand on the exterior RB, is negligible and well bounded by the available structural design margins (see Chapter 9B, Appendix 9B.G).

Identification and evaluation of potentially liquefiable cohesionless soil strata under the BWRX-300 Power Block structures is performed in accordance with CSA N289.3 (Reference 3.5-13) and in compliance with requirements of CNSC REGDOC-2.5.2, Section 7.15.1.

3.5.2.2.1 Bounding Equivalent Linear Subgrade Static Profiles

As described in Subsection 3.5.1.1, the structural design demands due to static earth pressures on the RB below grade exterior walls are obtained from the 1-g static analyses of the integrated RB FE model embedded in a layered half-space continuum model representing the surrounding soil and rock. To account for the interaction of the RB integrated structures with the surrounding subgrade, super-elements representing the stiffness properties of the layered subgrade materials are used in the static and thermal analyses, as described in Subsection 3.5.1.1.

The 1-g static SSI analyses, subgrade impedance analyses and thermal stress analyses use profiles of bounding equivalent linear soil and rock properties developed using information from the existing laboratory tests and in-situ measurements taken in the vicinity of the DNNP site and following the recommendations of NEDO-33914 (Reference 3.5-10), Section 5.2.1. They consist of:

- Effective unit weight that for soil materials below groundwater table are calculated as the total unit weight of soil minus the unit weight of water
- Elastic and shear Modulus representing linearized stiffness properties of the soil and rock for long-term static loading conditions
- Soil and rock Poisson's ratios representative of at-rest lateral pressure conditions

The bounding equivalent linear subgrade static profiles reflect anticipated as-built conditions at the site after construction of the BWRX-300 SMR that include engineered fill from about elevation 80 to 82 m CGD to the final grade at elevation 88 m CGD. The layering of the engineered fill, in-situ soil and rock materials in these bounding subgrade static profiles corresponds to the layering of dynamic subgrade properties described in Subsection 3.3.1.1.1 that are used as input for the DNNP site-specific seismic analyses.

Bounding static soil properties of in-situ soil materials are determined based on the results of in-situ tests and laboratory test results presented in the 2012 NK054-REF-01210-0418696 (Reference 3.5-14) and the 2013 NK054-REP-01210-00098 (Reference 3.5-15). SPT N-values are converted to N_{60} values (N value at 60 percent hammer energy) based on measured or assumed hammer energies for the automatic hammer and drill rigs used in the investigation, per the 2012 NK054-REF-01210-0418696 (Reference 3.5-14).

The drained friction angles for the soil layers are estimated using correlations based on relative density, N_{60} , and vertical effective stress for cohesionless soils provided in the 1986 DM 7.01 (Reference 3.5-16), the 1990 EPRI EL-6800 (Reference 3.5-17) and the 2016 Soil Properties and their Correlations (Reference 3.5-18). The different correlations are equally weighted to determine the final average drained friction angle value. The values for the coefficient of earth pressure at rest (K_0) are determined using effective angle of friction (ϕ_s) and over-consolidation ratio based on the 2021 NEDO-33914 (Reference 3.5-10).

Bounding properties of the engineered fill are developed based on the information obtained from compaction tests that were completed for the upper till, intermediate glacio-lacustrine, and lower till units presented in the 2009 DNNP Existing Environmental Conditions NK054-REP-07730-00005 (Reference 3.5-12). Based on the result from standard compaction tests, the relative density (D_r) and N_{60} values of the compacted soils are estimated. Relative density is estimated using the empirical relationship between D_r and compaction in the 2009 NK054-REP-07730-00005 (Reference 3.5-12). A relative compaction range of 85 to 100 percent is considered reasonable to cover the potential variations in placement and compaction of the on-site soils. The E_{st} of the compacted fill is determined from the estimated N_{60} values described in the 2016 Soil Properties and their Correlations (Reference 3.5-18) similar to the in-situ soils. The drained

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friction angle for the engineered or compacted fill is assumed to be similar to the in-situ soils that will be excavated.

Bounding values for the linearized E_{st} of the rock masses at the DNNP site are estimated based on the intact rock modulus (E_{ri}) and the rock mass classification determined from results of the site investigation program and an estimated Geologic Strength Index for the different bedrock formations. Results of Uniaxial Compression Tests performed on intact rock specimens and V_s and V_p measurements can serve as the basis for development of E_{ri} values. The ν_{st} values for rock masses are developed based on V_s and V_p measurements and the level of rock fracturing.

The intact rock elastic properties are estimated from shear wave velocities using elastic theory as outlined in the 2021 NEDO-33914 (Reference 3.5-10). Results of laboratory measurements on recovered rock provided in the 2012 NK054-REF-01210-0418696 (Reference 3.5-14) and the 2013 NK054-REP-01210-00098 (Reference 3.5-15) are also used to estimate the intact rock elastic properties of the Blue Mountain (Whitby) and Lindsay Formations. The laboratory measured elastic modulus values in the Blue Mountain (Whitby) and Lindsay Formations were, on average, 94 and 75 percent, respectively, of the estimated values from the V_s . This comparison likely represents the different strain levels as well as potential damage from rock coring. Based on this comparison, the estimates of the modulus for intact rock from bedrock units below Lindsay Formation are reduced by a factor of 0.75. In the Blue Mountain (Whitby) and Lindsay Formations (Lindsay 1), the lower intact rock deformation modulus from the laboratory testing results is used.

The rock ν_{st} values are based on the laboratory measured values and the estimates from V_s and V_p measurements. Based on this comparison the seismic wave estimated values are used without modification. Blue Mountain (Whitby) Formation is assigned ν_{st} value of 0.58 based on an at-rest stress ratio (K_0) that includes the estimated horizontal rock stresses at the site provided by Lo and Lukajic in (Reference 3.5-19) that are higher than the vertical stresses.

Table 3.5-1 provides a summary of bounding linearized static properties for in-situ soil and engineered fill layers in the as-built profiles. The summary of bounding static properties for the rock layers at the DNNP site are provided in Table 3.5-2.

UB values for soil effective unit weight and Poisson ratio are used as input for the static 1-g SSI analysis to conservatively address uncertainties in the consideration of earth pressure loads. In accordance with the guidance of NEDO 33914, Section 5.2.1.1, the soil Poisson ratios (ν_{st}) are calculated as follows using the at-rest lateral (k_0) coefficient values provided in Table 3.5-1:

$$\nu_{st} = \frac{K_0}{1 + K_0}$$

LB soil and rock stiffness properties are used for the static analyses including the 1-g SSI analyses resulting in larger deformation at soil-structure interfaces and conservative design stress demands. Thermal stress analyses are performed using UB soil and rock stiffness properties resulting in conservative thermal stress demands.

3.5.2.2.2 Soil Bearing Stability

The stability of soil supporting the BWRX-300 structural foundations is demonstrated in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.12.2 and per the regulatory guidance of US NUREG-0800 (Reference 3.5-20), SRP 2.5.4.10, and IAEA Safety Guide No. NS-G-3.6 (Reference 3.5-21).

The bearing capacity of the rock supporting the RB mat foundation is discussed in Chapter 2, Subsection 2.7.3.3.

Since the RB is deeply embedded, the bearing surface of the common foundation is below the depth of frost action to meet the requirements of NBC (Reference 3.5-22), Article 4.2.4.4.

Chapter 2, Subsection 2.7.3.3 also discusses the bearing capacity of the component in-situ soil materials supporting the shallow foundations surrounding the RB.

The calculation of the dynamic bearing pressure demands under DBE loads from the results of the seismic SSI analyses is described in Subsection 3.3.1.2.

Per Article 4.35 of IAEA Safety Guide No. NS-G-3.6, safety factors against potential bearing capacity failure of the subsurface materials depend on the method of bearing capacity evaluation and site conditions. If a conventional bearing capacity method is used, safety factors are not less than 3 under static loads and 1.5 under loads that include DBE.

3.5.2.2.3 Foundation Stability

Foundation stability is assessed against sliding and overturning due to earthquakes, wind and tornados, and flotation in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.12.2, following the regulatory guidance of US NUREG-0800, SRP 3.8.5 and in accordance with Clause 5.9 of CSA N289.3.

Explicit sliding and overturning stability evaluations are not performed for the deeply embedded RB since, in accordance with Sections 7.2.1 and 7.2.2 of ASCE/SEI 43 (Reference 3.5-23), its centre of gravity is below the grade elevation, and the structure is inherently stable against sliding and overturning. The foundation stability of the surrounding RWB, CB, TB, and Reactor Auxiliary Bay that are supported by surface mounted foundations is checked to ensure that there is no adverse interaction with the Seismic Category A RB during a DBE level event. Stability of the surface mounted foundations surrounding the RB under DBE loads is evaluated using the results of the seismic SSI analyses as described in Subsection 3.3.1.2.

Safety factors against sliding and overturning under normal operating conditions that include unfactored combination of dead loads, soil pressure loads, and design wind, and accidental conditions that include combination of dead loads, soil pressure loads, and DBE loads are presented in Table 3.5-3.

3.5.2.3 Loads and Load Combinations

3.5.2.3.1 Design Loads

Design loads of the containment and RB common mat foundation are those of the superstructures described in Subsections 3.5.3.2 and 3.5.5.2.

For foundation stability against flotation, the site-specific design basis flood is considered.

3.5.2.3.2 Design Load Combinations

Design load combinations of the containment and RB common mat foundation are those of the superstructures described in Subsections 3.5.3.2 and 3.5.5.2.

For the stability against flotation of the integrated RB foundation, the load combination is in accordance with U.S. NUREG-0800, SRP 3.8.5, where the design basis flood is considered in combination with the dead load.

3.5.2.4 Design and Analysis Procedures

The design of the deeply embedded foundation and foundation stability evaluations are in compliance with requirements in CNSC REGDOC-2.5.2, Section 7.15.1 and follow the BWRX-300 specific criteria and guidelines in NEDO-33914 Revision 2.

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The containment and RB common mat foundation is analyzed using the methods where the transfer of loads from the foundation mat to the supporting foundation media is determined by elastic methods. Demands for the design of the common mat foundation are obtained from the structural analyses described in Subsection 3.5.1.1 performed on the integrated RB structural model that include the effects of interaction of the structure with the surrounding subgrade and the effects of the foundations of the surrounding Power Block buildings.

The common Steel Bricks™ foundation mat is represented by thick shell elements in the integrated FE model. Properties assigned to the shell elements representing the common Steel Bricks™ foundation in the dynamic FE model used for the seismic SSI analyses are described in Subsection 3.3.1.2. Properties assigned to the foundation shell elements in the integrated FE models used for the static and thermal stress analyses are described in Subsection 3.5.1.1.

The containment foundation is designed in accordance with NEDC-33926P, consistent with U.S. NRC RG 1.136 (Reference 3.5-24). The non-pressure retaining portion of the containment-RB common foundation mat is designed to ANSI/AISC N690, supplemented by U.S. NRC RG 1.243 and NEDC-33926P.

Effects of normal and differential settlement of BWRX-300 structures is considered in the design and include consideration of the effects of fluctuating ground water on the foundations per CNSC REGDOC-2.5.2, Section 7.15.1, and CSA N291, Clause 6.4.3.

As mentioned in Subsection 3.5.1.1, contact springs are used to represent the stiffness properties of the foundation-subgrade interface. Vertical spring force results obtained from these spring elements serve for calculations of foundation bearing stresses.

3.5.2.5 Foundation Design Criteria

The structural acceptance criteria for the containment and RB common foundation are the same as those for their respective superstructures. Refer to Subsection 3.5.2.2 for safety factors considered for soil bearing and foundations stability.

3.5.2.6 Materials, Quality Control and Special Construction Techniques

3.5.2.6.1 Foundation Materials

Materials used for the construction of the containment and RB common foundation mat are the same as those of the superstructures discussed in Subsections 3.5.3.5 and 3.5.5.5.

3.5.2.6.2 Foundation Quality Control

Refer to Subsections 3.5.3.5 and 3.5.5.5 for discussion.

3.5.2.6.3 Foundation Special Construction Techniques

Refer to NEDO-33914 Revision 2, Section 1.4 for the preferred construction approach for the deeply embedded RB.

3.5.2.7 Testing and In-Service Inspection Requirements

The foundation inspection and testing follow the guidance of NEDO-33914 Revision 2, Sections 3.2.1 and 3.4, and also NEDC-33926P.

3.5.3 Containment

The BWRX-300 containment comprises a Steel-plate Composite Containment Vessel (SCCV), a steel containment closure head and other Class MC components. As described in Subsection 3.5.1, the BWRX-300 SCCV is constructed of Steel Bricks™.

3.5.3.1 Applicable Codes, Standards and Other Specifications

Codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and in-service inspection of the BWRX-300 containment are listed in Chapter 1, Appendix B.

The design of the BWRX-300 containment boundary structures, including the SCCV, containment closure head and other Class MC components complies with the regulatory requirements in CNSC REGDOC-2.5.2. The analysis and design, fabrication and testing of the SCCV is in accordance with the provisions of NEDC-33926P, which are based on analytical and engineering principles, including use of experimental results. Additional analysis and design requirements in U.S. NUREG-0800, SRP 3.8.1 and U.S. NRC RG 1.136 for concrete containment are also met, as applicable. The compliance with the provisions of NEDC-33926P and the regulatory guidance of U.S. NUREG-0800, SRP 3.8.1 and U.S. NRC RG 1.136 ensures a level of safety and performance for the SCCV compliant with CNSC REGDOC-2.5.2.

The containment closure head, and the other Class MC components that are part of the containment pressure boundary are analyzed, designed and inspected following the provisions of ASME Section III, Division 1, Subsection NE, ensuring compliance with the regulatory guidance of CNSC REGDOC-2.5.2.

3.5.3.1.1 Containment code Jurisdictional Boundary

For code applicability, the SCCV is designed in accordance with ASME BPVC Section III requirements. The code jurisdictional boundary for application of Section III of ASME BPVC to the SCCV is shown in Figure 3.5-1. The SCCV boundary extends to the:

1. Outside diameter of the SCCV wall from mat foundation to containment top slab including the welds connecting the SCCV with the RB structural members
2. Portion of the foundation mat foundation under SCCV including the welds connecting the SCCV portion of the mat foundation with the remaining part of the RB mat foundation
3. Containment top slab from containment closure head opening to the outside diameter of the SCCV including the welds connecting the slab with the RB structural members

The BWRX-300 containment closure head and other containment boundary metal components are ASME Code Class MC. The code jurisdictional boundary for application of ASME BPVC Section III, Division 1, Subsection NE, Class MC to the containment closure head, access hatches and penetrations are shown in Figure 3.5-2, Figure 3.5-3 and Figure 3.5-4, respectively.

The SCCV along with the containment closure head, access hatches and penetrations, provide the primary containment function as a leak-tight pressure boundary confining radioactive substances in different plant conditions. Although the internal RPV support pedestal, bioshield and other containment internal structures are completely within the containment, these internal structures do not serve any pressure retaining function and are, thus, outside the scope of ASME Code applicability. The design of welds connecting the containment internal structures to the containment pressure boundary are under ASME jurisdiction. The connections of the RB walls and floors to the outside face of the SCCV wall are outside ASME code jurisdiction, with the exception of attachment welds. Attachment welds are designed to follow ASME quality assurance and welding procedures and inspection requirements.

3.5.3.2 Load and Load Combinations

3.5.3.2.1 Containment Design Loads

Loads used in the design of the BWRX-300 containment structures, comprised of the SCCV, containment closure head, and other Class MC components, satisfy the loading requirements of

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the applicable regulations, design codes and standards in Subsection 3.5.3.1. These loads are in accordance with the provisions of ASME III Division 1, Subsection NE, ASME III Division 2 (Reference 3.5-25) and NEDC-33926P.

Loads considered in the design of the BWRX-300 containment structures are:

- Normal Loads:
 - Dead load (D) which includes permanent dead weight of structural and shielding elements, permanently located equipment and hydrostatic pressure of liquids in various pools
 - Live loads (L, L_o) which include any moveable equipment loads and other loads that vary in intensity and occurrence
 - Indirect Snow (S) and Rain (R) Loads
 - Thermal (T_o) effects and loads during normal operating, startup, or shutdown conditions
 - Pressure (P_o) loads resulting from the pressure difference between the interior and exterior of the containment, considering both interior pressure changes because of heating or cooling and exterior atmospheric pressure variations
 - Pipe reactions (R_o) during normal operating or shutdown conditions based on the most critical transient or steady-state conditions
 - Construction loads applied to the containment from start to completion of construction. The definitions for D, L and T_o given above are applicable, but are based on actual construction methods and/or conditions
 - Pressure Variant loads (P_v) which are the external pressure loads arising from variation either inside or outside the SCCV
 - Indirect Lateral Soil and groundwater pressure loads (H)
- Pre-operational Testing Loads:
 - Thermal (T_t) effects and loads during the SIT or Integrated Leak Rate Test (ILRT)
 - Test Pressure (P_t) Loads applied during the SIT or ILRT
- Severe Environment Loads:
 - Indirect design Wind Load (W) defined in Subsection 3.3.2
- Extreme Environmental Loads:
 - Indirect Tornado (W_t) Loads defined in Subsection 3.3.2
 - DBE seismic (E_s) loads determined for DNNP site-specific conditions taking into account SSI effects, as discussed in Subsection 3.3.1, and include associated hydrodynamic loads and dynamic incremental soil pressures
- Abnormal Plant Loads:
 - Accidental Thermal effects (T_a) due to LOCA
 - Accidental Pressure (P_a) loads within the containment generated by a LOCA
 - Accidental Pipe (R_a) reaction loads that consist of pipe reactions (including R_o) from thermal conditions generated by design basis accidents such as LOCA and DBE

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- Local effects on containment due to LOCA (R_r) and Blast Loads (R_b) which includes:
 - R_{rr} load on the containment generated by the reaction of a ruptured high-energy pipe during the postulated event of the DBA
 - R_{rj} Load on the containment generated by the jet impingement from a ruptured high-energy pipe during the postulated event of the DBA
 - R_{rm} load on the containment resulting from the impact of a ruptured high-energy pipe during the DBA
 - Additional blast loads that may result from a postulated instantaneous break of a large pipe that could occur prior to the jet loads and that do need to be combined with the other loads
- Internal flooding loads resulting from a DBA
- Hard objects drop impact loadings, as applicable

Loads associated with DEC representing a subset of beyond design basis accident conditions are discussed in Subsection 3.5.6.

3.5.3.2.2 Design Load Combinations for the SCCV

The SCCV portion of the BWRX-300 containment is designed for load combinations and associated load factors for applicable loading conditions in accordance with NEDC-33926P, supplemented by U.S. NRC RG 1.136.

3.5.3.2.3 Design Load Combinations for the Containment Closure Head and Other Class MC Components

Load combinations and associated load factors used in the design of the containment closure head and other Class MC components are in compliance with U.S. NRC RG 1.57 (Reference 3.5-26) and U.S. NUREG-0800, SRP 3.8.2.

The portion of the BWRX-300 containment closure head and other Class MC components backed by concrete are designed for the load combinations and associated load factors in accordance with NEDC-33926P, supplemented by US NRC RG 1.136.

3.5.3.3 Design and Analysis Procedures

3.5.3.3.1 Containment Structural Analysis Procedures

As mentioned in Subsection 3.5.1.1, the BWRX-300 RB, including the containment, the containment internal structures and their common foundation, are analyzed as one integrated structure.

The connections between the SCCV and the RB members in the integrated FE model are modeled to reflect the appropriate load transfer for gravity, lateral and thermal loads.

Analysis procedures for the integrated structure are discussed in Subsection 3.5.1.1.

3.5.3.3.2 Structural Design Method for SCCV

The design of the SCCV structure conforms to the requirements of NEDC-33926P and meets the acceptance criteria discussed in Subsection 3.5.3.4.

Membrane forces, shear forces and bending moments used in the design of SCCV sections are obtained from the linear elastic computer analyses for the integrated RB and SCCV FE model discussed in Subsection 3.5.1.1. Subsection 3.5.5.3.2 provides further details for the critical section identification and design.

3.5.3.3.3 *Structural Design Methods for Containment Closure Head and Other Class MC Components*

The design procedures for the containment closure head and other Class MC components are as shown in Figure 3.5-5 and Figure 3.5-6, respectively.

The BWRX-300 containment closure head and other Class MC components are designed in accordance with ASME BPVC, Section III, Division 1, Subsection NE, Subarticles NE-3100 (General Design), NE-3200 (Design by Analysis), and NE-3300 (Design by Formula), as applicable. The design meets the acceptance criteria discussed in Subsection 3.5.3.4, including buckling and fatigue evaluations as required. The design by analysis utilizes the demands from the analyses of appropriate finite element models as described in Subsection 3.5.1.1. The stresses, including discontinuity stresses induced by the combination of applicable loads during different plant conditions, are evaluated, as applicable.

The access hatch cover with the bolted flange is designed in accordance with Subarticle NE-3326 of ASME BPVC, Section III, Division 1, Subsection NE.

3.5.3.4 *Structural Acceptance Criteria*

3.5.3.4.1 *Design Basis Acceptance Criteria for SCCV*

The acceptance criteria for the design of the SCCV are in accordance with NEDC-33926P. The allowable stresses and strains in NEDC-33926P, for service and factored loads used in the design of the SCCV are provided in Table 3.5-4.

3.5.3.4.2 *Design Basis Acceptance Criteria for Containment closure Head and Other Class MC Components*

The acceptance criteria for the design basis loads of the steel containment closure head and other MC components are the allowable stress limits specified in ASME BPVC, Section III, Division 1, Subsection NE-3220. The structural acceptance criteria for the Post-flooding condition, which is only applicable for other Class MC components excluding the containment closure head, is in accordance with U.S. NUREG-0800, SRP 3.8.2. Table 3.5-5 and Table 3.5-6 summarize the acceptance criteria for testing, design, Level A, C and D, and Post-flooding conditions, as applicable, for the containment closure head and other Class MC components, respectively. Stability against compression buckling is assured by an adequate factor of safety.

3.5.3.4.3 *Containment Seismic Design Criteria*

The Seismic design criteria for the BWRX-300 containment are summarized in Table 3.3-1.

The seismic design of the BWRX-300 containment considers LS-D response in accordance with ASCE/SEI 43, ensuring an essentially elastic response without any significant permanent deformation when subjected to DBE, and complying with the regulatory requirements in CNSC REGDOC-2.5.2, Section 8.6.2.

Per CSA N289.3, Clause 7.5, the seismic design of the:

- SCCV is in accordance with NEDC-33926P
- Steel components at the containment boundary not backed by SCCV is in accordance with provisions of ASME BPVC, Section III, Division 1, Subsection NE

Also, in compliance with CNSC REGDOC-2.5.2, Section 8.6.2, the BWRX-300 containment meets the deformation acceptance criteria of ASCE/SEI 43, Section 5.2.3 and possesses ductility and energy absorbing capacity which permits inelastic deformation without failure under DEC's.

3.5.3.4.4 Containment Design Criteria for Impulsive and Impactive Loads

The BWRX-300 containment is designed for impulsive and impactive loads in compliance with requirements of Sections 7.15.1 and 7.15.3 of CNSC REGDOC-2.5.2 and the regulatory guidelines of U.S. NUREG-0800, SRP 3.8.1, Appendix A.

The design of the SCCV for impulsive and impactive loads follows the applicable requirements of the SCCV NEDC-33926P.

The design of the steel components of the containment not backed by SCCV follows the relevant regulatory guidance of U.S. NRC RG 1.57 and provisions of ASME BPVC, Section III, Division 1, Subsection NE.

3.5.3.4.5 Containment Robustness Acceptance Criteria

Complying with CNSC REGDOC-2.5.2, Section 6.1, the Level Four D-in-D described in Subsection 3.1.6 requires that the containment design be robust to provide adequate protection for the confinement function, including the use of complementary design features to prevent accident progression and to mitigate the consequences of DEC and BDBAs. Refer to Subsection 3.5.6.1 for a detailed discussion of the robustness design and acceptance criteria for the BWRX-300 containment. These acceptance criteria satisfy the requirements in CNSC REGDOC-2.5.2, Sections 7.22.3 and 8.6.12, ensuring there is sufficient structural integrity to protect important systems in event of a design basis threat.

The leak tightness at the boundary of the containment structure, including the SCCV, containment closure head, and other Class MC components, under DEC internal pressure loads meets the requirements of CNSC REGDOC-2.5.2 and U.S. NRC RG 1.216 (Reference 3.5-27).

3.5.3.5 Materials, Quality Control and Special Construction Techniques

3.5.3.5.1 Containment Materials

Materials used in the construction of the SCCV portion of the containment structure are in accordance with NEDC-33926P and U.S. NRC RG 1.136.

Steel materials used in the fabrication of the containment closure head and other Class MC components are in accordance with ASME Section III Subsection NE, Article NE-2000.

Details of materials used in the construction of the containment structures are provided in Chapter 9B, Subsection 9B.2.1.4.

3.5.3.5.2 Containment Quality Control

Quality control procedures are established for the containment structure in the construction, fabrication and installation specifications and implemented during fabrication, construction, installation, and inspection. These specifications cover the fabrication, furnishing, and installation of each structural item and specifies the inspection and documentation requirements to ensure that the requirements of NEDC-33926P, Articles NE-4000 and NE-5000 of ASME Section III, Division 1, Subsection NE, U.S. NRC RG 1.28 (Reference 3.5-28), U.S. NRC RG 1.136, and U.S. NUREG-0800, SRP 3.8.2 are met.

3.5.3.5.3 Containment Special Construction Techniques

The integrated RB, SCCV, RPV pedestal, and other structural components are constructed using modular construction technique as described in Subsection 3.5.5.5.

3.5.3.6 Testing and In-Service Inspection Requirements

Concrete and concrete constituents in the Steel Bricks™ modules of the SCCV are examined and tested in accordance with NEDC-33926P, as supplemented by the concrete sampling

requirements in NEDO-33914 Revision 2. Inspection of Steel Bricks™ welds is in accordance with NEDC-33926P.

3.5.3.6.1 Structural Integrity Test (SIT)/PRE-Operational Proof Test

The SCCV pre-service SIT plan and instrumentation is in compliance with NEDC-33926P and U.S. NRC RG 1.216. The SIT ensures compliance with containment pressure structure capability requirement for pressure tests in CNSC REGDOC-2.5.2, Section 8.6.3.

In accordance with NEDC-33926P, deformation, stress and strain measurements are made to evaluate the behavior of the containment and confirm that the actual structural response is within the limits predicted by analysis.

3.5.3.6.2 Containment Pre-Service and In-Service Inspection

The SCCV pre-service and periodic in-service inspection plan is in accordance with NEDC-33926P to comply with the requirements of CNSC REGDOC-2.5.2.

3.5.3.6.3 Integrated Leak Rate Testing

The SCCV is designed such that the periodic ILRT can be conducted at the design pressure to demonstrate the leak tightness integrity of the containment boundary in compliance with Section 8.6.4 of CNSC REGDOC-2.5.2. The ILRT is performed per criteria outlined in Chapter 6, Subsection 6.3.7.

The flange seals of the containment closure head and Class MC components that have potential for significant contribution to leakage are designed to be individually testable. Where resilient seals such as elastomeric seals are used, they have the capability for performing leak testing at the containment design pressure in compliance with Section 8.6.5 of CNSC REGDOC-2.5.2.

3.5.4 Containment Internal Structures

The BWRX-300 containment internal structures comprise the Steel Bricks™ RPV pedestal, the steel-plate composite bioshield surrounding the RPV pedestal and structural steel Containment Equipment and Piping Support Structure (CEPSS), including the support floor at Level -8.5 m, and support floors at Level -21 m and -29 m.

3.5.4.1 Applicable Codes, Standards and Other Specifications

Codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and in-service inspection of the BWRX-300 containment internal structures are listed in Chapter 1, Appendix B.

Similar to RB, the analysis and design, fabrication and testing of the containment internal structures is in accordance with the ANSI/AISC N690, including the supplemental requirements in U.S. NRC RG 1.243 and NEDC-33926P. This methodology ensures a level of safety and performance for the containment internal structures commensurate to that required by CSA N291 and ensures compliance with CNSC REGDOC-2.5.2.

Refer to Figure 3.5-1 for the jurisdictional boundary for the RPV pedestal, the bioshield and internal structural steel.

3.5.4.2 Loads and Load Combinations

Since the containment internal structures are completely contained within and are integrated with the RB and SCCV, the design of containment internal structures considers both design loads applied directly to the containment internal structures and those applied indirectly through the RB and SCCV.

3.5.4.2.1 Design Loads

Refer to Subsections 3.5.3.2 and 3.5.5.2 for the description of design loads applicable for the SCCV and RB structures that are also generally applicable for the design of containment internal structures. Since containment internal structures are inside the containment, some of the design loads applicable for the RB are not directly applicable for the containment internal structures. Additionally, the internal flooding condition associated with post-accident flooding is not considered in accordance with U.S. NUREG-0800, SRP 3.8.1 as noted in Table 9B-1 in Chapter 9B.

The design loads also include the reactions from the RPV at the support locations on the containment internal structures and other bracket and attachment loads applicable during different plant conditions. The RPV lumped mass beam model representing the mass and stiffness properties of the RPV is included in the integrated FE model discussed in Subsection 3.3.1.2, and the dead load and seismic load reactions from the RPV are obtained directly from the static and seismic analyses. Other normal and accidental plant operating loads are applied to the model as reaction force loads.

3.5.4.2.2 Design Load Combinations

Load combinations and load factors for the design of the Steel Bricks™ structures and structural steel that form the containment internal structures are in accordance with ANSI/AISC N690, including the supplemental regulatory guidance of U.S. NRC RG 1.243.

3.5.4.3 Design and Analysis Procedures

3.5.4.3.1 Structural Analysis Procedures

Analysis procedures for the containment internal structures are the same as those for the integrated RB structure discussed in Subsection 3.5.1.1 since containment internal structures are included in the integrated FE model used in the analyses.

The connections between the containment internal steel structures and the RPV, RPV pedestal, bioshield and SCCV are appropriately modeled in the integrated FE model to reflect the appropriate load transfer for gravity and lateral loads.

Local models may be used, if needed, for detailed design at opening and connection locations.

3.5.4.3.2 Structural Design Methods

For the design of containment internal structures, the design methodology is the same as that used for the design of the RB structure, discussed in Subsection 3.5.5.3.

3.5.4.4 Structural Acceptance Criteria

3.5.4.4.1 Design Basis Acceptance Criteria

The design basis acceptance criteria of the containment internal structures, including the Steel Bricks™ RPV pedestal, the steel-plate composite bioshield and containment internal steel structures, are same as those for the corresponding RB structural components described in Subsection 3.5.5.4.

3.5.4.4.2 Robustness Acceptance Criteria

The methodology and acceptance criteria for the robustness of the containment internal structures are described in Subsection 3.5.6.1.

3.5.4.5 Materials, Quality Control and Special Construction Techniques

3.5.4.5.1 Materials

The concrete and structural steel materials used for the construction of containment internal structures are same as those for the RB structure as described in Subsection 3.5.5.5, except that pool liners are not applicable.

3.5.4.5.2 Quality Control

The quality control requirements for containment internal structures are same as those for the RB structure as described in Subsection 3.5.5.5.

3.5.4.5.3 Special Construction Techniques

The integrated RB, SCCV, RPV pedestal, and other structural components are constructed using modular construction technique as described in Subsection 3.5.5.5.

3.5.4.6 Testing and In-Service Inspection Requirements

A formal program of testing and in-service inspection is not required for containment internal structures since they are not directly related to the functioning of the containment system. However, during the operating life of the plant, the condition of the containment internal structures is monitored per 10 CFR 50.65 in accordance with U.S. NRC RG 1.160 (Reference 3.5-29).

3.5.5 Reactor Building

3.5.5.1 Applicable Codes, Standards and Other Specifications

Codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and in-service inspection of the BWRX-300 RB are listed in Chapter 1, Appendix B.

Specifically, the analysis and design, fabrication and testing of the RB structure (including the Steel Bricks™ walls, slabs and mat foundation and the structural steel components, see Figure 3.5-1) is in accordance with the ANSI/AISC N690, including the supplemental requirements in U.S. NRC RG 1.243 and NEDC-33926P. This methodology ensures a level of safety and performance for the RB commensurate to that required by CSA N291 and ensures compliance with CNSC REGDOC-2.5.2.

The RB polar crane is designed and constructed to meet the requirements of ASME NOG-1 (Reference 3.5-30).

Crane loading is developed in accordance with NBC and ASCE/SEI 7 (Reference 3.5-31), Section 4.9.

3.5.5.2 Loads and Load Combinations

In addition to the loads applicable directly to the RB, loads considered in the design of the RB include loads applied to the SCCV that have an effect on the RB structure due to the common mat foundation, floor slabs, RB shear walls and other integrating structural components.

3.5.5.2.1 Design Loads

The RB structure is analyzed and designed in accordance with ANSI/AISC N690 for design basis load cases in compliance with CSA N291.

Loads, such as accident pressure and thermal transient loads due to a LOCA, internal to SCCV are considered for the design of structural components of the RB that are integrated with the SCCV.

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RB design loads consist of:

- Service category of loads that occur during construction, pre-operational testing, or normal operation. They include:
 - Dead loads (D) which consist of the weight of structures, weight of permanently attached major equipment, tanks, machinery, and cranes; weight of piping, cable, cable trays, duct supports; and hydrostatic pressure of liquids in various pools
 - Live loads (L , L_r) which consist of floor area loads, laydown loads, nuclear fuel, and equipment handling loads
 - Lateral Soil and groundwater pressure loads (H)
 - Snow/rain loads (S/R) discussed in Subsection 3.3.2
 - Normal plant operation and pre-operation pressure testing loads which consist of operation service pressure loads, pre-operation proof test pressure load, normal thermal conditions (T_o) and operation service pipe reaction loads (R_o)
 - Construction Loads
 - Settlement Loads
 - Crane Loads developed as discussed in Subsection 3.5.5.1.
- Abnormal and environmental category of loads that occur during postulated accident and/or severe or extreme environmental events. They include:
 - Abnormal plant operation loads which include accident pressure (P_a) and thermal (T_a) loads, accident pipe reaction loads (R_a), missile generation, pipe whip (Y_r), jet impingement from large pipe breaks (Y_j), blast pressure (Y_m), compartment pressurization and drop of large loads
 - Wind and Tornado loads (W , W_t) discussed in Subsection 3.3.2
 - Seismic loads (E_s) discussed in Subsection 3.3.1, including hydrodynamic loads on the pool walls calculated based on the approach described in ASCE/SEI 4 (Reference 3.5-32) and ACI 350.3 (Reference 3.5-33), and dynamic incremental soil pressures
- Hard objects drop impact loadings, as applicable
- Design Basis Threat loads discussed in Subsection 3.3.7.4

Loads associated with DEC representing a subset of beyond design basis accident conditions are discussed in Subsection 3.5.6.

3.5.5.2.2 Design Load Combinations

Load combinations and load factors for the design of the Steel Bricks™ module structures and structural steel in the RB are in accordance with the provisions of ANSI/AISC N690, Chapter NB2.6 including the supplemental regulatory guidance of U.S. NRC RG 1.243, Regulatory Positions 2.1 and 2.2.

3.5.5.3 Design and Analysis Procedures

3.5.5.3.1 Structural Analysis Procedures

Refer to Subsection 3.5.1.1 for analysis procedures.

3.5.5.3.2 Structural Design Methods

The design of the RB structure conforms to the requirements of ANSI/AISC N690, including the regulatory guidance in U.S. NRC RG 1.243 and meets the acceptance criteria discussed in Subsection 3.5.5.4 to ensure a level of safety and performance commensurate with the requirements in CSA N291.

Membrane forces, shear forces and bending moments used in the design of the RB Steel Bricks™ and steel sections are obtained from the linear elastic computer analyses for the integrated RB FE model discussed in Subsection 3.5.1.1.

Results from the FE analyses are evaluated to identify critical cross-sections where maximum structural demands occur for different controlling loads and load combinations. Key responses reviewed include:

- Membrane forces for the SCCV,
- In-plane shear demands at the base of major walls and at rock-soil interface elevation,
- Vertical bending moments and out-of-plane shear demands on the RB outer shaft and SCCV walls, at base of walls and at intermediate floor elevations and
- Out-of-plane demands for major floor slabs and RB foundation mat at mid-span and support locations.

The structural demands at the critical locations are used to perform the design of the critical cross-sections and connections using the applicable codes of record.

3.5.5.4 Structural Acceptance Criteria

3.5.5.4.1 Design Basis Acceptance Criteria

The RB Steel Bricks™ module structures and structural steel, including welded and bolted connections, are designed to meet the acceptance criteria outlined in ANSI/AISC N690.

The RB structure is evaluated for serviceability considerations including deflection, vibration, permanent deformation, cracking, and settlement. Serviceability evaluations meet the acceptance criteria in ANSI/AISC N690, Chapter NL.

Seismic Design Criteria

The Seismic design criteria for the BWRX-300 RB are summarized in Table 3.3-1.

The seismic design of the RB structure considers LS-D response in accordance with ASCE/SEI 43, ensuring an essentially elastic response without any significant permanent deformations when subjected to DBE and complying with the regulatory requirements in CNSC REGDOC-2.5.2, Section 8.6.2.

The BWRX-300 RB structure meets the deformation acceptance criteria of ASCE/SEI 43, Section 5.2.3 and possesses ductility and energy absorbing capacity which permits inelastic deformations without failure under DEC.

Evaluation Criteria for Structure Interaction Under Seismic and Extreme Wind

The interaction of the RB structure with the adjacent RWB, CB, TB and Reactor Auxiliary Bay is discussed in Subsections 3.3.1.2 and 3.3.2.8.

The stability of foundations under DBE and design basis tornado wind loads are checked following the criteria in Subsection 3.5.2.2.

RB Design for Impulsive and Impactive Loads

The RB structure is designed for impulsive and impactive loads per the requirements of Sections 7.15.1 and 7.15.3 of CNSC REGDOC-2.5.2 and the regulatory guidelines of U.S. NUREG-0800, SRP 3.8.4.

The RB design for impulsive and impactive loads follows the provisions of ANSI/AISC N690 and the relevant regulatory guidance of U.S. NRC RG 1.243.

Criteria used to define the heavy loads considered in the RB design are described in Subsection 3.4.5.1.

3.5.5.4.2 Robustness Acceptance Criteria for RB Structure

Refer to Subsection 3.5.6.1 for a detailed discussion of the robustness design and acceptance criteria for the BWRX-300 RB structure, which satisfy the requirements in CNSC REGDOC-2.5.2, Section 7.22.3.

3.5.5.5 Materials, Quality Control and Special Construction Techniques

3.5.5.5.1 Materials

Materials used in construction of the RB structure outside of the containment are in accordance with ANSI/AISC N690, Section NA3.

Details of materials used in the construction of the RB are provided in Chapter 9B, Subsection 9B.2.3.4.

3.5.5.5.2 Quality Control

Quality control procedures are established and implemented during the construction and inspection phases of the RB structure. These procedures cover the fabrication, furnishing, and installation of each structural item in the RB and specify the inspection and documentation requirements in accordance with the requirements in ANSI/AISC N690, Section NA5, Chapter NN with supplemental guidance provided in U.S. NRC RG 1.243.

3.5.5.5.3 Special Construction Techniques

The BWRX-300 Seismic Category A structures at the DNNP site are built using a modular construction technique using Steel Bricks™. (see Section 3.5.1).

The quality control procedures used in the structural modularization process implemented in the construction of the Steel Bricks are outlined in Subsection 3.5.5.5.2. These procedures are employed at the fabrication shop and the construction-site (both outside and inside the deep excavation pit necessary for the construction of RB), including pre-fabrication and pre-assembly, to ensure the Steel Bricks™ modular assemblies meet the necessary material quality, fabrication, and installation requirements per the applicable code of records.

For the preferred method of construction for the deeply embedded BWRX-300 RB shaft, refer to Section 1.4 of NEDO-33914 Revision 2.

For plant construction and commissioning activities, refer to Chapter 14.

3.5.5.6 Testing and In-Service Inspection Requirements

Per CNSC REGDOC-2.5.2, Section 7.15.2, periodic inspection, and in-service monitoring programs are implemented to ensure the RB structure continues to meet its functional and performance requirements.

Sections 3.2 through 3.4 of NEDO-33914 Revision 2 describe the approaches and guidelines for the BWRX-300 in-service testing, monitoring, and monitoring programs.

NEDC-33926P describes the in-service inspection and testing guidelines for the Steel Bricks™ to ensure that the integrated RB structures satisfy their functional and performance requirements through all phases of the plant's life cycle. The BWRX-300 implements a Structures Monitoring and Aging Management Program (SMAMP) that monitors the condition of structures and manages aging effects in accordance with CSA N291, clauses 9 and 10 and in compliance with CNSC REGDOC-2.5.2, Section 7.17. The program demonstrates that the facility is constructed to the requirements in the design drawings and specifications. A research and development program is also established to demonstrate the adequacy of Steel Bricks™ to maintain the structural integrity of the integrated RB structures and of inspection methods used in compliance with CNSC REGDOC-2.5.2, Section 5.4.

3.5.6 Robustness Design of Seismic Category A Structures

Consistent with the Level Four D-in-D requirements discussed in Subsection 3.1.6 and in Section 6.1 of CNSC REGDOC-2.5.2, the BWRX-300 containment and RB are robust structures, tolerant of a large spectrum of faults with a gradual degradation in their effectiveness, that would not fail catastrophically under operational states, DBAs and DEC's.

Evaluations performed to establish an understanding of safety margins, or the robustness of the design are consistent with the regulatory guidance of CNSC REGDOC-2.4.1 (Reference 3.5-34), Section 4.2.3 and U.S. NUREG-0800, SRP 19.0.

3.5.6.1 Design Extension Conditions

In accordance with Section 7.15.1 of CNSC REGDOC-2.5.2, DEC's considered in the design of the BWRX-300 Seismic Category A structures include severe accident conditions due to both internal and external hazards, whose probability of occurrence is lower than the probability of occurrence of the DBA.

Loads, load combinations, strength and safety requirements for assessing the BWRX-300 Seismic Category A structures (i.e., the integrated RB) are defined in accordance with Clause 6.1.4 of CSA N291.

Consistent with Section 7.3.4 of CNSC REGDOC-2.5.2 and Clause 5.6 of CSA N290.16 (Reference 3.5-35), deterministic safety analyses are used to determine the applicable DEC's and evaluate the consequences of the DEC's.

In accordance with the guidelines of CSA N290.16, Clause 4.3.5, a best estimate approach is used to obtain a reasonable confidence in the assessed response to DEC's.

A reasonable level of survivability of the structure under postulated DEC's is demonstrated following requirements of Clause 6.1.3.1 of CSA N290.16. Per Clause 4.5 of CSA N290.16, less stringent assumptions than those applied for design basis, such as the permissible variances in Annex C of CSA N290.16, may be used when evaluating SSC performance under DEC's.

3.5.6.1.1 Containment Severe Design Extension Condition Evaluations

Complying with Section 8.6.12 of CNSC REGDOC-2.5.2, the BWRX-300 containment design ensures the ability of the containment system to withstand loads associated with DEC's.

Consistent with CNSC REGDOC-2.5.2, Section 8.6.2, the containment structure is designed to possess ductility and energy absorbing capacity, which permits inelastic deformation without failure under DEC's.

The beyond design basis evaluations of the containment ensure the structural integrity and leak tightness of the containment structure under all applicable DEC loading cases in compliance with the regulatory guidance of CNSC REGDOC-2.5.2.

Containment Ultimate Pressure Capacity

The ultimate internal pressure capacity of the containment structure, including the SCCV, containment closure head and penetrations, is determined to ensure its structural integrity and leak tightness under DEC internal pressure loads to meet the requirements in CNSC REGDOC-2.5.2, Section 7.15.1, U.S. NRC RG 1.216, and U.S. NUREG-0800, SRP 3.8.1.

This ultimate pressure capacity is obtained from the results of non-linear finite element analysis consistent with the guidelines of Regulatory Position 1 of U.S. NRC RG 1.216.

Robustness Against Combustible Gas Pressure Loads

The BWRX-300 design demonstrates the ability of the containment to withstand DEC loads associated with combustion of gases consistent with requirements of Section 8.6.12 of CNSC REGDOC-2.5.2.

The containment is designed to ensure that its structural integrity is maintained to sustain the combustible gas pressure loads applicable for BWRX-300 consistent with the requirements in U.S. NRC RG 1.136 and U.S. NRC RG 1.57.

Containment Severe Accident Performance Goal

Consistent with guidance in CNSC REGDOC-2.5.2, Section 8.6.12, the BWRX-300 design is a fail-safe design that ensures that under DEC conditions with core damage, the containment:

- A. Maintains its role as a reliable leak-tight barrier for a minimum of 24 hrs following the onset of core damage
- B. Continues to provide a barrier against the uncontrolled release of fission products following the initial 24 hrs period

The methodology used to evaluate the robustness of the containment is per Regulatory Position 3 of U.S. NRC RG 1.216. The evaluation identifies pressure and temperature loadings associated with the more likely DEC challenges by considering the sequences of plant damage states that represent 90% or more of the core damage frequency. Analyses of global and local finite element models are performed to calculate the enveloping containment response for the identified accident challenges.

Criteria for factored load category in NEDC-33926P for the SCCV is used to demonstrate the containment deterministic performance goal for the initial 24 hours. The deterministic performance goal after the initial 24-hour period is demonstrated by showing that the containment leakage in a severe accident remains below the design leakage rate limit, consistent with CNSC REGDOC-2.5.2, Sections 8.6.4 and 8.6.12, for sufficient time to allow implementation of emergency measures.

During an extremely improbable severe accident in the BWRX-300, molten core debris may be present on the containment floor. A protective layer of refractory concrete prevents corium (as shown in Chapter 9B, Figure 9B-1) from degrading the SCCV inner steel faceplate that acts as the primary leak-tight boundary. Additional protection is provided by the outer steel faceplate for the SCCV foundation mat. The lower SCCV design has a provision for the installation of a severe accident core melt capture and retention structure with a spreadable area to prevent contact between the molten core and the containment liner and concrete. Refer to Chapter 15, Appendix 15B for more details on this corium shield and other complementary design features for BDBAs.

3.5.6.1.2 Beyond Design Basis Seismic Robustness

In accordance with CNSC REGDOC-2.5.2, Section 7.13.1, the design of the BWRX-300 Seismic Category A and Seismic Category B SSC credited to function during and after a Beyond-Design

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Basis Earthquake (BDBE) ensures their capability to maintain their structural integrity and to perform their intended safety function.

The BDBE is defined to meet the DEC identification requirements of CNSC REGDOC-2.5.2, Section 7.3.4. Per CNSC REGDOC-2.5.2, Section 7.13.1, a High Confidence ($\geq 95\%$) of Low Probability ($\leq 5\%$) of Failure (HCLPF) of at least 1.67 times that for the DBE is demonstrated for the SSC credited to function during and after a BDBE.

The methodology in Electrical Power Research Institute (EPRI) TR-103959 (Reference 3.5-36), TR-1002988 (Reference 3.5-37) and TR-1019200 (Reference 3.5-38), consistent with the recommendations of TR- 3002012994 (Reference 3.5-39) is used for the evaluations of seismic fragilities of BWRX-300 Seismic Category A and B SSC.

Following the regulatory guidance of CNSC REGDOC-2.5.2, Section 7.13.1, to ensure adequate margins for the BDBE, the seismic design satisfies the ductility detailing and design requirements for steel and steel-plate composite structures of ANSI/AISC N690, with the supplementary guidance of U.S. NRC RG 1.243 and NEDC-33926P. This approach meets the intent of CSA S16 (Reference 3.5-40), for Seismic Category A steel structures members and connections.

Checking Level Earthquake

Per Clause 5.4.5 of CSA N289.1 (Reference 3.5-41), a Checking Level Earthquake (CLE) defines the earthquake level for BDBE evaluations to ensure prescribed safety margins for earthquakes exceeding the DBE.

The BWRX-300 plant is assessed during the design process, in accordance with Clause 8.2 of CSA N289.3, using CLE to:

- Provide detailing for post-elastic behavior and energy absorption during BDBE events
- Identify any SSC that can have insufficient seismic ruggedness, ductility, or inelastic response capability to withstand and perform their safety function during and after BDBE
- To ensure no cliff-edge effects

The site-specific CLE ground motion spectra are defined as 1.5 times the DBE, which is at a level sufficiently larger than the DBE to support meeting the acceptable plant HCLPF criteria of CNSC REGDOC-2.5.2, Section 7.13.1. The site-specific CLE is representative of a seismic hazard exceedance probability that is lower than the seismic hazard probability of the DBE and meets the requirements of Clause C.3.3 of CSA N289.1.

The selected CLE maintains consistency with the performance objectives expressed in Chapter 1 of ASCE/SEI 43 and the precedence set for definition of BDBE motion in Chapter 9 of ASCE/SEI 43. The performance objectives in ASCE/SEI 43 aim to achieve 10% unacceptable performance for 150% of DBE level per U.S. NRC RG 1.208 (Reference 3.5-42). It is recognized that the redundancy in the SSC credited to function during and after a CLE is included in the calculation of a plant level HCLPF of at least 1.67 times the DBE.

CLE in-structure demands for BDBE evaluations are obtained from BE approach seismic response analyses performed following the guidance of CSA N289.1, Clause C.4.2, consistent with the criteria in Subsection 3.3.1.3. The SSI input soil profiles for the BDBE evaluations are obtained at strain levels consistent with the CLE motion. The SSI analyses for BDBE evaluations may use Response Level 3 damping values in accordance with ASCE/SEI 43

In accordance with Section 5.2.7 of CSA N289.1, CLE is considered in combination only with normal operating loads.

3.5.6.2 Design for Malevolent Acts

The BWRX-300 uses a security by design process that involves security reviews during plant design to resolve DBT and BDBT security issues at the earliest stage, when changes have the least impact on cost and performance. Placement and number of doors, wall thicknesses to optimize resistance to explosive breaching, and equipment placement to facilitate better target set diversity are all achievable when security is integrated at an early stage. Continual design reviews against the DBT and BDBT capabilities during the entire design evolution ensure that emergent issues are identified and addressed as early in the process as possible.

The defensive strategy approach focuses on protecting the passive plant features and other key reactor components from hostile action by creating a robust perimeter. By analyzing the potential adversary pathways to critical components, determining adversary resources required to execute the path, and slowing the adversary movements and depleting the adversaries' resources before the path can be completed to the extent possible, the design limits the ability of malicious individuals to cause damage to key systems. This, along with the inherent slower accident progression of the BWRX-300 reactor, reduces or eliminates the reliance on immediate on-site armed responders to prevent substantial off-site radiological releases, which allows for longer term off-site response, interdiction, and neutralization.

Malevolent Acts Design Methods

The BWRX-300 design for DBTs and BDBTs satisfies the requirements of CNSC REGDOC-2.5.2, Section 7.22.2.

The design considers the following two types of structural failure modes with distinct loading characteristics and structural responses:

1. Local effects that in general would not result in structural collapse but may affect the functions of safety class SSC
2. Global failure modes characterized by major structural damage, such as significant perforation or collapse of large portions of the building walls, floors, and load carrying frames

These failure modes are considered separately with a consideration given that for some threats, such as an aircraft crash, they may act simultaneously or quasi-simultaneously.

Applicable local damage modes are considered in the design and empirical formulas are used to assess the structural behavior under local and concentrated loading.

The BWRX-300 design applies the Nuclear Energy Institute's methodology in NEI 07-13 (Reference 3.5-43) for aircraft crash evaluations with CNSC input and other detailed computer analytical methods, where appropriate, to evaluate the consequences of regulatory defined threats on a BWRX-300 reactor site. The CNSC acceptance criteria are then applied to the results.

Evaluations include:

- RB structural integrity including enclosed safety features as applicable:
 - Global failure (plastic collapse)
 - Local perforation (hard missile)
 - The acceptance criteria for both local and global behavior are satisfied simultaneously
- Containment and fuel pool heat removal capability
- Reactivity control following regulatory defined threats

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- Containment isolation following regulatory defined threats
- Fuel intrusion prevention
- Shock and vibration impact of critical equipment
- Short and long-term mitigation efforts required following commercial aircraft impact

Malevolent Acts Design Acceptance Criteria

The design of the BWRX-300 Seismic Category A structures meets the following acceptance criteria for local response under malevolent acts depending on the structural system used:

1. For DBTs, no scabbing of the rear face of structural elements, possibly with limited, easily repairable, superficial spalling of concrete
2. For severe BDBTs, no scabbing of the rear face of structural element, or possible limited scabbing if confined by the steel liner that should remain leak-tight
3. For extreme BDBTs, no perforation, according to the applicable formula with a corresponding increase factor of 1.2 applied to the calculated thickness
4. For Steel Bricks™ members, the steel faceplate thickness to prevent perforation is at least 1.25 times that required by use of rational methods in accordance with ANSI/AISC N690 and NEDC-33926P

The structural acceptance criteria for global response are related to:

- The limitation of structural deflections for DBT and severe BDBT; or
- Overall damage for extreme BDBT

Special attention is given to:

- Damage to the containment and internal structures due to extensive deformations of the containment
- Shock damage to fragile components directly attached to the containment wall
- Induced vibration
- Post-event fireball explosions or blast waves
- Structural integrity of the polar crane

The acceptance criteria for local and global structural response are satisfied simultaneously.

Design criteria for the BWRX-300 RB specifies no global failure, no perforation, no spalling, and no fuel intrusion from the regulatory defined threats.

The design of BWRX-300 containment meets the malevolent acts acceptance criteria in NEDC-33926P that is consistent with the regulatory guidance in Table 1 of CNSC REGDOC-2.5.2, Appendix A.

The BWRX-300 Security Annex describes design methods and acceptance criteria for malevolent acts in greater details.

3.5.7 References

- 3.5-1 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."

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- 3.5-2 CNSC Regulatory Document REGDOC-1.1.5, "Reactor Facilities: Supplemental Information for Small Modular Reactor Proponents."
- 3.5-3 CNSC Regulatory Document REGDOC-3.5.3, "CNSC Processes and Practices, Regulatory Framework."
- 3.5-4 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.5-5 NEDC-33926P, "BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.5-6 ASME BPVC-III NE-2021, "BPVC Section III - Rules for Construction of Nuclear Facility Components-Division 1 - Subsection NE – Class MC Components," American Society of Mechanical Engineers.
- 3.5-7 ANSI/AISC N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities," American Institute of Steel Construction.
- 3.5-8 USNRC Regulatory Guide 1.243, "Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments."
- 3.5-9 CSA N291, "Requirements for Safety-Related Structures for Nuclear Power Plants," CSA Group.
- 3.5-10 NEDO-33914, "BWRX-300 Advanced Civil Construction and Design Approach," GE-Hitachi Nuclear Energy Americas, LLC. (Reference 2.7-35),
- 3.5-11 NK054-REP-01210-00011 R001, "Site Evaluation of The OPG New Nuclear at Darlington - Part 6: Evaluation of Geotechnical Aspects," Ontario Power Generation. 2009 (Reference 2.7-1)
- 3.5-12 NK054-REP-07730-00005 Rev. R000, Geological and Hydrogeological Environment, Existing Environmental Conditions, Technical Support Document, New Nuclear – Darlington Environmental Assessment," Ontario Power Generation. 2009 (Reference 2.7-41)
- 3.5-13 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
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- 3.5-17 EPRI EL-6800, "Manual on Estimating Soil Properties for Foundation Design," Electric Power Research Institute. 1990 (Reference 2.7-39)
- 3.5-18 Carter, M. and Bentley, S., "Soil Properties and their Correlations," John Wiley & Sons, West Sussex, UK, 2016. (Reference 2.7-40)
- 3.5-19 Lo, K.Y., and B. Lukajic, "Predicted and Measured Stresses and Displacements around the Darlington Intake Tunnel," Canadian Geotechnical Journal, 21:147-165. (Reference 2.7-33)

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- 3.5-20 USNRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition."
- 3.5-21 IAEA Safety Standards Series No. NS-G-3.6, "Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants," International Atomic Energy Agency.
- 3.5-22 Canadian Commission on Building and Fire Codes, "National Building Code of Canada," National Resource Council of Canada.
- 3.5-23 ASCE/SEI 43, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," American Society of Civil Engineers.
- 3.5-24 USNRC Regulatory Guide 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments, U.S. Nuclear Regulatory Commission."
- 3.5-25 ASME BPVC-III-2, "Section III: Rules for Construction of Nuclear Facility Components – Division 2- Code for Concrete Containments," American Society of Mechanical Engineers.
- 3.5-26 USNRC Regulatory Guide 1.57, "Design Limits and Load Combinations for Metal Primary Reactor Containment System Components."
- 3.5-27 USNRC Regulatory Guide 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design Basis Pressure."
- 3.5-28 USNRC Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)."
- 3.5-29 USNRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."
- 3.5-30 ASME NOG-1, "Cranes, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," American Society of Mechanical Engineers.
- 3.5-31 ASCE/SEI 7, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," American Society of Civil Engineers.
- 3.5-32 ASCE/SEI 4, "Seismic Analysis of Safety-Related Nuclear Structures," American Society of Civil Engineers.
- 3.5-33 ACI 350.3-20, "Code Requirements for Seismic Analysis and Design of Liquid-Containing Concrete Structures (ACI 350.3-20) and Commentary," American Concrete Institute.
- 3.5-34 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 3.5-35 CSA N290.16, "Requirements for beyond design basis accidents," CSA Group.
- 3.5-36 EPRI TR-103959, "Methodology for Developing Seismic Fragilities," Electric Power Research Institute.
- 3.5-37 EPRI TR-1002988, "Seismic Fragility Application Guide," Electric Power Research Institute.
- 3.5-38 EPRI TR-1019200, "Seismic Fragility Application Guide Update", Electric Power Research Institute.
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- 3.5-42 USNRC Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion."
- 3.5-43 NEI 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Nuclear Energy Institute.
- 3.5-44 ASME BPVC-II D, "Section II: Materials-Part D-Properties (Customary), American Society of Mechanical Engineers.
- 3.5-45 ASME BPVC-III, "Appendix XXVII: Design by Analysis for Service Level D," American Society of Mechanical Engineers.

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Table 3.5-1: As-Built Static Properties for Soil Layers

Layer	Layer Thickness (m)	Total Unit Weight (kN/m ³)	Drained Friction Angle (degrees)		Elastic Modulus (MPa)		At-Rest Lateral Earth Pressure Coefficient	
		Ave.	Ave.	Range	Lower	Upper	Ave.	Range
Fill 1	1.0	22.0	34	29 – 37	15.1	60.8	0.55	0.51 – 0.63
Fill 2	1.0	22.0	34	29 – 37	17.0	77.5	0.55	0.51 – 0.63
Fill 3	1.0	22.0	34	29 – 37	18.8	91.3	0.55	0.51 – 0.63
Fill 4	1.0	22.0	34	29 – 37	20.5	104	0.55	0.51 – 0.63
Fill 5	1.0	22.0	34	29 – 37	22.4	116	0.55	0.51 – 0.63
Fill 6	1.0	22.0	34	29 – 37	24.0	127	0.55	0.51 – 0.63
Fill 7	2.0	22.0	34	29 – 37	25.8	138	0.55	0.51 – 0.63
Upper till	1.1	23.8	37	37	37.0	482	0.32	0.32 – 0.33
Interm. Glacio-lacustrine (Sandy)	7.2	20.9	36	36	36.2	411	0.35	0.34 – 0.35
Interm. Glacio-lacustrine (Silty)	2.8	21.1	30	28 – 32	33.9	379	0.83	0.80 – 0.86
Lower till	4.8	23.5	34	33 – 35	38.1	496	0.78	0.77 – 0.78

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Table 3.5-2: Summary of Static Rock Properties

Layer	Total Unit Weight (kN/m ³)	Intact Rock Deformation Modulus (GPa)	Rock Mass Deformation Modulus (GPa)		Poisson's Ratio
			Average	Range	
Blue Mountain (Whitby)	26.4	31.8	6.4	4.7 – 8.4	0.30/0.58
Lindsay 1	26.6	39.1	13.2	10.4 – 16.1	0.31
Lindsay 2	26.6	35.7	12.1	9.5 – 14.7	0.31
Lindsay 3	26.6	44.4	32.5	28.0 – 36.2	0.31
Verulam 1	26.4	25.7	18.9	16.3 – 21.0	0.33
Verulam 2	26.4	33.1	24.2	20.9 – 27.0	0.31
Verulam 3	26.4	36.3	26.6	22.9 – 29.7	0.31
Verulam 4	26.4	40.3	29.5	25.5 – 32.9	0.31
Bobcaygeon	26.3	44.6	32.7	28.1 – 36.4	0.31
Gull River	26.5	52.8	38.7	33.3 – 43.1	0.32
Shadow Lake	25.7	38.0	27.8	24.0 – 31.0	0.30
Gneiss	27.3	52.6	16.2	11.8 – 21.5	0.28

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Table 3.5-3: Stability Requirements for RB and Containment Common Mat Foundation

Load Combination	Overturning	Sliding	Flotation
$D + H + W$	1.5	1.5	
$D + H + E'$	1.1	1.1	
$D + F'$			1.1
<i>where</i>			
D = Dead Load, W = Wind			
H = Lateral soil pressure, E' = Design Basis Earthquake			
F' = Buoyant forces of design basis flood			

Note:

If quasi-static method using the maximum force effects from the SSI analysis results is used for seismic stability evaluations, the minimum factor of safety against sliding and overturning is no less than 1.25 in accordance with Clause 5.9 of CSA N289.3.

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Table 3.5-4: Acceptance Criteria for SCCV

(a) Allowable Stress/Strain Limits for Factored Loads

Material	Force Classification	Type of Force Action	Criteria for Factored Loads	
			Stress Limit	Strain Limit, if any
Concrete	Primary	Membrane	$0.60f_c'$	-
		Membrane + Bending	$0.75f_c'$	-
	Primary + Secondary	Membrane	$0.75f_c'$	-
		Membrane + Bending	$0.85f_c'$	0.002
Steel Plates	Primary	Membrane or Membrane + Bending	$0.90F_y$	-
	Primary + Secondary	Membrane or Membrane + Bending	-	$2\varepsilon_y^*$

* Limit for mechanical (net) strain, calculated by subtracting strain induced by secondary force from total strain.

(b) Allowable Stresses for Service Loads

Material	Force Classification	Type of Force Action	Criteria for Service Loads
			Stress Limit
Concrete	Primary	Membrane	$0.30f_c'$
		Membrane + Bending	$0.45f_c'$
	Primary + Secondary	Membrane	$0.45f_c'$
		Membrane + Bending	$0.60f_c'$
Steel Plates	Primary	Membrane or Membrane + Bending	$0.50F_y$
	Primary + Secondary	Membrane or Membrane + Bending	$0.67F_y$

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Table 3.5-5: Acceptance Criteria for Containment Closure Head

Service Level	Acceptance Criteria ^{*1}			
	P_m	P_L	$P_L + P_b$ ^{*2}	$P_L + P_b + Q$
Test Condition	$0.8 S_y$	$1.15 S_y$	$1.15 S_y$	N/A ^{*3}
Design Condition	$1.0 S_{mc}$	$1.5 S_{mc}$	$1.5 S_{mc}$	N/A ^{*3}
Level A	$1.0 S_{mc}$	$1.5 S_{mc}$	$1.5 S_{mc}$	$3.0 S_m$
Level C	$1.2 S_{mc}$ or ^{*4} $1.0 S_y$	$1.8 S_{mc}$ or ^{*4} $1.5 S_y$	$1.8 S_{mc}$ or ^{*4} $1.5 S_y$	N/A ^{*3}
Level D	S_f	$1.5 S_f$	$1.5 S_f$	N/A ^{*3}

*1: Acceptance Criteria is defined by ASME BPVC, Subsection NE Subarticles NE-3221.1 through 3221.4.

P_m = primary stress: general membrane.

P_L = primary stress: local membrane.

P_b = primary stress: bending.

Q = secondary stress: membrane plus bending.

S_y = material's yield strength at temperature as in ASME BPVC Section II, Part D (Reference 3.5-44), Table Y-1.

S_m = allowable stress intensity S_m is the value given in ASME BPVC Section II Part D, Subpart 1, Tables 2A and 2B.

S_{mc} = allowable stress intensity S_{mc} is 1.1 times the S listed in ASME BPVC Section II Part D, Subpart 1, Tables 1A and 1B, except S_{mc} shall not exceed 90% of the material's yield strength at temperature shown in ASME BPVC Section II, Part D, Subpart 1, Tables Y-1.

S_f = 85% of the general primary membrane allowable permitted in Mandatory Appendix XXVII, ASME BPVC Code Section III (Reference 3.5-45). In the application of Appendix XXVII, S_m , if applicable, is as specified in NE-3112.4(a)(1).

*2: Values shown are for a rectangular section. See ASME BPVC, Subsection NE, Subarticle NE-3221.3(d) for other than a solid rectangular section.

*3: N/A = Not applicable. No evaluation required.

*4: The larger of the two values listed is chosen as a limit load.

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Table 3.5-6: Acceptance Criteria for Other MC Components

Service Level	Acceptance Criteria ^{*1}			
	P_m	P_L	$P_L + P_b$ ^{*2}	$P_L + P_b + Q$
Test Condition	$0.8 S_y$	$1.15 S_y$	$1.15 S_y$	N/A ^{*3}
Design Condition	$1.0 S_{mc}$	$1.5 S_{mc}$	$1.5 S_{mc}$	N/A ^{*3}
Level A, B	$1.0 S_{mc}$	$1.5 S_{mc}$	$1.5 S_{mc}$	$3.0 S_m$
Level C	$1.2 S_{mc}$ or ^{*4} $1.0 S_y$	$1.8 S_{mc}$ or ^{*4} $1.5 S_y$	$1.8 S_{mc}$ or ^{*4} $1.5 S_y$	N/A ^{*3}
Level D	S_f	$1.5 S_f$	$1.5 S_f$	N/A ^{*3}
Post-flooding Condition	$1.2 S_{mc}$ or ^{*4} $1.0 S_y$	$1.8 S_{mc}$ or ^{*4} $1.5 S_y$	$1.8 S_{mc}$ or ^{*4} $1.5 S_y$	$3.0 S_m$

*1: Acceptance Criteria for other than Post-flooding Condition is defined by ASME BPVC, Subsection NE Subarticles NE-3221.1 through 3221.4. For Post-flooding Condition, Service Level C limits apply to primary stress, and Service Level B limits apply to primary plus secondary stress, per item 5 of SRP Acceptance Criteria in U.S. NUREG-0800 SRP 3.8.2.

*2: Values shown are for a rectangular section. See ASME BPVC, Subsection NE, Subarticle NE-3221.3(d) for other than a solid rectangular section.

*3: N/A = Not applicable. No evaluation required.

*4: The larger of the two values listed is chosen as a limit load

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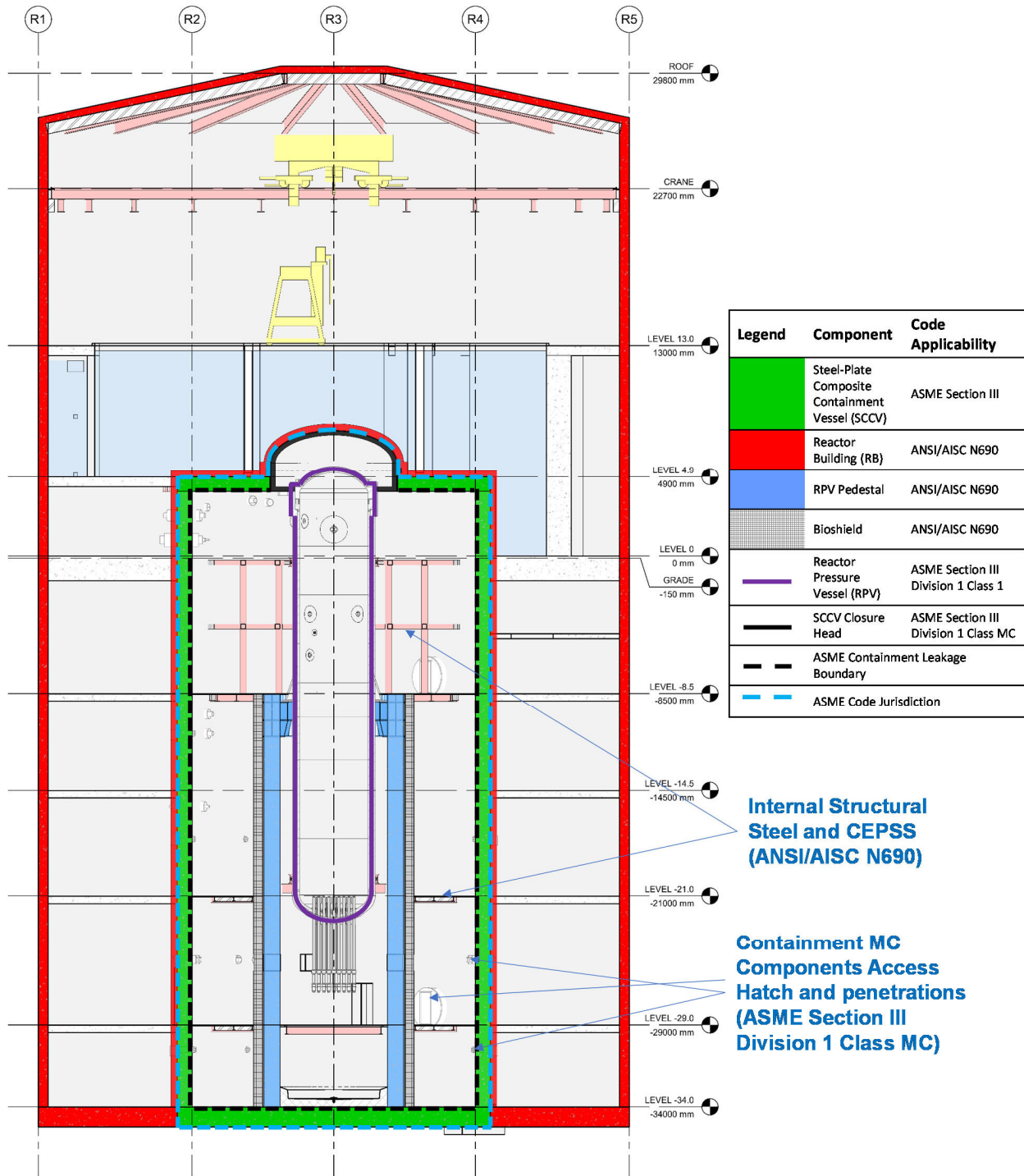


Figure 3.5-1: Structural Boundary of the BWRX-300 Containment, Containment Internal Structures and Reactor Building

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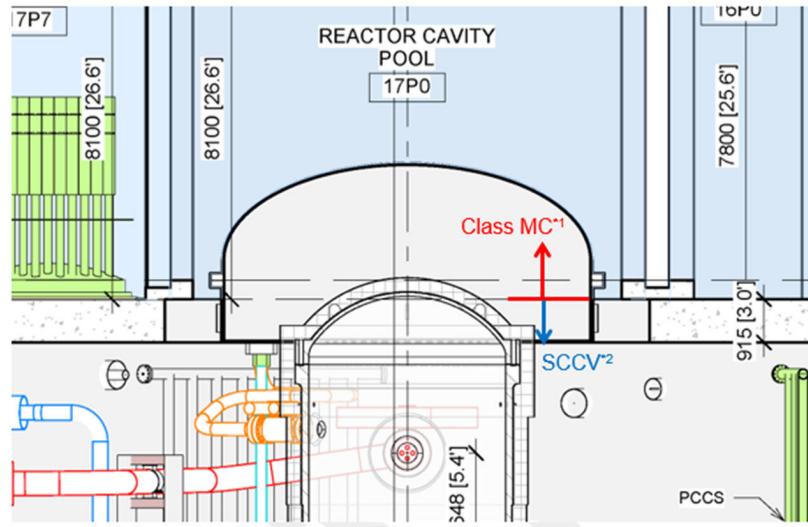


Figure 3.5-2: Containment Closure Head Structure Boundary

*1: Is designed in accordance with ASME Section III Subsection NE (for Class MC)

*2: Is designed in accordance with NEDC-33926P

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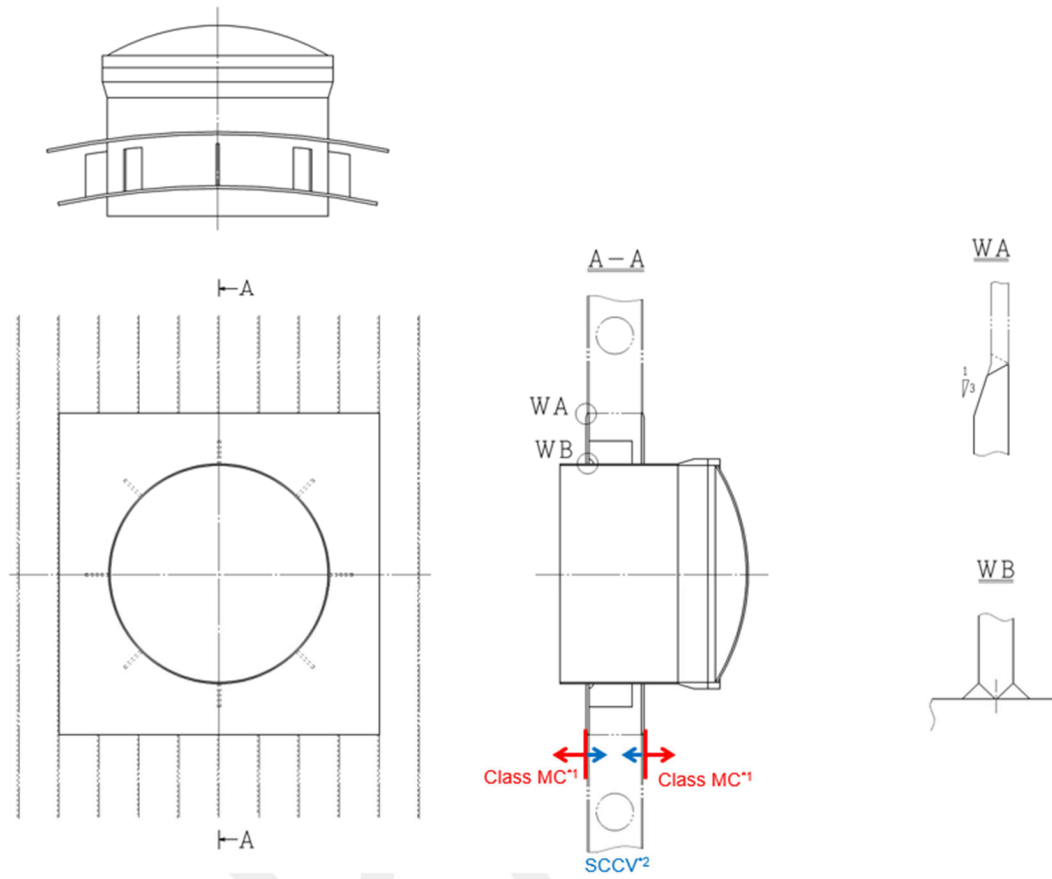


Figure 3.5-3: Access Hatch Code Jurisdictional Boundary

*1: Is designed in accordance with ASME Section III Subsection NE (for Class MC)

*2: Is designed in accordance with NEDC-33926P

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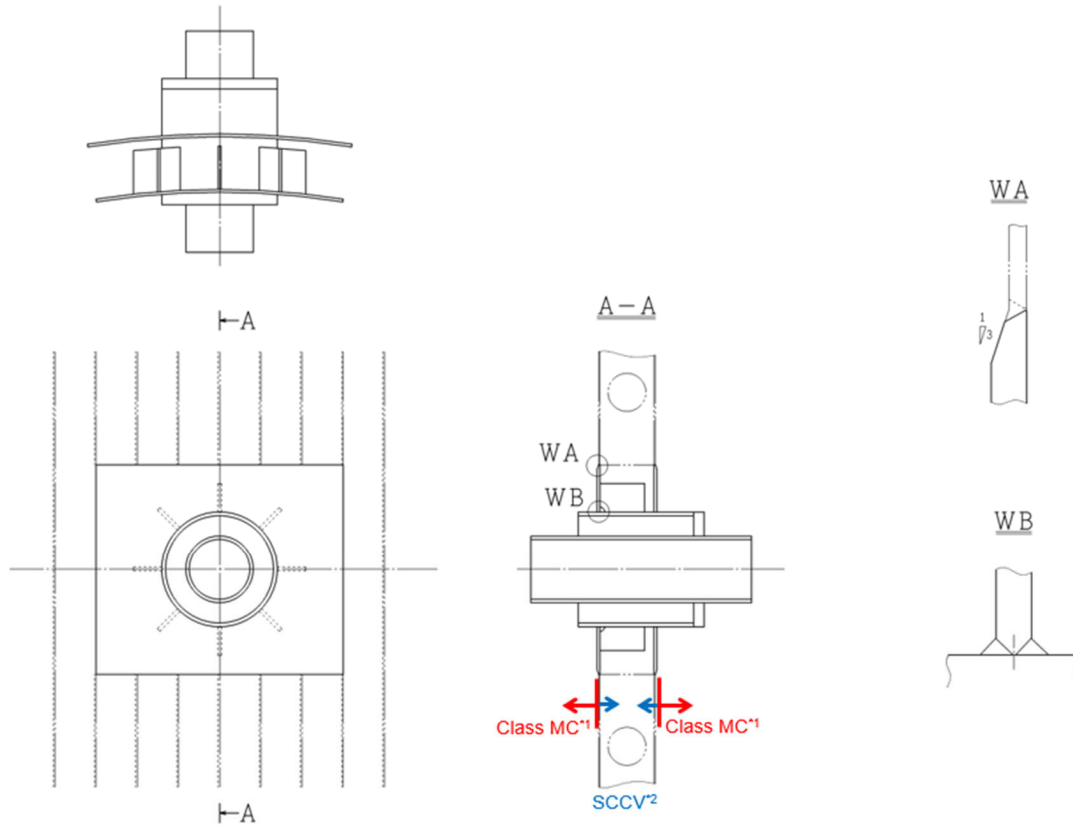


Figure 3.5-4: Penetrations Jurisdictional Boundary

*1: Is designed in accordance with ASME Section III Subsection NE (for Class MC)

*2: Is designed in accordance with NEDC-33926P

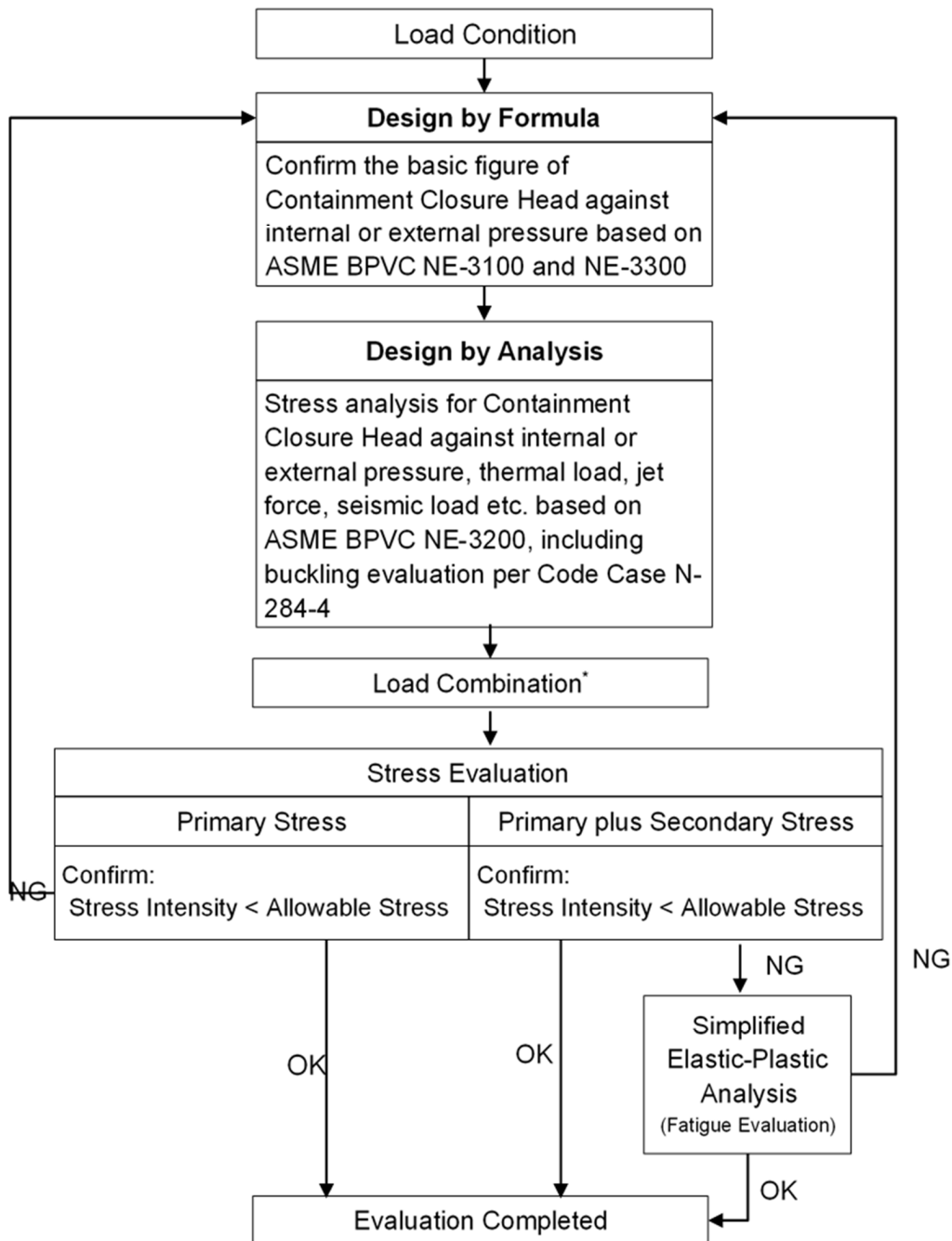


Figure 3.5-5: Design Procedures for the Containment Closure Head

*: Steel Portion: U.S. NRC RG 1.57 and U.S. NUREG-0800 SRP 3.8.2

Concrete Portion: NEDC-33926P

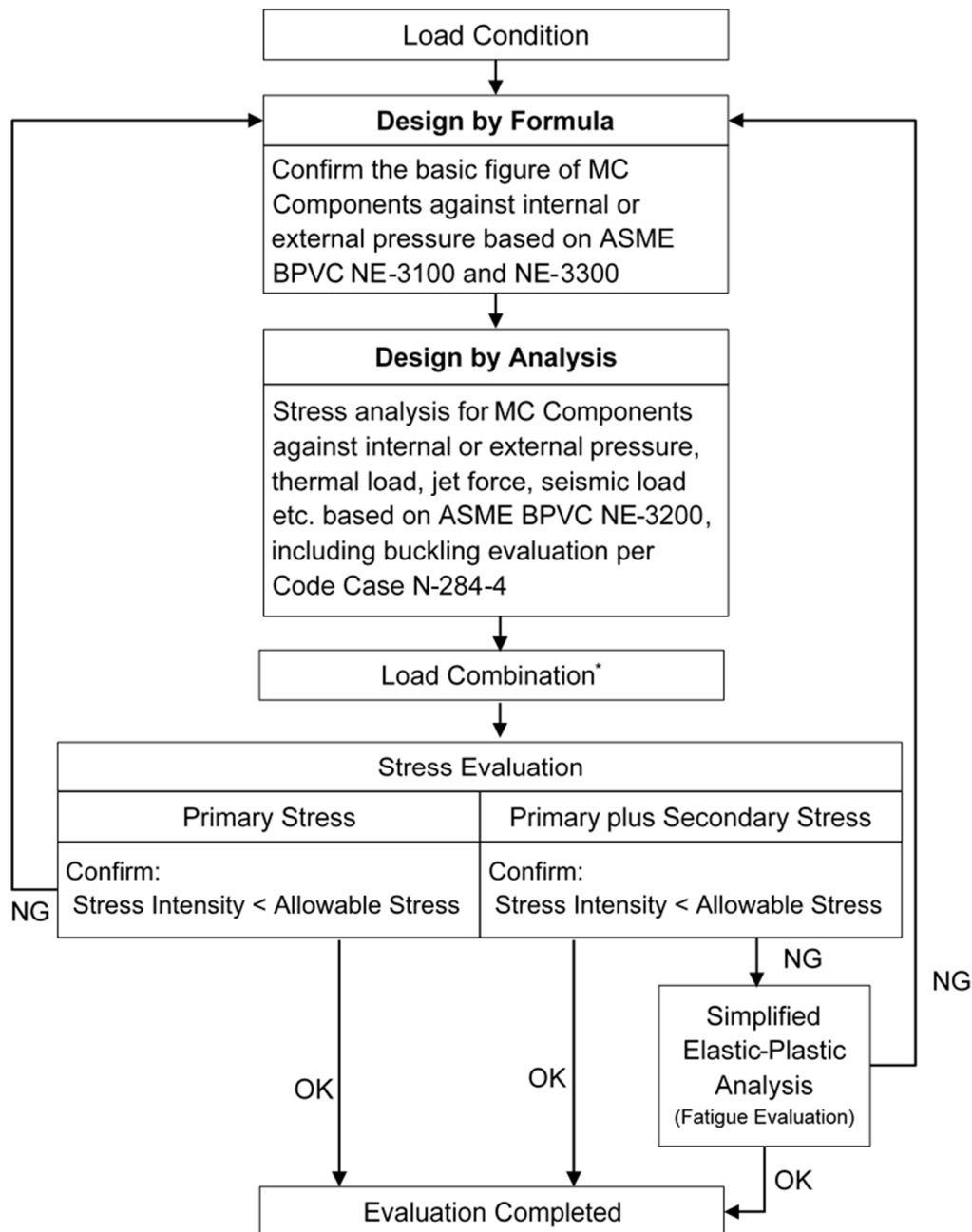


Figure 3.5-6: Design Procedures for the MC Components

*: Steel Portion: US NRC RG 1.57 and U.S. NUREG-0800 SRP 3.8.2

Concrete Portion: NEDC-33926P

3.6 General Design Aspects for Mechanical Systems and Components

Section 3.6 provides the general design aspects used for safety class and non-safety class mechanical systems and components. It includes special considerations for mechanical components, dynamic testing and analysis of structures, systems, and components, required codes for ASME BPVC Section III Division 1 Class 1, 2, and 3 components, and component supports, including core support structures. In addition, general design aspects for Control Rod Drive System, Reactor Vessel Internals, system piping, and threaded fasteners are presented. Further, this section discusses the functional design, qualification and in-service testing program requirements for pumps, valves, and dynamic restraints.

Chapter 1 provides the codes and standards and editions that are applicable to the design of mechanical systems and components and is used as input to Section 3.6.

Sections 3.1 and 3.2 are used as input to Section 3.6 and provide the general design principles, criteria, and classification used for design of mechanical systems and components. Among these principles are design for robustness, reliability, and fail-safe operation. Additionally, the systems and components are required to be redundant, diverse, independent, separate and of supply quality commensurate with the safety classification, seismic category, and supply category. The design and qualification of mechanical components is performed using a graded approach with the highest level of rigor applied to Safety Class 1 (SC1) components.

Subsection 3.3.1 develops the seismic input criteria and building spectra used as input to Section 3.6 for seismic qualification of Seismic Category B active mechanical components and system functionality. Additionally, Seismic Category A passive mechanical component supports, and equipment supports use the seismic spectra for qualification.

Section 3.9 provides the equipment qualification requirements including environmental, dynamic, functional qualification, and Electromagnetic Compatibility (EMC), which are used as input to Section 3.6.

Codes and Standards Used in the Design of Mechanical Systems and Components

ASME BPVC Section III Division 1, ASME B31.1 (Reference 3.6-10), and ASME B31.3 (Reference 3.6-12) are applied for the design of mechanical systems, components and piping including piping components.

Table 3.6-1 provides the pressure boundary codes and standards utilized in the BWRX-300 mechanical system and component design.

Mechanical Equipment Separation for Safety Class 1

Mechanical equipment separation measures for the BWRX-300 contribute to system reliability in the performance of any Safety Category 1 function including (but not necessarily limited to) interconnecting piping, valves, and associated mechanical controls and instrumentation. Additionally, where necessary adjacent systems are considered in mechanical equipment separation (as related to human factors, mechanical maintenance, and seismic interaction).

Principles of physical separation include:

- A. Separation by geometry (layout, distance, orientation, elevation, and including separate structures)
- B. Separation by barriers (e.g., walls, shields), both vertical and horizontal
- C. Separation by a combination of (A) and (B)

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Per CNSC REGDOC-2.5.2, Section 7.4.1 (Reference 3.6-16), the plant design takes into account the potential for internal hazards such as flooding, missile generation, pipe whip, jet impact, fire, smoke, and combustion by-products, or release of fluid from failed systems or from other installations on the site. Appropriate preventive and mitigation measures are provided to ensure that nuclear safety is not compromised.

Per CNSC REGDOC-2.5.2, Section 7.6.1.1, vertical separation, or other protection is provided where physical separation by horizontal distance alone may not be sufficient for some common cause failures such as flooding.

Defense Line (DL) functions that mitigate the same event are independent from each other to the extent practicable. All PIEs with a frequency greater than 1E-05 can be mitigated by functions in DL3 and separately by functions in either DL2 or 4a. Therefore, SSC performing DL3 functions are separate, to the extent practicable, from SSC that perform Safety Category functions in DL2 and DL4a. Separation is also provided between redundant SSC that perform DL3 functions (Safety Category 1) to the extent practicable.

The redundancy methods are used to protect from Single Active Failures or events; examples include utilization of safety class structures, spatial separation, three-hour rated fire barriers, and isolation devices.

The application of the single failure criterion to fluid systems is described in Subsection 3.1.7.5.

Separation of components may be by physical distance or by barriers. An example is the provision of principal fire barriers to delineate individual fire zones; such barriers may also serve as barriers to other hazards, as per CNSC REGDOC-2.5.2 Section 7.6.1.1.

The following SC mechanical equipment items are considered:

- Piping Systems
- Valves
- Rotating Equipment
- Vessels
- Ductwork Systems
- Instrumentation

Piping Systems

Piping systems include piping to and from SC and SCN SSC. These include their connected bellows, mechanical connections, support guides, and structural supports. They may include wall or floor sleeves and penetrations, pipe fittings including welds and branch connections, structural restraints (and appurtenances), and attached sampling. Piping systems also include vent/drain/test/flush/clean-out taps including closures, instrument sensing line piping or tubing and instrument racks. Finally, they also include pneumatic or hydraulic system tubing, manifolds and controls appurtenances.

Valves

Valves include those that control fluid flow to and from SC and SCN SSC. Valves include the valve body assembly, actuators, appurtenances, and all non-electrical connections.

Rotating Equipment

Rotating equipment includes pumps, fans and compressors, gear sets or power coupling subsystems, and electric motors or other rotary-power driven subsystems. Their components include rotating casing, including base, frame, supports and drive.

Vessels

Vessels include heat exchangers and tanks, including their supports, filter assemblies, and nozzles.

Ductwork Systems

Ductwork systems include:

- Duct runs
- Active and pre-set dampers
- Fire dampers
- Screens
- Vents/reliefs/blow out panels
- Filters or air filtration assemblies/subsystems

Instrumentation

Instrumentation includes:

- Mechanically activated instruments used to monitor reactor and plant processes
- The associated non-electrical transmission
- Sensors
- Actuator systems
- In-line instruments with associated taps

Zone of Influence

The degree and type of separation required varies with the following potential hazards in a power plant zone:

1. **Missiles** - A missile is an unrestrained mass with sufficient kinetic energy to cause damage to the safety systems or required safety components. Definition of missile and missile protection requirements are addressed in Subsection 3.3.5
2. **Pipe Whip** - Pipe whip is usually consequent to a pipe failure resulting in a complete segment separation break. The area in the vicinity of the postulated break of high-energy piping is defined as the pipe whip damage zone. Pipe whip protection requirements are addressed in Subsection 3.4.4.
3. **Fluid Jet** - The fluid jet is usually consequent to a high-energy pipe break but may also be the result of intentional equipment action. Jet impingement protection requirements are addressed in Subsection 3.4.4

Fire Area and Fire Zone

A fire area is an area sufficiently bounded to withstand the hazards associated with the fire area and, as necessary, to protect important equipment within the fire area from a fire outside the area. A fire zone, however, is a subdivision of fire area(s) for analysis purposes that is not necessarily bound by fire-rated barriers.

Fire zone protection requirements are addressed in Chapter 9A, Section 9A.6. Separation of vulnerable mechanical equipment from areas containing significant combustible materials is provided by fire barrier materials or housings, fire-rated walls or doors (including consideration for ductwork isolations), barrier piping around processes containing flammable or combustible fluids to isolate the hazard, and in certain locations by atmospheric inerting (oxygen concentration suppression below combustible level or replacement with nitrogen, such as in containment).

Flood Zone

Internally generated flooding may occur by pipe or tank failure, fire suppression system operation, misaligned systems with openings in the affected zone, maintenance errors, or failure of a drainage system. Flood protection requirements are addressed in Subsections 3.3.3.1 and 3.4.2.

Separation by flood hazard containment walls, dikes, curbs, trenches or pits, watertight doors, elevated equipment mounting location (mezzanine or different floor) or pedestals or placing vulnerable equipment in watertight housings may be used.

Design Load and Load Combination for Mechanical Systems and Components

Design loads and loading combinations are based on normal operation and off-normal operation. Subsection 3.6.1.1 below provides the operational transients, resulting loads, and load combinations.

Design loads and load combinations for fixed mechanical equipment are provided in Table 3.6-2. Fixed equipment includes the mechanical, electrical, and instrument components, and the component housings and structural supports that are anchored to civil structure(s) but are not a part of the civil structure itself, such as mechanical or electrical penetrations. Examples include the reactor pressure vessel (RPV), RPV Internals, RPV supports, instrumentation, piping, electrical equipment, and the component supports.

A discussion of plant normal and off-normal operation can be found in Chapter 1, Section 1.8, and Chapter 6, Sections 6.2 and 6.4.

Design for System Duty of Mechanical Systems Based on Event Frequencies

Table 3.6-3 is used as a general event list for all hardware system duty design specifications. Events are mainly classified into:

- Design Condition 1 (DC-1): Normal Planned Operation
- Design Condition 2 (DC-2): Anticipated Operational Occurrence
- Design Condition 3 (DC-3): Design Basis Accident
- Design Condition 4 (DC-4): Design Extension Condition

The BWRX-300 utilizes the four Service Levels used in the ASME Code, Levels A, B, C and D, as well as testing conditions, in the design of fixed equipment. The design basis specifies the capabilities that are necessary for the plant in various operational states.

Conservative design measures and sound engineering practices are applied in the design basis for plant states. This approach provides a high degree of assurance that no significant damage

will occur to the reactor core, and that radiation doses will remain within established regulatory limits.

3.6.1 Special Topics for Mechanical Components

This subsection addresses information concerning methods of analysis for components and supports.

3.6.1.1 Computer Programs Used in Analyses

The major computer programs used in the mechanical system and component analyses of the major safety class components are described in Chapter 3, Appendix 3C .

The computer programs used in the analyses of Seismic Category A and B components are maintained either by General Electric Company (GE) or by outside computer program developers.

The GEH Software is controlled under NEDO-11209-A (Ref. 3.6-17). CSA N286.7 (Ref. 3.6-14) is used to determine acceptability of code use for the BWRX-300 in Canada. In either case, the quality of the programs and the computed results are controlled. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature.

3.6.1.2 Operational Transients, Resulting Loads and Load Combinations

The plant duty cycles represent transient conditions that are used for development of the BWRX-300 system and component design during Normal Operation, Anticipated Operational Occurrence (AOOs), Design Basis Accidents (DBAs), and Design Extension Conditions (DECs), which are Beyond Design Basis Events. Requirements are evaluated for the system design and performance as it relates to complete reactor operation. The duty is recorded as inputs to the system design for each specific primary and auxiliary hardware system. Duty can be defined from a pressure and temperature perspective, mostly when variations in either variable are expected in important locations for the reactor.

The number of cycles associated with each event for the design of the Reactor Pressure Vessel (RPV), Reactor Coolant Pressure Boundary (RCPB), and other ASME pressure boundary components designed for fatigue are listed in Table 3.6-9. Tables 3.6-4 through Table 3.6-8 break down the operational cycles by plant condition. The plant operating conditions are identified as normal, AOO, DBA, DEC, or testing as defined in Subsection 3.6.3.2. Appropriate Service Levels (A, B, C, D, or testing), as defined in the ASME BPVC, are designated for design limits. The design and analyses of ASME Class piping and equipment using specific applicable thermal-hydraulic transients, which are derived from the system behavior during the events listed in Table 3.6-3, are documented in the design specifications and/or stress reports of the respective equipment. Table 3.6-2 shows the load combinations and the standard acceptance criteria for ASME Section III components. Tables 3.6-10, 3.6-11, and 3.6-12 provide the specific load combinations and acceptance criteria for piping systems.

3.6.1.3 Experimental Stress Analysis

Experimental stress analysis methods are used in compliance with the provisions of ASME BPVC Section III Division 1, Mandatory Appendix II (Reference 3.6-9). ASME Class 1 and some ASME Class 2 mechanical components that require both functionality and adequate structural capacity during seismic events, are laboratory tested in accordance with CSA N289.4 (Reference 3.6-13) and ASME Standard QME-1 (Reference 3.6-20) as discussed in Subsection 3.9.3.2.1.

3.6.1.4 Considerations for the Evaluation of Fault Conditions

All equipment designed to ASME BPVC Section III Division 1 is evaluated for the faulted (Service Level D) loading conditions. In all cases, the calculated actual stresses are compared to the allowable ASME BPVC Section III Division 1 Service Level D limits. The following subsections address the evaluation methods and stress limits used for the equipment and identify the major components evaluated for faulted conditions.

Deformations under faulted conditions are evaluated in critical areas and the necessary design deformation limits, such as clearance limits, are satisfied.

3.6.1.4.1 Fine Motion Control Rod Drive

The Fine Motion Control Rod Drive (FMCRD) major components that are part of the RCPB are analyzed and evaluated for the ASME Service Level D faulted conditions in accordance with the ASME BPVC Section III Division 1, Subsection NB (Reference 3.6-3). Refer to Chapter 4, Subsection 4.6.2.1.1 for FMCRD mechanism details.

3.6.1.4.2 CRD Hydraulic Control Unit

The Hydraulic Control Unit (HCU) is analyzed and tested for withstanding the faulted condition loads. Dynamic tests that are part of the seismic and dynamic qualification program establish the loads in the horizontal and vertical directions as the HCU capability for the frequency range that is likely to be experienced in the plant. These tests also ensure that the reactor trip function of the HCU can be performed under these loads. Dynamic analysis of the HCU with the mounting beams is performed to assure that the maximum faulted condition loads remain below the HCU capability. Refer to Chapter 4, Subsection 4.6.2.1.3 for HCU details.

3.6.1.4.3 Reactor Pressure Vessel Assembly

The design of the RPV assembly, out to and including the integral Reactor Isolation Valves (appurtenances), RPV Top Head, and housings for FMCRD and in-core Nuclear Instrumentation complies with Subsections NB and NG of the ASME BPVC Section III Division 1 as applicable. For faulted conditions, the reactor vessel is evaluated using elastic analysis.

Elastic analysis methods and standard design rules, as defined in the ASME BPVC, are utilized in the analysis of the pressure boundary, Seismic Category B, ASME BPVC Section III, Division 1, Class 1 valves. The ASME BPVC Section III Division 1 allowable stress is applied to assure integrity under applicable loading conditions including faulted condition. The functional qualification of the Reactor Isolation Valves (RIVs), is analyzed and/or tested for seismic and other dynamic conditions.

3.6.1.4.4 Core Support Structures and Other Safety Class Reactor Internal Components

The core support structures, the internal portion of Nuclear Instrument and CRI housings, and other safety class reactor internal components are evaluated for faulted conditions. The basis for determining the faulted loads for seismic events and other dynamic events is given in Subsection 3.6.2.3 and Subsection 3.6.2.2, respectively. The allowable Service Level D limits for evaluation of these structures are per ASME BPVC Section III Division 1, Service Level D equations.

For the shroud support, an elastic analysis is performed, and buckling is evaluated for compressive load cases for certain locations in the assembly.

3.6.1.4.5 RPV Stabilizers, Reactor Skirt and FMCRD Housing and Nuclear Instrumentation Housing Restraints (Supports)

The calculated maximum stresses to meet the allowable stress limits are based on the ASME BPVC Section III Division 1, Subsection NF (Reference 3.6-7), for the RPV stabilizer, RPV skirt

and supports for the FMCRD housing and Nuclear Instrumentation housing for faulted conditions. These supports restrain the components during earthquake, pipe rupture or other Reactor Building Vibration events.

3.6.1.4.6 Reactor Isolation Valves, and Other ASME BPVC Section III Division 1 Class 1 and 2 Valves

Elastic analysis methods and standard design rules, as defined in the ASME BPVC, are utilized in the analysis of the pressure boundary, Seismic Category B, ASME BPVC Section III Division 1 Class 1 and 2 valves. The ASME BPVC Section III Division 1 allowable stresses are applied to assure integrity under applicable loading conditions including faulted condition. The functional qualification of the major active valves, including Reactor Isolation Valve (RIVs), Containment Isolation Valves (CIVs), ICS Purge valves, and ICS Condensate Return valves are analyzed and/or tested for seismic and/or other dynamic conditions.

3.6.1.4.7 Fuel Storage and Refueling Equipment

The fuel storage and fuel handling equipment is described in detail in Section 9A.1. This includes the Fuel Pool structure, Fuel Racks, Fuel Cooling system, and Fuel Handling Equipment.

CNSC REGDOC 2.5.2 Section 6.2, Subsection 7.3.4.1, and Subsection 8.12.2, require that the same Section 3.1 fundamental safety functions as those that apply to the Reactor be utilized for fuel storage and handling. Due to physical and structural separation, Safety Class equipment cannot be affected by a fuel handling accident.

A summary of the design considerations used to establish nuclear criticality safety under all operational and faulted (ASME Service Level D) conditions is described below.

All fuel storage racks are designed and qualified to operate within their performance requirements under the anticipated ranges of the normal, abnormal or accident plant environments and are designed to withstand a Design Basis Earthquake (DBE) without failure of the basic structure or damage to the active region of irradiated fuel.

3.6.1.4.8 Fuel Assembly (Including Channel)

The Fuel Assembly including channel is described in detail in Section 4.2.3.

The channel is subjected to mechanical tests to demonstrate the adequacy of the GNF2 channel for seismic/dynamic loads. The channel was tested to determine the allowable bending load that could be sustained without buckling or collapsing the channel.

The Fuel Assemblies are designed for worst-case conditions that evaluate maximum stresses, fatigue, control rod insertion, fretting, corrosion/hydriding, and compatibility/dimensional changes. The results of the testing and analysis requires that the safety class components maintain the required functionality and structural capacity during ASME Level D service conditions.

3.6.1.4.9 ASME BPVC Section III Division 1 Class 2 and 3 Vessels

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 vessels. The equivalent allowable stresses using elastic techniques are obtained from Articles NCD-3300 and NCD-3200 of the ASME BPVC Section III Division 1 Subsection NCD (Reference 3.6-4).

3.6.1.4.10 ASME BPVC Section III Division 1 Class 2 and 3 Pumps

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 pumps. The equivalent allowable stresses for nonactive pumps using elastic techniques are obtained from Article NCD-3400 the ASME BPVC Section III Division 1 Subsection NCD.

3.6.1.4.11 ASME BPVC Section III Division 1 Class 2 and 3 Valves

Elastic analysis methods and standard design rules are used for evaluating faulted loading conditions for Class 2 and 3 valves. The equivalent allowable stresses for valves using elastic techniques are obtained from Article NCD-3500 of the ASME BPVC Section III Division 1 Subsection NCD.

3.6.1.4.12 ASME BPVC Section III Division 1 Class 1, 2 and 3 Piping

Elastic analysis methods are used for evaluating faulted loading conditions for Class 1, 2, and 3 piping. The equivalent allowable stresses using elastic techniques are obtained from Article NB-3600 (for Class 1 piping) of the ASME BPVC Section III Division 1 Subsection NB and Article NCD-3600 (for Class 2 and 3 piping) of the ASME BPVC Section III Division 1 Subsection NCD.

3.6.1.4.13 Inelastic Analysis Methods

Inelastic analysis is only applied to BWRX-300 components to demonstrate the acceptability of two types of postulated events. Each event is an extremely low-probability occurrence and the equipment affected by these events would not be reused. These two events are as follows:

- Postulated gross piping failure
- Postulated blow out of a Control Rod Drive housing caused by a weld failure

The design criteria for pipe failure effects and mitigating features are provided in Subsection 3.4.4.1. Except for the analysis of pipe failures, inelastic methods are not used in BWRX-300 piping design.

The mitigation of the CRDH attachment weld failure relies on components with regular functions to mitigate the weld failure effect. The components are specifically:

- Core support plate
- Control Rod Guide Tube
- CRD Housing
- Control Rod Drive (CRD) outer tube
- Bayonet Fingers

Only the bodies of the CRGT, CRDH, and CRD outer tube are analyzed for energy absorption by inelastic deformation.

3.6.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

This Subsection 3.6.2 presents the criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports (including supports for conduit and cable trays, and ventilation ducts) under vibratory loadings, including those due to fluid flow and postulated seismic events. Structural requirements for conduits and cable tray supports and Heating, Ventilation and Air Conditioning duct supports are developed as discussed in Subsection 3.6.2.5.7.

3.6.2.1 Piping Vibration, Thermal Expansion and Dynamic Effects

The overall test program is divided into two phases:

1. Pre-operational test phase
2. Initial startup test phase

Piping vibration, thermal expansion, and dynamic effects testing is performed during both of these phases. Discussed below are the general requirements for this testing. It is noted that because one goal of the dynamic effects testing is to verify the adequacy of the piping support system, such components are addressed in the subsections that follow.

3.6.2.1.1 Vibration and Dynamic Effects Testing

The purpose of these tests is to confirm that the piping, components, restraints, and supports of specified high- and moderate-energy systems have been designed to withstand the dynamic effects of steady-state Flow Induced Vibration (FIV) and anticipated operational transient conditions.

3.6.2.1.2 Seismic Qualification of Safety Class Mechanical Equipment

Section 3.9 provides methodology for qualification of SC1 Mechanical equipment.

3.6.2.1.3 Tests and Analysis Criteria and Methods

Section 3.9 provides tests and analysis criteria methods.

3.6.2.2 Qualification of Safety Category Mechanical Equipment

The following subsections discuss the testing or analytical qualification of the safety class major mechanical equipment, and other ASME BPVC Section III Division 1 equipment including equipment supports.

3.6.2.2.1 CRD and CRDH

The qualification of the CRDH (with enclosed FMCRD) is done analytically, and the stress results of the analysis establish the structural integrity of these components. Dynamic tests are conducted to verify the operability of the CRD during a dynamic event. A simulated test, imposing dynamic deflection in the fuel channels up to values greater than the expected seismic response, is performed.

The correlation of the test with analysis is via the channel deflection, not the housing structural analysis, because insert ability is controlled by channel deflection, not housing deflection.

3.6.2.2.2 Core Support (Fuel Support and Control Rod Guide Tube)

A detailed analysis imposing dynamic effects due to seismic and other RBV events is performed to show that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.6.2.2.3 CRD Hydraulic Control Unit

The HCU is analyzed for the seismic and other RBV loads in the faulted condition and the maximum stress on the HCU frame is calculated to be below the maximum allowable for the faulted condition.

3.6.2.2.4 Fuel Assembly (Including Channel)

The Fuel Assembly (including channel) qualification for seismic and faulted load conditions is described in Chapter 4, Subsections 4.2.2 and 4.2.3.

3.6.2.2.5 Containment Isolation Valves and Reactor Isolation Valves

The CIVs for main steam and other process system piping that penetrates containment, and RIVs are qualified for seismic and other RBV loads. The fundamental requirement following a Design Basis Earthquake (DBE) or other faulted RBV loadings is to close and remain closed after the event. This capability is demonstrated by the test and analysis.

3.6.2.2.6 Other ASME BPVC Section III Division 1 SSCs

Other equipment, including associated supports, is qualified for seismic and other RBV loads to ensure its functional integrity during and after the dynamic event. The equipment is tested, if necessary, to ensure its ability to perform its specified function before, during, and following a seismic event.

Dynamic load qualification is done by testing, analysis, or both as described in Section 3.9.

Refer to Section 3.9 for additional information on the dynamic qualification of valves.

3.6.2.2.7 Supports

Analyses or tests are performed for component supports to assure their structural capability to withstand seismic, faulted, and other dynamic excitations. Pre-qualified manufactured standard component supports, or engineered component supports that are qualified to specified required service levels for seismic, faulted, and dynamic excitation do not require additional analyses or testing.

3.6.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The major reactor internal components within the vessel are subjected to extensive testing, coupled with dynamic system analyses, to properly evaluate the resulting FIV phenomena during normal reactor operation and from anticipated operational transients.

3.6.2.3.1 Initial Startup Flow Induced Vibration Testing of Reactor Internals

A reactor internals vibration measurement and inspection program is conducted only during initial startup testing. These reactor internal inspections and tests consist of evaluating Flow Induced Vibrations, including any flow excited acoustic and structural resonance that is detected in initial startup testing. Analytical thermal-hydraulic fluid models are developed that replicate plant startup conditions to predict resonance effects on the reactor internals. These predictive models are used in design to eliminate undesired acoustics and structural resonances to a practical extent.

3.6.2.3.2 Initial Startup Testing

Vibration measurements are made during reactor startup at conditions up to 100% rated flow and power. Steady-state and transient conditions of natural circulation flow operation are evaluated. The primary purpose of this test series is to verify the anticipated effect of single- and two-phase flow on the vibration response of internals.

3.6.2.3.3 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

The loads to the Reactor Internals that occur because of faulted events and the deterministic analyses performed to determine the response of the reactor internals are as follows:

- Reactor Internal Pressures
- External Pressure and Forces on the Reactor Vessel
- LOCA Loads
- Seismic Loads

3.6.2.3.4 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Prior to initiation of the instrumented vibration measurement program for a prototype plant, extensive dynamic analyses of the reactor and internals are performed. The results of these

analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test are analyzed in detail.

The results of the data analyses, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been used in the generation of the dynamic models for seismic and LOCA analyses for this plant. The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.

3.6.3 Codes for ASME BPVC Section III Division 1, Class 1, 2 and 3 Components, Component Supports and Core Support Structure

Subsection 3.6.3 discusses the structural integrity and/or functional integrity requirements of pressure-retaining components, their supports, and core support structures that are designed in accordance with the rules of the ASME BPVC Section III Division 1.

The ASME BPVC Section III Division 1, Section III, requires that a design specification be prepared for ASME BPVC Section III Division 1 Class 1, 2 and 3 components. The design specifications for ASME BPVC Section III Division 1 Class 1, 2 and 3 components, supports, and appurtenances are prepared under administrative procedures that meet the ASME BPVC Section III Division 1 rules. The specifications conform to and are certified to the requirements of the applicable subsection of the ASME BPVC Section III Division 1. The ASME BPVC Section III Division 1 also requires design reports for Class 1, 2 or 3 components be prepared which demonstrate that the as-built components satisfy the requirements of the respective ASME design specification for each component and the applicable ASME BPVC Section III Division 1. These design specifications and the design reports are completed by the licence applicant, or the applicant's authorized agent, in accordance with the responsibilities outlined under the ASME BPVC Section III Division 1. The ASME BPVC Section III Division 1 design reports include the record of as-built reconciliations, for example, the evaluations of changes to piping support locations, the pre-operational testing, and results, and reported construction deviation resolution, and includes the small-bore piping analysis.

3.6.3.1 Loading Combinations, Design Transients and Stress Limits

Subsection 3.6.3.2 delineates the criteria for selection and definition of design limits and loading combinations associated with Normal Operation, Anticipated Operational Occurrence (AOO), Design Basis Accidents (DBAs), Design Extended Conditions (DECs) and specified seismic and other RBV events for the design of safety ASME BPVC Section III Division 1 components (except containment components which are discussed in Section 3.5).

This section discusses the ASME BPVC Section III Division 1 Class 1, 2, and 3 equipment and associated pressure-retaining parts and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. A discussion of major equipment is included on a component-by-component basis to provide examples. Design transients and dynamic loading for ASME BPVC Section III Division 1 Class 1, 2 and 3 equipment are covered in Subsections 3.6.1.1, 3.6.3.6 and 3.6.3.7. Seismic-related loads and dynamic analyses are discussed in Subsection 3.3.1. Table 3.6-9 presents the plant events to be considered for the design and analysis of all BWRX-300 ASME BPVC Section III Division 1 Class 1, 2, and 3 components, component supports, equipment, and core support structures per ASME BPVC Section III Division 1 Subsection NG (Reference 3.6-8). Specific loading combinations considered for evaluation of specific equipment are derived from Table 3.6-2 and are contained in the design specifications and design reports for the respective equipment. For Class 1 components where analysis for

cyclic operation is evaluated in accordance with ASME BPVC Section III Division 1 subarticle NB-3222.4, the fatigue usage evaluation includes the use of environmental fatigue curves.

Specific load combinations and acceptance criteria for Class 1 piping are shown in Table 3.6-10. Also, for Class 1 piping, the operating temperatures above ambient or below ambient are included in the fatigue analysis. The installation temperature state for the piping system is defined as a temperature of 21 C for Class 1, 2, 3 or ASME B31.1 piping.

The design life for the BWRX-300 Standard Plant is 60 years. A 60-year design life is a requirement for all major plant components. Additional life is added for components required during decommissioning. However, all plant operational components and equipment except the reactor vessel are designed to be replaceable. The design life requirement allows for refurbishment and repair, as appropriate, to assure that the design life of the overall plant is achieved.

3.6.3.2 Events Considered in Evaluating Effect of Loads on Fixed Equipment

All events that the BWRX-300 might credibly experience during a reactor-year are evaluated in Chapter 15, to establish the plant design basis, including plant fixed equipment. The associated loads and duty cycles associated with each event are considered in combination with additional events in load combinations as applicable. These event combinations are divided into the four plant conditions with associated frequency of occurrence and ASME BPVC Section III Division 1 design levels.

The following are the plant condition events and transients associated with the BWRX-300 design:

3.6.3.2.1 Normal Operation

Normal planned operation is operation under any condition permitted within specified Operational Limits and Conditions (OLCs) irrespective of the anticipated frequency of occurrence of that condition, which is planned and deliberate and not in specific response to Postulated Initiating Events (PIEs). Normal planned operations include startup, power operation, shutting down, shutdown, maintenance, testing, and refueling.

Adequate evaluation of normal operation loads includes loads due to dead weight, temperature, prestress, pressure, fluid flow (including FIV when applicable), thermal and fluid reaction forces and other loads due to moving parts within a component or system. Such loads are considered in the design, installation, and mounting, of equipment and components.

3.6.3.2.2 Anticipated Operational Occurrences

Anticipated Operational Occurrences (AOO) are those operating transient events that are expected to occur more frequently than 1E-02 per reactor-year. Chapter 15, Subsection 15.5.3 provides event analyses of Level B PIE AOOs.

Adequate evaluation of associated loads, load combinations, and duty cycles of the AOO transient effects are considered in the design, installation, and mounting, of equipment and components.

3.6.3.2.3 Design Basis Accident Events

Design Basis Accidents (DBA) are those events with frequencies of occurrence between 1E-02 to 1E-05 per reactor-year DBAs are mitigated by Defense Line 3. Chapter 15, Subsection 15.5.4 provides event analyses of Level C PIE DBAs.

3.6.3.2.4 Design Extension Condition Events (DEC)

Design Extension Conditions (DEC) are events that are less frequent than 1E-05 reactor-year. DEC event analyses demonstrate the capability of the plant to cope with scenarios involving

Defense Line 3 Common Cause Failures (CCFs) and provide a systematic evaluation of potential cliff-edge effects outside the plant design bases. DEC transient events are mitigated by SSC associated with Defense Line 4a and DL2 functions that are unaffected by the PIE and additional failures identified in the event sequence. Chapter 15, Subsections 15.5.5 through 15.5.9 provides event analyses of Level D PIE DEC's.

3.6.3.2.5 Seismic Events

Seismic design parameters and associated seismic events defined in Subsection 3.3.1 are used in qualification of mechanical system components. The magnitude of seismic events is determined by Ground Response Spectra accelerations applied to Building Structures and creating Amplified Response Spectra (ARS) accelerations at various building elevations where the components are located. These ARS accelerations are used in qualification of Mechanical systems and equipment and a determination of component and system structural and/or functional capacity is determined. Seismic Category A (passive components) require only structural code adequacy. Seismic Category B (active components) such as valves and pumps require both structural code adequacy and functional capacity under seismic demand. Chapter 3, Subsection 3.9.3 provides seismic qualification methodology to assure both component structural and/or functional capacity under seismic operational conditions are met.

The seismic categorization of SSC is defined in Section 3.2 and related to the seismic category to the more general safety strategy defense lines. In summary, Defense Lines 3 and 4b are generally Seismic A or B and Defense Line 4b also has an additional requirement of satisfying the plant-level High Confidence of Low Probability of Failure (HCLPF) criteria.

3.6.3.2.6 Non-LOCA Fault

Non-LOCA Fault consists of any DEC event not considering a LOCA which has a significantly low frequency of occurrence to be considered as a faulted event.

3.6.3.2.7 Plant Testing

Plant testing events are occasional operating loads imposed during pre-operational testing or periodic operational testing.

3.6.3.3 Classification of Components

All SSC of the BWRX-300 design are designated by Safety Class, Quality Group, and Seismic Category according to guidance in Section 3.2 which are consistent with their Defence-in-Depth categorization defined in the BWRX-300 Safety Strategy, in Section 3.1. Appendix 3A provides the Classification Table for Plant SSC.

3.6.3.4 Establishment of Design, Service, and Test Loadings and Limits

Design, Service, and Test Loadings and Limits for fixed equipment components and supports are in accordance with ASME BPVC Section III Division 1 (Reference 3.6-5).

For IEEE Equipment, SC1 electrical equipment is evaluated with respect to the load combinations in this document using IEC/IEEE 60980--323 and IEC/IEEE 60980--344 Acceptance Criteria, Codes and Standard (References 3.6-18 and 3.6-19).

For SC1, actuators and power operated valve assemblies are evaluated with respect to the load combinations in this document in accordance with the provisions of ASME Standard QME-1.

3.6.3.5 Acceptance Criteria

Components and supports comply with the design rules established for design, service, and test loadings in the appropriate with the appropriate subsection of the ASME BPVC, Section III, Division 1 (References 3.6-1 through 3.6-8).

Design documentation is completed in accordance with the requirements of the Subsection of the ASME BPVC applicable to the component or support.

3.6.3.6 Loading Criteria

3.6.3.6.1 Loading Conditions

The loadings that are considered in designing a component include, but are not limited to, those in (a) through (g) below:

- a. Internal and external pressure
- b. Impact loads, including rapidly fluctuating pressures
- c. Weight of the component and normal contents under operating or test conditions
- d. Superimposed loads such as other components, operating equipment, insulation, corrosion resistant or erosion resistant linings, and piping
- e. Wind loads, snow loads, vibrations, and earthquake loads, where specified
- f. Reactions of supporting lugs, rings, saddles, or other types of supports
- g. Temperature effects

As appropriate ASME BPVC, Division 1, Section III, Paragraph, NB-3111, NCD-3111, NE-3111, NF-3111 or NG-3111, is applied for a complete list of required load conditions to consider.

Consistent with the ASME BPVC Section III Division 1, the stresses resulting from differential anchor movements during dynamic events are considered secondary stresses.

3.6.3.6.2 Design Loadings

The Design Loadings are established in accordance with ASME BPVC Section III Division 1, Paragraph NB-3112, NCD-3112, NE-3112, NF-3112 or NG-3112, as applicable.

3.6.3.6.3 Service Conditions

The Design Loadings are established in accordance with ASME BPVC Section III Division 1, Paragraph NB-3113, NCD-3113, NE-3113, NF-3113 or NG-3113, as applicable.

Each service condition to which the components may be subjected is classified in accordance with Service Limits designated in the Component Design Specifications in such detail as will provide a complete basis for design, construction, and inspection.

For ASME BPVC Section III Division 1, Class 1 Components, the requirements of (1) and (2) below apply.

1. Level B Conditions. The estimated duration of service conditions for which Level B Limits are specified are included in the Design Specifications.
2. Level C Conditions. The total number of postulated occurrences for all specified service conditions for which Level C Limits are specified are limited to no more than 25 stress cycles having a S_a value greater than that for 10^6 cycles from the applicable fatigue design curves of Section III Appendices, Mandatory Appendix I.

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When the Component Design Specification requires computations to demonstrate compliance with specified Service Limits, the Component Design Specification provides information from which Service Loadings can be identified (pressure, temperature, mechanical loads, cycles, or transients).

Design Pressure - The specified internal and external Design Pressure is not to be less than the maximum difference in pressure between the inside and outside of the item, or between any two chambers of a combination unit, which exists under the most severe loadings for which the Level A Service Limits are applicable.

The Design Pressure includes allowances for pressure surges.

Design Temperature - Except as otherwise defined in ASME BPVC, Division 1, NB-3112 for Class 1 components, the specified Design Temperature is not less than the expected maximum mean metal temperature through the thickness of the part considered for which Level A Limits are specified.

Design Mechanical Loads - The specified Design Mechanical Loads are in accordance with NCA-2142.1C.

3.6.3.6.4 Test Loadings

Test Pressure - The specified internal and external test pressures are as required by the ASME BPVC, Section III, Division 1.

Test Loads - Loads due to other types of required tests are included as required by the ASME BPVC, Section III, Division 1.

Test Temperature - Test temperature is defined to ensure that thermal effects are considered in test loads.

3.6.3.7 Loading Phenomena

Section 3.6.3.7 describes the types of load phenomena, that is considered for components, as applicable.

3.6.3.7.1 Flow Induced Vibration

Flow of fluids past objects creates local pressure disturbances, which exert forces on the object. These forces can cause dynamic responses depending on the forcing function and dynamic characteristics of the object. Flow induced vibrations have been noted in nuclear power plant systems, which produce vortex shedding (e.g., heat exchangers), pump (reciprocating or centrifugal), and thermodynamic instability conditions. Design changes are reviewed for potential FIV mechanisms, evaluating all modes of system operation including both normal and abnormal conditions. Requirements for vibration monitoring are not within the scope of this document.

FIV loads may be associated with Service Level A for those structures (e.g., reactor internals) where the loads exist during normal operation. For FIV loads associated with transients that are not considered part of normal operation, the FIV loads are evaluated as part of the alternative service level.

Vortex Shedding

Vortex shedding occurs at certain fluid velocities when a fluid flows past objects. The dynamic response is controlled by proper spacing of the support plates for the tube bundle. The vibration cannot be eliminated but it can usually be controlled. It is important that these cases consider all potential modes of component operation. Vortex shedding hydrodynamic mass effects are

considered. Other components susceptible to flow induced vibration are pressure, flow, and temperature sensors, which encroach upon the flow stream.

Pressure Fluctuations

Pressure fluctuations in a vapor or gas-state fluid (e.g., steam) occur due to flow past branch piping connections and branch connected components (e.g., safety valve “bell chamber” resonance), flow through short radius elbow fittings that induce flow separation effects, flow passing through valve chambers, flow past sharp-edged in-line pipe components (e.g., orifices, weld joint backing rings, valve seat rings), or two or more individual flows entering a common header or drum that generates an acoustic response. These various flow disturbances generate acoustic waves that can travel forward and backward in a piping system. If of sufficient strength and at a component’s susceptible frequency, these acoustic resonances can cause cyclic fatigue and result in component failure.

Pumps create pressure fluctuations in a fluid system. In most system designs, these fluctuations are insignificant. However, the possibility exists that these fluctuations, coupled with unintentional but improper system or component structural characteristics, can cause resonant vibrational response in the system or component. Component structural characteristics are designed to assure a resonance value sufficiently high to avoid excitation by evaluated system fluid fluctuations. Pressure attenuation devices are used as applicable to significantly reduce the effects of this phenomenon.

Thermodynamic Instability

Under certain system design features and operating modes, fluid dynamic forces can be generated, which create large pressure variations. These have been noted in certain feedwater systems where a relatively cold fluid layer is in contact with a relatively hot steam region; under certain operating modes significant water-hammer-type phenomena have occurred causing a breach of the pressure-retaining boundary.

3.6.3.7.2 Rapid Valve Closure or Opening

Extremely rapid valve closure or opening in a fluid system can create large pressure waves which can propagate through a piping system and into connected components. This rapid motion could be caused by operating characteristics of the valve (e.g., stiffness of diaphragm in pneumatic operators), the fluid flow forces acting on the valve parts during all modes of operations.

For example, TSV closure has been identified as being capable of generating large pressure waves which could cause significant dynamic response. Prior to TSV closure, saturated steam flows through main steam piping at nuclear boiler rated pressure and mass rate. Steam flow to the turbine comes to a stop at the instant the turbine stop valve closes. The flow of steam travels in the main steam line through the vessel nozzle and into the vessel. This results in a compressive acoustic load on steam dryer outer hood, as well as steam impingement load on steam dryer outer hood. Additionally, repeated reflections of the compression wave in the main steam line generate time-varying forces in the main steam piping. System, components, and structures in the Reactor Building, Steam Tunnel and Turbine Building may be affected.

3.6.3.7.3 Isolation Condenser Operation

The thermal effects associated with operation of ICS and the loads such as pressure resulting from operation of ICS are considered. Loads associated with the breaks of ICS high pressure lines in the pool are considered. The major loads imposed on ICS result from:

- Sudden reactor isolation at power operating conditions
- During station blackout (i.e., unavailability of all alternate current power)

- Failure to Scram
- LOCA

3.6.3.7.4 Failures of High-Energy Fluid System Piping

The effects of postulated pipe breaks in high-energy fluid systems as well as measures used to protect SSCs are defined in Subsection 3.4.4.

3.6.3.7.5 Failures of Moderate-Energy Fluid System Piping

The effects of postulated pipe cracks in moderate-energy fluid systems as well as measures used to protect SSCs are defined in Subsection 3.4.4.

3.6.3.7.6 Fuel Lift Loads

Fuel lift is the postulated process under which a combination of vertical motion of the RPV support, scram uplift forces on the fuel assemblies and vertical hydraulic forces result in fuel assemblies lifting off from their seating surfaces on the fuel support. The reaction load of the fuel on the core support structures is considered.

3.6.3.8 Safety Class Functional Criteria

For any normal or off-normal design condition event, safety class equipment and piping can accomplish the safety class functions as required by the event and incurring no permanent changes that could deteriorate the ability to accomplish safety class functions as required by any subsequent design-condition event.

For any emergency or faulted design-condition event, safety class equipment, and piping are capable of accomplishing their safety class functions as required by the event, but repairs could be required to ensure their ability to accomplish safety class functions as required by any subsequent design-condition event.

3.6.3.9 Reactor Pressure Vessel Assembly

The reactor vessel assembly includes: the RPV pressure boundary out to and including the nozzles, the RIV's, and the housings for FMCRD and nuclear instrumentations. The RPV assembly is an ASME BPVC Section III, Division 1, Class 1.

The feedwater nozzle design does not allow incoming feedwater flow to have direct contact with the nozzle bore region. A double thermal sleeve design provides protection against thermal cycling on the nozzle bore. The ICS Condensate Return nozzles use a similar single thermal sleeve design to mitigate thermal cycling of the nozzle bore during initial IC train operation when accumulated condensate is draining.

The stress analysis is performed on the RPV for various plant operating conditions (including faulted conditions) by using elastic methods, except as noted in Subsection 3.6.1.4.3. Loading conditions, design stress limits, and methods of stress analysis for the core support structures and other reactor internals are provided in Table 3.6-2.

The RPV internals are classified in Chapter 3, Section 3.2, and Appendix 3A. Complete stress reports on these components are prepared in accordance with the ASME BPVC Section III, Division 1, requirements.

3.6.3.10 Main Steam Piping

The MS piping trains extending from the outboard MSRIV to and including Seismic Interface Restraints (SIR) that are outboard of the MSCIVs are designed and constructed in accordance with the ASME BPVC Section III Division 1 rules for Class 2 Nuclear Components. Stresses are

calculated on an elastic basis for each service level and evaluated in accordance with NCD-3600 of the ASME BPVC Section III Division 1. Table 3.6-11 shows the specific load combinations and acceptance criteria for Class 2 piping that apply to this piping.

The MSCIVs, are designed and constructed in accordance with the ASME BPVC III Division 1, NCD-3500 requirements for Class 2 components.

The MS system piping extending from the outboard SIR to the turbine stop valve is constructed in accordance with the ASME B31.1 Criteria.

3.6.3.11 Other Components

3.6.3.11.1 Isolation Condenser System (ICS) Condenser and Piping

The ICS piping inside the primary containment between the RPV and the Isolation Condenser Heat Exchanger is designed and constructed in accordance with the ASME BPVC Section III Division 1 requirements for Class 1 piping. The isolation condenser and piping outside containment are designed and constructed in accordance with ASME BPVC Section III Division 1 Class 2 requirements.

3.6.3.11.2 CUW System Heat Exchangers

The CUW heat exchangers (regenerative) are not part of a safety system. However, the heat exchangers are Seismic Category NS equipment. The ASME BPVC Section III Division 1 requirements for Class 3 components are used in the design and construction of the CUW System heat exchanger components.

3.6.3.11.3 SDC System Pump and Heat Exchangers

The SDC heat exchangers (nonregenerative) are not part of a safety system. However, the pumps and heat exchangers are Seismic Category NS equipment respectively. The ASME BPVC Section III Division 1 requirements for Class 3 components are used in the design and construction of the SDC System pump and heat exchanger components.

3.6.3.11.4 ASME BPVC Section III Division 1, Class 2 and 3 Vessels

ASME BPVC Section III Division 1, Class 2 and 3 vessels are constructed in accordance with the ASME BPVC Section III Division 1. The analysis of these vessels is performed using elastic methods.

3.6.3.11.5 ASME BPVC Section III Division 1, Class 1, 2 and 3 Valves

ASME BPVC Section III Division 1, Class 1, 2, and 3 valves are constructed in accordance with the ASME BPVC Section III Division 1.

All valves and their extended structures are designed to withstand the accelerations due to seismic and other RBV loads. The analysis of these valves is performed using elastic methods. Refer to Subsection 3.6.3.9 for additional information on valve operability.

3.6.3.11.6 ASME BPVC III Division 1, Class 1, 2 and 3 Piping

ASME BPVC Section III Division 1, Class 1, 2 and 3 piping is constructed in accordance with the ASME BPVC Section III Division 1. For ASME BPVC Section III Division 1, Class 1 piping, stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 of the ASME BPVC Section III Division 1, and fatigue usage is determined. For ASME BPVC Section III Division 1, Class 2 and 3 piping, stresses are calculated on an elastic basis and evaluated in accordance with NCD-3600 of the ASME BPVC Section III Division 1. If a NB-3600 analysis is performed for ASME BPVC Section III Division 1, Class 2 or 3 pipe, all analyses required for ASME BPVC Section III Division 1, Class 1 pipe as specified in this document and the ASME

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BPVC is performed. Tables 3.6.10 and 3.6.11 shows the specific load combinations and acceptance criteria for ASME BPVC Section III Division 1, Class 1, 2, and 3 piping systems.

3.6.3.12 Valve Operability Assurance

This subsection discusses operability assurance of active ASME BPVC Section III Division 1 valves, including actuators (Refer to Subsection 3.9.6.2).

Valves that perform an active Safety Category 1 function are functionally qualified to perform their required functions. For valve designs developed for the BWRX-300 that were not previously qualified, the qualification programs meet the requirements of ASME QME-1 (For valve designs previously qualified to standards other than ASME QME-1), the following approach is used:

1. Qualification specifications (e.g., design specifications) consistent with Appendices QV-I and QV-A of QME-1 are prepared to ensure the operating conditions and safety class functions for which the valves are to be qualified are communicated to the manufacturer or qualification facility.
2. Suppliers are required to submit, for review and approval, application reports, as described in QME-1, that describe the basis for the application of specific predictive methods and/or qualification test data to a valve application.
3. The application reports provided by the suppliers are reviewed for adherence to specification requirements to ensure the methods used are applicable and justified and to verify any extrapolation techniques used are justified. A gap analysis is performed to identify any deviations from QME-1 in the valve qualification. Each deviation is evaluated for impact on the overall valve qualification. If the conclusion of the gap analysis is that the valve qualification is inadequate, then the valve may be qualified using a test-based methodology, as allowed by QME-1.

Functional qualification addresses key lessons learned from industry efforts, particularly on air- and motor-operated valves, many of which are discussed in Section QVG of QME-1. For example:

1. Evaluation of valve performance is based on a combination of testing and analysis, using design similarity to apply test results to specific valve designs.
2. Testing to verify proper valve setup and acceptable operating margin is performed using diagnostic equipment to measure stem thrust and torque, as appropriate.
3. Sliding friction coefficients used to evaluate valve performance (e.g., disk-to-seat friction coefficients for gate valves and bearing coefficients for butterfly valves) account for the effects of temperature, cycle history, load, and internal parts geometry.
4. Actuator sizing allows margin for aging/degradation, test equipment accuracy and other uncertainties, as appropriate.
5. Material combinations that may be susceptible to galling or other damage mechanisms under certain conditions are not used.

Subsection 3.9 provide details on the seismic qualification of valves and on the Environmental Qualification of valves.

The major safety class active valves are the RIVs, Condensate Return Valves and CIVs. These valves are designed to meet the ASME BPVC Section III Division 1 BPVC requirements and perform their mechanical motion in conjunction with a dynamic (SSE and other RBV) load event. The dynamic qualification for operability is unique for each valve type; therefore, each method of qualification is provided individually below.

3.6.3.13 Main Steam Containment Isolation Valves

The MSCIVs are evaluated by analysis and test for capability to operate under the design loads that envelop the predicted loads during a Design Basis Accident (DBA) and DBE.

3.6.3.14 Other Active Valves

Other safety class active valves are ASME BPVC Section III Division 1 Class 1, 2 or 3 and are designed to perform their mechanical motion during dynamic loading conditions. The operability assurance program ensures that these valves operate during a dynamic seismic and other RBV event.

3.6.3.14.1 Procedures

Qualification tests accompanied by analyses are conducted for all active valves. Procedures for qualifying electrical and instrumentation components, which are depended upon to cause the valve to accomplish its intended function, are developed to assure these functions are accomplished.

3.6.3.14.2 Tests

Prior to installation of the SC1 valves, the following tests are performed at the factory facility as required in the field:

- Shell hydrostatic test to the ASME BPVC Section III Division 1 requirements
- Seat leakage tests
- Obturator hydrostatic test
- Functional tests to verify that the valve opens and closes within the specified time limits when subject to the design differential pressure

The results of all required tests are properly documented and included as a part of the operability acceptance documentation package.

3.6.3.14.3 Check Valves

Due to the simple characteristics of the check valves, the active check valves are qualified by a combination of the following tests and analysis:

- Stress analysis including the dynamic loads where applicable
- In-shop hydrostatic tests
- In-shop seat leakage test

3.6.3.15 Qualification of Electrical and Instrumentation Components Controlling Valve Actuation

A practical problem arises in attempting to describe tests for simple devices (e.g., relays, motors, sensors, etc.) as well as for complex assemblies such as control panels. It is reasonable to assume that a simple device, that is an integral part of an assembly, may be subjected to the same dynamic load tests while in an operating condition. Thus, the performance of a simple device may be monitored during the test. However, for complex panels, such a test is not always practical. In this situation, the following alternate approach may be followed.

The individual devices are tested separately in an operating condition and the test levels recorded as the qualification levels of the devices. The panel, with similar but inoperative devices installed, is vibration tested to determine if the panel response accelerations. Installing the non-operating devices assures that the test panel has representative structural characteristics of a production

panel. The accelerations are measured by accelerometers installed at the device attachment locations. The accelerations are less than the levels at which the devices were qualified. If the acceleration levels at all the device locations are found to be less than the levels to which the devices are qualified, then the total assembly is considered qualified. Otherwise, either the panel is redesigned to reduce the acceleration level to the device locations and retested, or the devices are requalified to the higher levels.

3.6.3.16 Design of Pressure Relief Devices

The NBS system does not utilize safety or relief valves for overpressure relief. During normal operation, the mainstream flow to the turbine is throttled to control system pressure. Chapter 6, Section 6.2 describes the method of overpressure relief.

3.6.3.17 Component Supports

The establishment of the design/service loadings and limits is in accordance with the ASME Section III, Division 1, Article NCA-2000 and Subsection NF. These loadings and stress limits apply to the structural integrity of components and supports when subjected to combinations of loadings derived from plant and system operating conditions and postulated plant events. The combination of loadings and stress limits are included in the Design Specification of each component and support.

ASME Section III component supports are designed, manufactured, installed, and tested in accordance with all applicable codes and standards. Supports include hangers, snubbers, struts, spring hangers, frames, energy absorbers and limit stops, Pipe whip restraints are not considered as pipe supports.

The design of bolts for component supports is specified in the ASME BPVC III Division 1, Subsection NF. Stress limits for bolts are given in NF-3225. The rules and stress limits which must be satisfied are those given in NF-3324.6 multiplied by the appropriate stress limit factor for the particular service loading level and stress category specified in Table NF-3225.2-1.

3.6.3.18 Piping Supports

Supports and their attachments for ASME BPVC Section III Division 1 Class 1, 2, and 3 piping are designed in accordance with Subsection NF up to the interface of the building structure, with jurisdictional boundaries as defined by Subsection NF. The building structure component supports (connecting the NF support boundary component to the existing building structure) are designed as specified in Section 3.5.

The design of supports for the non-nuclear piping satisfies the requirements of ASME B31.1 Power Piping Code, Paragraphs 120 and.

3.6.3.19 Reactor Pressure Vessel Stabilizer

The RPV stabilizer is designed as a SC1 linear type component support in accordance with the requirements of ASME BPVC Section III Division 1 Subsection NF. The stabilizer provides a reaction point near the upper end and lower end of the RPV to resist horizontal loads caused by effects such as earthquake, pipe rupture, and RBV. The design loading conditions, and stress criteria and the calculated stresses will meet the ASME BPVC Section III Division 1 allowable stresses in the critical support areas for various plant operating conditions.

3.6.3.20 Floor-Mounted Major Equipment

The condenser modules in the Isolation Condenser System (ICS) are analyzed to verify the adequacy of their support structure under various plant operating conditions. The analysis applies

the maximum shear, moment, and accelerations calculated from the seismic response analysis for the Reactor Building at the attachment locations on the pool floor for the ICS.

In the ICS module analysis, no credit is taken for damping effects of the pool water. Additionally, the mass of the condensers is increased by an amount equivalent to the weight of water they displace. This conservative factor accounts for the hydrodynamic effects that include impulsive loads and convective loads (sloshing of the pool water).

In all cases, the load stresses in the critical support areas of the ICS modules are maintained within ASME BPVC Section III Division 1 allowable.

3.6.3.21 Other ASME BPVC Component Supports

The ASME BPVC Section III Division 1 component supports and their attachments (other than those discussed in the preceding subsection) are designed in accordance with ASME BPVC Section III Division 1, Subsection NF up to the interface with the building structure. The loading combinations for the various operating conditions correspond to those used to design the supported component. The component loading combinations are discussed in Table 3.6-2. Active component supports are discussed in Subsection 3.6.3.18. The stress limits are per ASME BPVC Section III Division 1, Subsection NF, and NB-3600 and NCD-3600. The supports are evaluated for buckling in accordance with ASME BPVC Section III Division 1.

3.6.4 Control Rod Drive System

The CRD system consists of mechanical components that provide the means for movement of the control rods. The CRD system provides one of the independent reactivity control systems. The control rods and the drive mechanisms are capable of reliably controlling reactivity changes either under conditions of AOOs, or under DBA conditions. A positive means for inserting the rods is always maintained to ensure appropriate margin for malfunction, such as stuck rods. Because the CRD system is a safety class system and portions of the CRD system are a part of the RCPB, the system is designed, fabricated, and tested to quality standards commensurate with the safety class functions to be performed. This provides an extremely high probability of accomplishing the safety class functions either in the event of AOOs or in withstanding the effects of DBAs and natural phenomena such as earthquakes.

The CRD system includes the FMCRD mechanisms, the HCU assemblies, and the CRD hydraulic system. The system extends inside the RPV to the coupling interface with the control rod blades.

3.6.4.1 Descriptive Information on Control Rod Drive System

Descriptive information on the FMCRDs as well as the entire CRD system is contained in Chapter 4, Subsection 4.6.

3.6.4.2 Applicable Control Rod Drive System Design Specification

The CRD system, which is designed to meet the functional design criteria outlined in Chapter 4, Subsection 4.6.1, consists of the following:

- Electro-hydraulic fine motion control rod drive
- Hydraulic Control Unit (HCU)
- Hydraulic pumps
- Electric power supply R20 system to the FMCRD motors – CRD Boundary is at the motor
- Interconnecting piping

- Flow control valves
- Instrumentation

Those components of the CRD system forming part of the primary pressure boundary are designed according to ASME BPVC Section III Division 1 BPVC, Class 1 requirements.

The quality group classification of the components of the CRD system is outlined in Appendix 3A and are designed to the codes and standards in accordance with their individual quality groups.

Pertinent aspects of the design and qualification of the CRD system components are discussed in the following locations: transients in Chapter 3, Subsections 3.6.1.1, 3.6.3.6 and 3.6.3.7, faulted conditions in Chapter 3, Subsections 3.6.1.4.1 and 3.6.1.4.2, and seismic testing in Chapter 4, Subsections 4.6.1 and 4.6.2.

3.6.4.3 Design Loads and Stress Limits

3.6.4.3.1 Allowable Deformations

The ASME BPVC Section III Division 1, Subsection NB components of the CRD system are evaluated analytically and the design loading conditions, and stress criteria are as given in Table 3.6-2.

3.6.5 Reactor Pressure Vessel Internals

Reactor pressure vessel internals are described in Chapter 5, Section 5.4.

3.6.6 Functional Design, Qualification and In-service Testing Programs for Pumps, Valves, and Dynamic Restraints

Chapter 3, Section 3.9, Equipment Qualification provides the methodology for qualification of Pumps and Valves. The qualification involves both determining component functionality while maintaining structural integrity. Seismic testing of components is performed as well as use of analytical methods.

Chapter 3, Subsection 3.6.3.17 discusses methodology for qualification of dynamic restraints.

In-service Testing Programs are developed for required operability and functional tests for components as described in Chapter 3, Subsection 3.10.3.

3.6.7 Piping Design

The design of safety class piping systems, piping components and pipe supports is based on the code rules established under the ASME BPVC Section III, Division 1 code for Class 1, Class 2, and Class 3 nuclear piping, components and supports. For non-ASME Code class components, ASME B31.1 power piping, and ASME B31.3 process piping codes are used. Safety classifications of safety, seismic categories, and quality groups for piping SSCs are established within the system chapters. The simplified schematic diagrams within the system chapters identify the system safety class, seismic class, and quality boundaries. The functional, operational, and safety requirements are unique to each system and the required loading conditions are applied as specified in the specific ASME Code class sections.

3.6.7.1 ASME Class 1 Piping Design Rules and Analysis

ASME Class 1 piping design conforms to the requirements of ASME BPVC Section III Division 1 Paragraph NB code rules that covers both piping and piping components. The pipe supports attached to the ASME Class 1 piping meet the appropriate requirements of ASME BPVC Section III Division 1, Paragraph NF. The anchor sleeve of the containment structure penetrations meets the requirements of ASME BPVC Section III Division 1, Paragraph NE (Reference 3.6-6).

3.6.7.1.1 Overpressure Protection

The details and certification of Overpressure Protection design for each piping system are in the System Overpressure Protection Reports.

3.6.7.1.2 Boundaries

The boundaries of the Class 1 piping in each system are outlined in the system Piping and Instrumentation Diagrams (P&IDs).

Support design jurisdictional boundaries at interfaces between piping and structure by intervening elements that are defined per ASME BPVC Section III Division 1 - Subsection NF – Supports, Subarticle NF-1130. If piping supports transmit loads to surface-mounted baseplates as discussed in Subparagraph NF-1132(d), the baseplates are within the building structure jurisdiction.

Where ASME BPVC Section III Division 1 Class 2 piping is connected to ASME BPVC Section III Division 1 Class 1 piping, the rules for expansion and flexibility for A ASME BPVC Section III Division 1 Class 1 piping applies out to the first anchor in the ASME BPVC Section III Division 1 Class 2 piping system. However, the resulting solution of forces and moments are used to evaluate stresses in accordance with the allowable criterion of ASME BPVC Section III Division 1 Subarticle NCD-3650.

3.6.7.1.3 Classifications

Code Classification

Piping that is classified as Quality Group A meets the requirements for ASME BPVC III Division 1 Class 1 components provided in ASME BPVC Section III Sub Article NB-3600.

The pipe supports attached to Quality Group A piping meet the appropriate requirements of ASME BPVC Section III Paragraph NF.

Seismic Classification

Seismic categories are to be in accordance with those listed on the system P&ID.

Energy (High/Moderate) Classification

Piping is classified as High or Moderate-Energy for use in pipe failure postulation. Refer to Section 3.4.4.2 for further explanation.

3.6.7.1.4 Material Requirements

The material properties used in Class 1 analyses is in accordance with ASME BPVC Section II – Materials – Part D – Properties (Metric).

Examination and Repair

The examination and repair of all Class 1 materials and welds is performed using the methods and acceptance standards as specified in ASME BPVC Section III Subarticle NB-2500.

In-service inspection requirements for Class 2 and 3 piping and components are defined in Subsection 3.10.5.

Fracture Toughness Requirements

Pressure-retaining ferritic material, and material welded thereto are impacted tested in accordance with the requirements of NB-2300 and NB-2400 to ensure adequate fracture toughness properties.

3.6.7.1.5 Design Conditions

Design Service Life

The design service life of the BWRX-300 Nuclear Power Plant is 60 operational years. Additional time in-service for startup and decommissioning activities is included as applicable.

Design Pressure and Temperatures

The design pressures and temperatures of each piping system are identified in the respective system design documentation.

Design Duty Cycles

The pressure-temperature duty cycles to be used in the fatigue analysis are specified in the respective system Pressure-Temperature Duty Cycle drawings. Assumptions regarding the pressure and temperature cycles used to determine allowable stress reduction factors or any other analysis input are included in the design report with a basis of 60 years design life.

Environmental Conditions

All SC1 piping, and components, are capable of performing their safety class functions when exposed to specified environmental conditions specified in the Environmental Qualification Envelope. Piping system active components are environmentally qualified as specified in Subsections 3.9.3 and 3.9.4.

3.6.7.1.6 Test Loads

The only test loads on the piping system are due to hydrostatic testing. The loads due to hydrostatic testing are in accordance with NB-6000.

3.6.7.1.7 Static Loads

Pressure

The design pressure and operating pressure for each system/component are as specified in the respective system design documentation.

Weight

The weight of the piping system includes the weight of the pipe, in-line components, fluid contents, and insulation, as applicable. In addition, the weight of support components attached to the pipe are considered.

Support systems for piping that normally carries steam but will be filled with water during a hydrostatic test and/or refueling outage are designed to accommodate the increased weight.

Thermal Expansion

The analysis of thermal expansion includes all thermal operating modes, environmental conditions, cold water modes, and thermal attenuation.

Sufficient thermal expansion cases shall be established to account for various operating conditions and for calculating the range of thermal expansion stresses between all pairs of load sets.

The installation temperature for the piping systems is defined as a temperature of 21° C for Class 1 piping unless basis is provided to use a higher temperature. The ambient state shall be included as an analysis load set with defined cycles.

Applicable equipment nozzle movements are considered for their effect with respect to each operating mode.

Support movements due to thermal expansion are included in the design.

Thermal Attenuation/Stratification

Thermal attenuation/stratification are considered in the design whenever fluids at different temperatures mix.

On run/branch connections where there is a closed valve and the resulting "dead leg" temperature tends toward ambient, the temperature distribution in the run/branch line are considered and properly included in the thermal expansion analysis.

3.6.7.1.8 Dynamic Loads

Dynamic loads include both the inertial effect and support displacements (i.e., anchor movement). Categories of loads and load conditions considered include (but are not limited to) the following:

- Seismic
- Loss-of-Coolant Accident Loads
- Reactor Pressure Vessel and Containment Isolation Valve Transients
- Thermal Stratification

3.6.7.1.9 Plant Events and Load Combinations

Plant states are based on expected frequency of occurrence of Postulated Initiating Events (PIEs) which are the plant events that lead to deviations from normal operation (AOOs, DBAs or DECAs depending on the additional failures that occur) and are related to ASME service levels as shown in Table 3.6-3.

Load combinations and acceptance criteria for the BWRX-300 Class 1 piping are provided in Table 3.6-10.

3.6.7.1.10 Analytical Computer Codes Used for Piping Stress, Component Stress, and Support Structural Qualifications

Chapter 3, Appendix 3C provides a listing of and description of applicable safety computer codes used for qualification of piping, mechanical components, and pipe supports.

3.6.7.1.11 Analysis Methodology and Stress Reports

Piping system stresses shall be calculated on an elastic basis for each service level.

For ASME BPVC Section III Division 1 Class 1 piping systems and components, stress reports are prepared in accordance with ASME BPVC Section III Division 1 Class 1 requirements and include applicable equipment qualification reports for active components.

3.6.7.2 ASME BPVC Section III Division 1 Class 2/3 Piping Design Rules and Analysis

ASME BPVC Section III Division 1 Class 2/3 piping design conforms to the requirements of ASME BPVC Section III Division 1 Subsection NCD that covers both piping and piping components. Load combinations and acceptance criteria for the BWRX-300 Class 2 piping are provided in Table 3.6-11.

The containment penetration sleeve of ASME Class 2 piping is an anchor for the piping. The sleeve of the containment structure penetrations meets the requirements of ASME BPVC Section III Division 1, Subsection NE (Reference 3.6-6).

3.6.7.2.1 *Overpressure Protection*

The details and certification of Overpressure Protection design for each piping system are in the System Overpressure Protection Reports.

3.6.7.2.2 *Boundaries*

The boundaries of the Class 2 and 3 piping in each system are outlined in the system P&IDs and simplified diagrams shown in the system PSAR chapters.

Support design jurisdictional boundaries at interfaces with piping, structure, or intervening elements are defined in ASME BPVC Section III Division 1, Subsection NF-1130. If piping supports transmit loads to surface-mounted baseplates as discussed in Subsection NF-1132(d), the baseplates are within the building structure jurisdiction.

3.6.7.2.3 *Classifications*

Code Classifications

Detailed classifications of pipe and components are defined in the system design documents. Piping that is classified as ASME BPVC Section III Division 1 Class 2 or ASME BPVC Section III Division 1 Class 3 meet the requirements for ASME BPVC Section III Division 1 Class 2 and 3 components provided in Subsection NCD-3600 of the ASME Code.

Where ASME BPVC Section III Division 1 Class 2 piping is connected to ASME BPVC Section III Division 1 Class 1 piping, the rules for expansion and flexibility for Class 1 piping apply out to the first anchor in the ASME BPVC Section III Division 1 Class 2 piping system. However, the resulting solution of forces and moments are used to evaluate stresses in accordance with the allowable criterion of NCD-3650.

The pipe supports attached to the ASME BPVC Section III Division 1 Class 2 and 3 piping meet the appropriate requirements of Subsection NF of the ASME Code.

Seismic Classification

Seismic categories are to be in accordance with those listed on the system design documents.

Energy (High/Moderate) Classification

Piping is classified as High or Moderate-Energy for use in pipe failure postulation. Refer to Subsection 3.4.4.2 for further explanation.

3.6.7.2.4 *Materials*

Material Specifications

The material properties used in Class 2 or 3 analyses are in accordance with ASME BPVC Section II – Materials – Part D – Properties (Metric) (Reference 3.6-1).

Examination and Repair

The examination and repair of all Class 2 and 3 materials and welds are performed using the methods and acceptance standards as specified in NCD-2500.

In-service inspection requirements for Class 2 and 3 piping and components are defined in Subsection 3.10.5.

Fracture Toughness Requirements

Pressure-retaining ferritic material, and material welded thereto are impact tested in accordance with the requirements of NCD-2300 and NCD-2400 to ensure adequate fracture toughness properties.

3.6.7.2.5 *Design Conditions*

Design Service Life

The design service life of the BWRX-300 Nuclear Power Plant is 60 operational years plus any additional time in-service for startup and decommissioning activities as applicable.

Design Pressures and Temperatures

The design pressures and temperatures of each piping system are identified in the respective system design documents.

Design Duty Cycles

Assumptions regarding the pressure and temperature cycles used to determine allowable stress reduction factors or any other analysis input are included in the design report with a basis of 60 years design life.

Environmental Conditions

All SC1 piping, and components, are capable of performing their Safety Category functions when exposed to the environmental conditions.

3.6.7.2.6 *Design Input Loads*

Test Loads

The only test loads on the piping system are due to hydrostatic testing. The loads due to hydrostatic testing are in accordance with NCD-6000.

Static Loads

Pressure

The design pressure and operating pressure for each system/component are as specified in the respective System Line list.

Weight

The weight of the piping system includes the weight of the pipe, in-line components, fluid contents, and insulation, as applicable. In addition, the weight of support components attached to the pipe are considered.

Support systems for piping that normally carries steam but will be filled with water during a hydrostatic test and/or refueling outage are designed to accommodate the increased weight.

Thermal Expansion

The analysis of thermal expansion includes all thermal operating modes, environmental conditions, cold water modes, and thermal attenuation.

Sufficient thermal expansion cases are established to account for various operating conditions to determine the maximum range of thermal expansion stresses.

The installation temperature for the piping systems is defined as a temperature of 21 °C for Class 2 and 3 piping.

Applicable equipment nozzle movements are considered for their effect with respect to each operating mode.

Support movements due to thermal expansion are included in the design. Thermal anchor movements of less than 1.6 mm are considered negligible and do not need to be considered in the analysis.

Thermal Attenuation/Stratification

Thermal attenuation/stratification are considered in the design whenever fluids at different temperatures mix.

On run/branch connections where there is a closed valve and the resulting "dead leg" temperature tends toward ambient, the temperature distribution in the run/branch line are considered and properly included in the thermal expansion analysis.

3.6.7.2.7 Dynamic Loads

Dynamic loads include both the inertial effect and support displacements (i.e., anchor movement).

Categories of loads and load conditions considered include (but are not limited to) the following:

- Seismic
- Loss-of-Coolant Accident Loads
- Turbine Stop Valve Closure
- Reactor Pressure Vessel and Containment Isolation Valve Transients

3.6.7.2.8 Plant Events and Load Combinations

Plant states are based on expected frequency of occurrence of Postulated Initiating Events (PIEs) which are the plant events that lead to deviations from normal operation (AOOs, DBAs or DECAs depending on the additional failures that occur) and are related to ASME service levels as shown in Table 3.6-3.

Load Combinations

The load combinations and acceptance criteria in Table 3.6-11 are applicable to all ASME BPVC Section III Division 1 Class 2 and 3 piping systems, structures, and components.

Load Combinations for Piping and Components

The load combinations and acceptance criteria in Table 3.6-11 are applied to the analysis of ASME BPVC Section III Division 1 Class 2 and 3 piping systems and components.

3.6.7.3 ASME B31.1 Piping Design Rules and Analysis

Non-Safety class power piping conforms to ASME B31.1 code.

Load combinations and acceptance criteria for the BWRX-300 Class 1 piping are provided in Table 3.6-12.

Each Non-Safety class power piping systems includes the piping, pipe supports, penetrations, and welds joining the piping to adjacent components within the prescribed boundaries.

Descriptions of systems that contain ASME B31.1 piping and components including their functions are described in the system chapters.

3.6.7.3.1 Overpressure Protection

The details and certification of overpressure protection design for each piping system are in the System Overpressure Protection Reports.

3.6.7.3.2 Boundaries

The boundaries of the ASME B31.1 piping in each system are outlined in the respective system P&ID and indicated in the simplified diagrams provided in each chapter.

3.6.7.3.3 Classifications

Code Classification

Detailed classifications of pipe and components are defined in the system design documents.

Portions of the ASME BPVC Section III Division 1 Class 2 or 3 piping system analysis may contain ASME B31.1 piping beyond a normally closed valve which may define the boundary out to the first anchor in the ASME B31.1 piping system.

The pipe supports attached to the ASME B31.1 piping meet the appropriate requirements of ASME B31.1.

Seismic Classification

Seismic categories are to be in accordance with those listed on the system design documents.

Energy (High/Moderate) Classification

Piping is classified as High or Moderate-Energy for use in pipe failure postulation. Refer to Subsection 3.4.4.2 for further explanation.

3.6.7.3.4 Materials

Material Specifications

The material properties used in ASME B31.1 system analysis are in accordance with ASME B31.1.

Examination and Repair

The examination and repair of all ASME B31.1 piping materials and welds are performed using the methods and acceptance standards as specified in ASME B31.

The recommended practice for operation, maintenance, and modification of ASME B31.1 piping, and components is in accordance with the applicable local jurisdiction standard and code.

Fracture Toughness Requirements

The requirements of ASME B31T, *Standard Toughness Requirements for Piping*, Paragraphs 3, 4, and Appendix A are met.

3.6.7.3.5 Design Conditions

Design Service Life

The design service life of the BWRX-300 Nuclear Power Plant is 60 operational years plus any additional time in-service for startup and decommissioning activities as applicable.

Design Pressures and Temperatures

The design pressures and temperatures of each piping system are identified in the respective system design documents.

Design Duty Cycles

Assumptions regarding the pressure and temperature cycles used to determine allowable stress reduction factors or any other analysis input are included in the design report with a basis of 60 years design life.

Environmental Conditions

Consideration of environmental conditions for functional qualification is not applicable to ASME B31.1 piping systems.

Recommended practices related to the protection of piping systems against detrimental effects of environmental conditions are provided in ASME B31.1 Appendices IV and V.

3.6.7.3.6 Design Input Loads

Test Loads

The only test loads on the piping system are due to hydrostatic testing.

Static Loads

Pressure

The design pressure and operating pressure for each system/component are as specified in the respective Process Flow Diagram.

Weight

The weight of the piping system includes the weight of the pipe, in-line components, fluid contents, and insulation, as applicable. In addition, the weight of support components attached to the pipe is considered.

Support systems for piping that normally carries steam but will be filled with water during a hydrostatic test and refueling outage is designed to accommodate the increased weight.

Thermal Expansion

The analysis of thermal expansion includes all thermal operating modes, environmental conditions, cold water modes, and thermal attenuation.

Sufficient thermal expansion cases are established to account for various operating conditions to determine the maximum range of thermal expansion stresses.

The installation temperature for the piping systems is defined as a temperature of 21° C for non-nuclear (ASME B31.1) piping.

Applicable equipment nozzle movements are considered for their effect with respect to each operating mode.

Support movements due to thermal expansion are included in the design. Thermal anchor movements of less than 1.6 mm are considered negligible and do not need to be considered in the analysis.

Thermal Attenuation/Stratification

Thermal attenuation/stratification are considered in the design whenever fluids at different temperatures mix.

On run/branch connections where there is a closed valve and the resulting "dead leg" temperature tends toward ambient, the temperature distribution in the run/branch line are considered and properly included in the thermal expansion analysis.

3.6.7.3.7 Dynamic Loads

Dynamic loads include both the inertial effect and support displacements (i.e., anchor movement). Categories of loads and load conditions considered include (but are not limited to) the following:

- Seismic
- Turbine Stop Valve Closure
- Reactor Pressure Vessel and Containment Isolation Valve Transients

3.6.7.3.8 *Plant Events and Load Combinations*

Plant states are based on expected frequency of occurrence of Postulated Initiating Events (PIEs) which are the plant events that lead to deviations from normal operation (AOOs, DBAs or DECAs depending on the additional failures that occur) and are related to ASME service levels as shown in Table 3.6-3.

Load Combinations

The load combinations and acceptance criteria presented in this specification are applicable to all ASME B31.1 piping systems, structures, and components within the scope of this document.

Load Combinations for Piping and Components

The load combinations and acceptance criteria in Table 3.6-12 are applied to the analysis of ASME B31.1 piping systems and components.

3.6.8 Threaded Fasteners – Codes for ASME BPVC Section III Division 1 Class 1, 2, and 3

3.6.8.1 Material Selection

Material used for threaded fasteners complies with the requirements of ASME BPVC Section III Division 1 Article NB-2000, NCD-2000, or NF-2000 as appropriate. Fracture toughness testing is performed in accordance with ASME BPVC Section III Division 1 Subarticle NB-2300, or NCD-2300, as appropriate. For verification of conformance to the applicable ASME BPVC requirements, a chemical analysis is required for each heat of material and testing for mechanical properties is required on samples representing each heat of material and, where applicable, each heat-treat lot.

The criteria of ASME BPVC Section III Division 1 Subarticle NB-2200, or NCD-2200, rather than the material specification criteria applicable to the mechanical testing is applied if there is a conflict between the two sets of criteria. For threaded fasteners, documentation related to fracture toughness (as applicable) and certified material test reports are provided as part of the ASME BPVC Section III Division 1 records that are provided at the time the parts are shipped and are part of the required records that are maintained at the site.

Threaded fasteners are selected for compatibility with the materials of the component being joined and the piping system fluids. The selection process considers deterioration that may occur during service as a result of corrosion, radiation effects, or instability of material.

3.6.8.2 Special Materials Fabrication Processes and Special Controls

The design of threaded fasteners complies with ASME BPVC Section III Division 1 Article NB-3000 or NCD-3000, as appropriate. Fabrication of threaded fasteners complies with ASME BPVC Section III Division 1 Article NB-4000, NCD-4000, as appropriate. Inspection of threaded fasteners complies with ASME BPVC Section III Division 1 NB-2500, or NCD-2500, as applicable.

3.6.8.3 Pre-service and In-service Inspection Requirements

Pre-service and In-service requirements of ASME BPVC Section III Division 1 Class 1, 2, and 3 Mechanical Systems and Components are based on a graded approach with SC1 equipment receiving the most pre-service required qualification. Chapter 3, Section 3.9 Equipment Qualification provides the required qualifications and tests for safety components. Chapter 3, Subsection 3.10.5 provides the In-service Inspection requirements for SSCs.

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3.6.9 References

- 3.6-1 ASME BPVC-IID (Metric), "Section II - Materials - Part D - Properties - (Metric)," American Society of Mechanical Engineers.
- 3.6-2 ASME BPVC-III APP, "Section III - Rules for Construction of Nuclear Facility Components - Appendices," American Society of Mechanical Engineers.
- 3.6-3 ASME BPVC-III NB, "Section III - Rules for Construction of Nuclear Facility Components, Subsection NB - Class 1 Components," American Society of Mechanical Engineers.
- 3.6-4 ASME BPVC-III NCD, "Section III - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NCD - Class 2 and Class 3 Components," American Society of Mechanical Engineers.
- 3.6-5 ASME BPVC-III NCA, "Section III - Division 1 and 2 - Subsection NCA, Rules for Construction of Nuclear Facility Components - General Requirements for Division 1 and Division 2," American Society of Mechanical Engineers.
- 3.6-6 ASME BPVC-III NE, "Section III Division 1 - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NE - Class MC Components," American Society of Mechanical Engineers.
- 3.6-7 ASME BPVC-III NF, "Section III - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NF - Supports," American Society of Mechanical Engineers.
- 3.6-8 ASME BPVC-III NG, "Section III - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NG - Core Support Structures," American Society of Mechanical Engineers.
- 3.6-9 ASME BPVC-III APP, "Section III - Rules for Construction of Nuclear Facility Components – Appendices - Mandatory Appendix II," American Society of Mechanical Engineers.
- 3.6-10 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 3.6-11 ASME B31T, "Standard Toughness Requirements for Piping," American Society of Mechanical Engineers.
- 3.6-12 ASME B31.3, "Process Piping," American Society of Mechanical Engineers.
- 3.6-13 CSA N289.4, "Testing procedures for seismic qualification of nuclear power plant structures, systems, and components." CSA Group.
- 3.6-14 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 3.6-15 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.6-16 CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants, Version 1.
- 3.6-17 NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.6-18 IEC/IEEE 60780-323, "Nuclear facilities – Electrical equipment important to safety – Qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers.

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- 3.6-19 IEC/IEEE 60980-344, "Nuclear facilities – Equipment important to safety – Seismic qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers.
- 3.6-20 ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," American Society of Mechanical Engineers.

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Table 3.6-1: Applicable Pressure Boundary Codes and Standards

Code or Standard Number	Title/Description
ASME Section III Division 1 BPVC Section II	Materials
ASME BPVC Section III, Division 1	BPVC Section III, Rules for Construction of Nuclear Facility Components (NCA, NB, NCD, NE, NF, NG)
ASME BPVC Section V	Nondestructive Examination
ASME BPVC Section VIII, Division 1	BPVC Section VIII-Rules for Construction of Pressure Vessel
ASME BPVC Section IX	Welding and Brazing Qualifications
ASME BPVC Section XI	Rules for In-service Inspection of Nuclear Power Plant Components
ASME B31.1	Power Piping
ASME B31.3	Process Piping
ASME B31.5	Refrigeration Piping and Heat Transfer Component Code
ASTM	American Society for Testing and Materials (various material and forms specifications for piping and related components)
API-620 (or equivalent)	Design and Construction of Large, Welded, Low-Pressure Storage Tanks
API-650 (or equivalent)	Welded Tanks for Oil Storage
AWWA-D100	Welded Carbon Steel Tanks for Water Storage

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Table 3.6-2: Load Combinations and Acceptance Criteria

Plant Event / Event Combination	Service Loading Combination⁽¹⁾⁽²⁾⁽³⁾⁽¹⁰⁾	Comments	ASME Service Level⁽⁴⁾
Design	$P_D + T_D + R_D$ Design		N/A
Normal Operation	N		A
Plant/System AOO	(a) $N + AOO_A$ (b) $N + AOO_B$		B
Normal Operation + SOE	$N + SOE^{(11)}$	OPG/CSA requirement for SOE ⁽¹¹⁾ for Level B	B ^{(6) (7)}
Design Basis Accident	(a) $N + DBA_A$ (b) $N + DBA_B$ Loadings	OPG/CSA requirement for DBE ⁽¹¹⁾ for Level C	C
Design Extension Condition	(a) $N^{(5)} + DEC_A$ (b) $N^{(5)} + DEC_B$	OPG/CSA requirement for CLE ⁽¹¹⁾ for Level D	D
Test ⁽⁹⁾	$P_t + T_t + D_t$		Testing Limit ⁽⁸⁾

- (1) The service loading combination also applies to Seismic Category A and B instrumentation and electrical equipment.
- (2) For vessels, loads induced by the attached piping are included as identified in their design specification. For piping systems, water (steam) hammer loads are included as identified in their design specification.
- (3) The method of combination of the loads is in accordance with NUREG-0484, Revision 1.
- (4) Service level requirements are only applicable to ASME BPVC Code, Section III components. The service levels are as defined in appropriate subsection of ASME BPVC Code, Section III, Division 1.
- (5) The Reactor Coolant Pressure Boundary (RCPB) is evaluated in the load combination using the maximum pressure expected to occur during the Postulated Accident.
- (6) Applies only to fatigue evaluation of ASME BPVC Code Class 1 components and core support structures.
- (7) For ASME BPVC Code Class 1, 2 and 3 piping changes and additions to ASME BPVC Code Section III NB-3600, NCD-3600 may be necessary to evaluate and meet stress limits.
- (8) Testing limits are per ASME BPVC Code Section III NB-3226.
- (9) Test conditions are only applicable to ASME components.
- (10) Nomenclature:
- a. AOO_x Loads for AOO event x
 - b. D Dead Load
 - c. D_t Dead Load for Test Condition
 - d. DBE Design Basis Earthquake Loads
 - e. DEC_x Loads for DEC event x
 - f. N Normal Operation Loads
 - g. P_D Design Pressure
 - h. P_t Test Pressure
 - i. DBA_x Loads for DBA event x

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- j. R_D Design Mechanical Loads
 - k. R_t Test Mechanical Loads
 - l. T_D Design Temperature
 - m. T_t Test Temperature
- (11) For. OPG, SOE, DBE and CLE are the earthquake levels defined in Section 3.2.5. Per OPG PSAR, $SOE = (1/3) * DBE$. CLE is defined in Supporting documents (6), but is expected to be $(1.5 \text{ to } 1.67) * DBE$.

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Table 3.6-3: Comparison of Event Frequency to Plant Conditions and Service Loadings

Design Condition (DC)	ASME Service Level	Quantitative Frequency (F) (1/year)
Normal Planned Operation (DC-1)	A; - loading during plant startup, operation, refueling, and shutdown.	Planned Operation
Anticipated Operational Occurrences (AOO) (DC-2)	B; - incidents of moderate frequency occasional, infrequent loadings without sustaining any damage or reduction in function.	$< 1\text{E-}02$
Design Basis Accidents (DBAs) (DC-3)	C; - incidents of low frequency – infrequent loadings causing no significant loss of integrity.	$1\text{E-}02 > F \geq 1\text{E-}05$
Design Extension Conditions (DECs) (DC-4)	D; - incidents of extremely low frequency loadings associated with beyond design basis accidents.	$F \leq 1\text{E-}05$

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Table 3.6-4: Normal Operating Events (DC-1)

Description	Number of Cycles/60 Years
Boltup	72
Startup	200
Turbine Roll and Increase to Rated Power	200
Daily/Weekly Load Reduction and Recovery	20,805
Rod Sequence/Pattern Change	30
Rated Power Operation	-
Reduction to 0% Power	200
Hot Standby	200
Shutdown	200
Vessel Flooding/Shutdown Cooling	72
Unbolt	72
Refuel	72

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Table 3.6-5: Test Events (DC-1)

Description	Number of Cycles/60 Years
Design/System Leakage Hydrostatic Testing	150
Turbine Stop Valve Test	3,120
Turbine Bypass Valve Test	720
Turbine control Valve Test	720
MSIV Closure Test	720

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Table 3.6-6: Anticipated Operational Occurrences (DC-2)

Description	Number of Cycles/60 Years
Loss of Feedwater Heaters – Partial	50
Loss of Feedwater Heaters – Total	10
Rod Withdraw Error at Startup	7
Turbine Generator Trip. Load Rejection – with Bypass	60
Turbine Control Valve Fail Open	1
Loss of Feedwater	15
Loss-of-Offsite Power	8
Loss of Condenser Vacuum	10
Inadvertent MSIV Closure (all MSIVs)	20

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Table 3.6-7: Design Basis Accidents (DC-3)

Description	Number of Cycles/60 Years
Improper Startup – Hot Cleanup Water System	1 (freq \leq 0.1)
Turbine Generator Trip. Load Rejection – Without Bypass	1 (freq \leq 0.1)
Reactor Overpressure – Backup Scram	1 (freq \leq 0.1)
Shutdown due to Inadvertent Isolation Condenser System (ICS) Initiation	1 (freq \leq 0.1)
Inadvertent Sodium Pentaborate Injection	1 (freq \leq 0.1)
Excessive Cooldown Rate	2 (freq \leq 0.1)

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Table 3.6-8: Design Extension Condition (DC-4)

Description	Number of Cycles/60 Years
Bounding Transient without Scram	≤ 0.001
Pipe Rupture – Loss-of-Coolant Accident	≤ 0.001
Ultimate Overpressure Protection	≤ 0.001

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Table 3.6-9: Summary of Cycles of Events

Event #	Description	Design Basis Number of Cycles
1	Boltup	72
2	Design/System Leakage Hydrostatic Testing	150
3	Startup	200
4	Turbine Roll and Increase to Rated Power	200
5/6	Daily/Weekly Load Reduction and Recovery	20,805
7	Rod Sequence/Pattern Change	30
8	Loss of Feedwater Heaters – Partial	50
9	Loss of Feedwater Heaters – Total	10
10/11	Turbine Generator Trip, Other Scrams with Bypass Flow	67
12	Rated Power Operation	-
13	Reduction to 0% Power	200
14	Hot Standby	200
15	Shutdown	200
16/17	Vessel Flooding/Shutdown Cooling	72
18	Unbolt	72
19	Refuel	72
20	Scrams Without Bypass	55
21	Improper Startup – Hot Reactor Water Cleanup System	1
22	Reactor Overpressure – Backup Scram	1
23	Shutdown due to Inadvertent Isolation Condenser System (ICS) Initiation	1
24	Improper Startup/Sodium Pentaborate Injection	1
25	Excessive Cooldown Rate	2
26	Bounding Transient Without Scram	1
27	Pipe Rupture – Loss-of-Coolant Accident	1
28	Ultimate Overpressure Protection	1

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**Table 3.6-10: Load Combinations and Acceptance Criteria for ASME
BPVC Section III Division 1 Class 1 Piping Systems**

Condition	Load Combination for all Terms ⁽²⁾⁽³⁾	Acceptance Criteria per ASME Code ⁽¹⁾⁽⁴⁾
Design	PD + WT	NB-3652
Service Level A and B ⁽⁵⁾	PP, TE, $\Delta T1$, $\Delta T2$, TA-TB, AOO, DBEI, DBED	NB-3653
Service Level B	PP + WT + AOO	NB-3654
Service Level C	PP + WT + DBA Where DBA includes but is not limited to: LOCA DBE	NB-3655
Service Level D	PP + WT + DEC Where DEC includes but is not limited to: SRSS (DBE+LOCA)	NB-3656

(1) Fatigue usage and stress limits are reduced for piping locations exempt from pipe break consideration.

(2) Where:

- a. WT = Dead Weight
- b. PD = Design Pressure
- c. PP = Peak Pressure or the Operating Pressure Associated with that transient
- d. DBEI = Design Basis Earthquake (inertia Effect)
- e. DBED = Design Basis Earthquake (Anchor Displacement Loads)
- f. DBE = Design Basis Earthquake includes both DBEI and DBED which are combined using SRSS method
- g. AOO = Anticipated Operational Occurrence
- h. DBA = Design Basis Accident
- i. DEC = Design Extension Condition

(3) LOCA is intended to represent loads and the appropriate combination of loads resulting from postulated line breaks including but not limited to Acoustic Inertial, Jet Reaction, and Jet Impingement loads

(4) ASME BPVC SECTION III NB-2021

(5) DBEI and DBED are Service Level C loads but must be considered for fatigue usage.

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**Table 3.6-11: Load Combinations and Acceptance Criteria for ASME BPVC III Division 1
Class 2 and 3 Piping Systems**

Service Level	Load Combination for all Terms⁽¹⁾⁽²⁾⁽³⁾	Acceptance Criteria per ASME Code⁽⁴⁾⁽⁵⁾
Design	PD + WT	NCD-3652
A & B	TE	NCD-3653.2
A & B	Single Non-repeated Anchor Movement	NCD-3653.2
A & B	PD + WT + TE	NCD-3653.2
B	PP + WT + AOO Where AOO includes but is not limited to: TSV	NCD-3653.1
C	PP + WT + DBA Where DBA includes but is not limited to: LOCA DBE	NCD-3654.2
C	PP	NCD-3654.1
D	PP + WT + DEC Where DEC includes but is not limited to: SRSS (DBE + TSV) SRSS (DBE + LOCA)	NCD-3655
D	PP	NCD-3655

(1) TSV loads are used for MS lines only

(2) Where:

- a. WT = Dead Weight
- b. PD = Design Pressure
- c. PP = Peak Pressure or the Operating Pressure Associated with that transient
- d. DBEI = Design Basis Earthquake (inertia Effect)
- e. DBED = Design Basis Earthquake (Anchor Displacement Loads)
- f. DBE = Design Basis Earthquake includes both DBEI and DBED which are combined using SRSS method
- g. AOO = Anticipated Operational Occurrence
- h. DBA = Design Basis Accident
- i. DEC = Design Extension Condition

(3) LOCA is intended to represent loads and the appropriate combination of loads resulting from postulated line breaks including but not limited to Acoustic Inertial (ACI), JR, and JI loads

(4) ASME BPVC SECTION III NCD-2021

(5) Stress limits are reduced for piping locations exempt from pipe break consideration.

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Table 3.6-12: Load Combinations and Acceptance Criteria for Non-Safety Class Power Piping Systems

Description	Load Combination	Acceptance Criteria per ASME Code⁽²⁾
Sustained	Design Pressure + Weight + other Sustained Loads	Paragraphs 102.3 and 104.8.1
Occasional	Design Pressure + Weight + Other Sustained Loads + Seismic	Paragraphs 102.3 and 104.8.2
Occasional	Design Pressure + Weight + Occasional event other than Seismic	Paragraphs 102.3 and 104.8.2
Thermal	Displacement Load Ranges	Paragraphs 102.3 and 104.8.3
Test	Test Pressure + Weight	Paragraph 102.3.3

(1) Stated in CSA N289.3: Clause 7.5.1 (Reference 3.6-15). For Class 6 piping in accordance with ASME B31.1-2020 rules, the k factor in the equation for stresses due to occasional loads including seismic loading is increased to 1.8. Alternatively, a conservative approach can be adopted in which the seismic stresses in the stress combination for occasional loads can be multiplied by factor 2/3 with the k factor equal to 1.2.

(2) ASME B31.1-2020

3.7 General Design Aspects for Instrumentation and Control Systems and Components

The BWRX-300 Distributed Control and Information System (DCIS) is an integrated control and monitoring system for the power plant. The DCIS is arranged in three safety classified DCIS segments that have appropriate levels of hardware and software quality corresponding to the system functions they control and their allocation to the Defense Lines (DL). The DCIS provides control, monitoring, alarming and recording functions. Although normally integrated, the various components of the DCIS are designed to operate independently.

The relationship between Instrumentation and Control (I&C) Functions and plant-level DLs is described in Chapter 7, Section 7.1.1. The classification of I&C systems is described in Chapter 7, Section 7.1.2, and is based on the general classification criteria described in Sections 3.2.1 and 3.2.2. The I&C system of systems is described in Chapter 7, Section 7.2. The individual I&C systems are described in Chapter 7, Section 7.3.

3.7.1 Performance

The system design bases, and associated safety functions, are described for the DL3 systems in Chapter 7, Subsection 7.3.1.2, for the DL4a systems in Subsection 7.3.2.2, for the DL2 systems in Subsection 7.3.3.2, and for the non-classified systems in Subsection 7.3.4.2.

3.7.2 Design for Reliability

The system reliability requirements and associated design features are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.2, for the DL4a systems in Subsection 7.3.2.3.2, for the DL2 systems in Subsection 7.3.3.3.2, and for the non-classified systems in Subsection 7.3.4.3.2.

3.7.3 Independence

The system independence requirements and associated design features are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.3, for the DL4a systems in Subsection 7.3.2.3.3, for the DL2 systems in Subsection 7.3.3.3.3, and for the non-classified systems in Subsection 7.3.4.3.3.

3.7.4 Qualification

The system qualification requirements are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.1, for the DL4a systems in Subsection 7.3.2.3.1, for the DL2 systems in Subsection 7.3.3.3.1, and for non-classified systems in Subsection 7.3.4.3.1.

3.7.5 Verification and Validation

The system verification and validation requirements for I&C systems are described in Chapter 7, Section 7.4.3.

3.7.6 Failure Modes

The application of the single failure criterion to DL3 systems is described in Chapter 7, Subsection 7.3.1.3.3. The effects of failures and associated design features to minimize or eliminate adverse effects of anticipated failures are described for the DL4a systems in Subsection 7.3.2.3.3, for the DL2 systems in Subsection 7.3.3.3.3, and for the non-classified systems in Subsection 7.3.4.3.3.

The use of diversity to eliminate common cause failure vulnerabilities or minimize the effects of postulated common cause failures is described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.5, for the DL4a systems in Subsection 7.3.2.3.5, for the DL2 systems in Subsection 7.3.3.3.5, and for the non-classified systems in Subsection 7.3.4.3.5.

3.7.7 Control of Access to Equipment

The system security requirements (including control of access) are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.4, for the DL4a systems in Subsection 7.3.2.3.4, for the DL2 systems in Subsection 7.3.3.3.4, and for the non-classified systems in Subsection 7.3.4.3.4.

3.7.8 Quality

The codes and standards used for the I&C systems are described in Chapter 7, Section 7.1.3. The system quality requirements are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.1, for the DL4a systems in Subsection 7.3.2.3.1, for the DL2 systems in Subsection 7.3.3.3.1, and for the non-classified systems in Subsection 7.3.4.3.1.

3.7.9 Testing and Testability

The system testing requirements (including design features to support testability) are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.2, for the DL4a systems in Subsection 7.3.2.3.2, for the DL2 systems in Subsection 7.3.3.3.2, and for the non-classified systems in Subsection 7.3.4.3.2.

3.7.10 Maintainability

The system maintainability requirements and associated design features are described for the DL3 systems in Chapter 7, Subsection 7.3.1.3.2, for the DL4a systems in Subsection 7.3.2.3.2, for the DL2 systems in Subsection 7.3.3.3.2, and for the non-classified systems in Subsection 7.3.4.3.2.

3.7.11 Identification of Items Important to Safety

The I&C system classification information is described in Section 7.1.2.

3.8 General Design Aspects for Electrical Systems and Components

The BWRX-300 electrical power system has been designed as a minimum to meet the requirements of CNSC REGDOC 1.1.2 and CNSC REGDOC 2.5.2.

The electrical power system design is a 60 Hz Alternating Current (AC) power system, with 4.16 kV for the Medium Voltage (MV) level and 600 V for the Low Voltage (LV) level.

The off-site electrical system is provided and managed by OPG. The function of the BWRX-300 off-site electrical system is to provide electrical power to the Hydro One managed grid that is compatible and consistent for OPG purposes. The output of the BWRX-300 is monitored for over voltage and over/under current as protective design features to prevent possible grid disruptions. The off-site power system can be automatically or manually disconnected from the grid if the electrical power is found to be disrupted for any reason.

On-site electrical systems are designed to support the normal operations of the BWRX-300. A unique feature of the BWRX-300 plant is that the on-site AC power system is not required to be operational to support the safe shutdown of the reactor and for at least the first 72 hours following shutdown. The reactor cooldown is accomplished through natural circulation and passive cooling via the ICS system.

The off-site preferred power system is designed to provide a continuous source of power to the on-site AC power system throughout plant startup, normal operation (including shutdown), and abnormal operations. The off-site power system provides no credited safety function. As a result, the total loss-of-offsite power results in no impact on nuclear safety.

Refer to Chapter 8 – Electrical Power for a detailed discussion on the Electrical power systems for the BWRX-300.

The on-site AC power system consists of SCN, SC1, SC2, and SC3 power systems. The two off-site power sources provide the normal preferred and alternate preferred AC power to SCN, SC1, SC2 and SC3 loads.

The normal preferred off-site power source is connected to the GSU, which is connected to the plant generator and the UAT. The normal preferred power source is distributed from the UAT secondary windings to MV SCN busses, which further distribute the power to SCN loads and the SC3 LV busses. The SC3 LV busses serve LV SC3 loads and provide normal AC power to the SC1 and SC2 electrical power systems.

The alternate preferred off-site power source is connected to the RAT, which has two MV secondary windings like the UAT. The RAT provides alternate power feeds to the MV SCN busses for cases when the UAT is not in-service.

The SC3 LV busses also have backup power in the form of standby diesel generators. Each SC3 LV bus is connected to a standby diesel generator that automatically starts and loads if the normal power to the SC3 LV bus becomes unavailable (loss of power or degraded).

There are three divisions of SC1 DC power, two load groups of SC2 DC power, and 2 sets of SCN DC power connected to the diesel-backed SC3 busses. Add that each DC power system includes battery chargers, batteries, and UPSs to supply uninterruptible AC and DC power during loss of power events.

The BWRX-300 electrical AC power systems (on-site or off-site) are not relied upon to support the safe shutdown and cooldown of the reactor in the event of a design basis accident. No operator actions are credited in the safe shutdown or cooldown of the reactor in the event of a design basis accident.

3.8.1 Redundancy

As discussed above, two off-site power sources provide the normal preferred and alternate preferred AC power to SCN, SC1, SC2 and SC3 loads. In the event of total loss-of-offsite power sources SC3 standby diesel generators are provided to power the plant SC1, SC2 and SC3 loads.

Three divisions of SC1 DC power are not only redundant to each other, but also have redundant UPSs in each divisions for further reliability. The SC2 DC power load groups are redundant to each other as well. There are also two sets of SCN DC systems that can provide redundant power to select equipment as needed.

There are two redundant SC2 Direct Current (DC) load groups and one SC3 Direct Current (DC) load group each with a UPS to provide power to the respective SC2 and SC3 loads.

There are three independent SC1 Direct Current (DC) divisions with UPS to provide power to SC1 loads.

Redundancy for the BWRX-300 electrical power systems is discussed in more detail in Chapter 8.

3.8.2 Independence

As discussed above, in the event of total loss-of-offsite power sources two on-site SC3 standby diesel generators are provided to power the plant SC1, SC2 and SC3 loads. Either SDG can support the required SC1, SC2, and SC3 loads needed for active decay heat removal. The SDG's are located in independent fire-barriered rooms.

The 3 divisions of SC1 DC power are electrically and physically independent from each other. There are no electrical connections between the divisions and the equipment is located by division in separate fire and flood-barriered rooms.

It is also the same for the SC2 load groups, (i.e., the two SC2 load groups are similarly independent from each other).

There are two independent SC2 Direct Current (DC) load groups and one SC3 Direct Current (DC) load group each with a UPS to provide power to the respective SC2 and SC3 loads.

There are three independent SC1 Direct Current (DC) divisions with UPS to provide power to SC1 loads.

Independence of the electrical power systems and components is discussed in more detail in various Chapter 8 sections. Refer to Chapter 8 for further discussion of this topic.

3.8.3 Diversity

The EDS is designed along a Defence-in-Depth philosophy and along Defense Lines. Section 3.6 provides a discussion on philosophy. The electrical systems are diverse from each based on defense lines.

3.8.4 Controls and Monitoring

On-site and Off-site electrical power system controls and monitoring for the BWRX-300 will be accomplished by both Main and Secondary Control rooms monitors or controls that are remote "at the panel" monitoring and controls should it be necessary to operate the electrical systems in a remote "away from the CR" fashion.

Controls and Monitoring is discussed in Chapter 8.

3.8.5 Identification

Refer to Section 8.4 for details on the electrical system safety classification and a description of the major electrical power system equipment.

3.8.6 Capacity and Capability of Systems for Different Plant States

The capacity and capability of the Electrical Power Systems is designed to provide a minimum of 100% of the required electrical loading needed for the normal operation of the BWRX-300. Equipment sizing includes consideration of design margin as appropriate for all facets of plant operation.

As stated above, the BWRX-300 does not rely on electrical power to safely shutdown and cool the reactor. Electrical power is not relied upon to place the reactor into a safe shutdown and to maintain the reactor in a safe shutdown condition.

As mentioned previously, SDG capacity can support required SC1/2/3 loads needed for active decay heat removal.

DC power from batteries will be used primarily to monitor the cooldown and condition of the reactor.

The capacity and capability of electrical power system is further discussed in Chapter 8.

3.8.7 External Grid and Related Issues

External Grid operation and management is the responsibility of OPG. The BWRX-300 safety design does not require off-site power to be present to mitigate any design basis accidents.

OPG's grid connection project is currently in the conceptual and planning stage.

With input and interfacing support, OPG plans on designing and building a local switchyard to consolidate power output from the BWRX-300 SMR Facility and connect it with Ontario electrical power grid. Hydro One is the grid transmitter and the Independent Electricity System Operator (IESO) is the electrical system operator.

At this time, OPG is expected to be the operator of the local switchyard via the Main Control Room (MCR) in the SMR Facility. The SMR Facility electrical AC power system will have two high voltage connections with the local switchyard at a 230kV voltage level. One line to output power from the Generator Step Up Transformer (GSU) and one line to supply power to the Reserve Auxiliary Transformer (RAT). The local switchyard will have two redundant 230kV connections with the transmitter. Each line will be designed to transmit the full generation capacity of the SMR Facility. The transmitter is responsible for building the transmission infrastructure needed to connect the local switchyard to Clarington TS, 22km North of the DNNP site. The two lines are expected to share the same tower structure. *(The 230kV voltage level and connection with Clarington TS is to be confirmed in 2022 through an IESO Feasibility Study.)*

The local switchyard will be of an indoor Gas Insulated Switchgear type, following a breaker and half arrangement with two redundant busses. The local switchyard will be designed to have local and remote-control capability. The plan for the local DNNP switchyard is that it will be located North of the SMR Facility, East of the Extended Holt Rd and South of the CN Rail tracks. The local switchyard control and protection designs will be coordinated with the SMR Facility controls and protections to meet IESO, NPCC and NERC codes and standards.

Power Quality

The BWRX-300 electrical power systems will be monitored for power quality issues (voltage/frequency/harmonics) that may arise and maintained such that any abnormal fluctuations in the voltage, current or capacity is alarmed in the Main Control Room so operators can evaluate and manually respond to the alarm condition.

3.9 Equipment Qualification

3.9.1 Purpose

This section defines the requirements related to equipment qualification in alignment with CNSC REGDOC-2.5.2, Section 5.5 (Reference 3.9-1).

Equipment qualification is the process carried out (including the generation and maintenance of evidence) to ensure SSC can perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform.

The conditions impacting equipment qualification include seismic/dynamic, environmental, functional/aging stressors, and electromagnetic interference.

3.9.2 Scope

Equipment qualification requirements are applied to BWRX-300 equipment based on the assigned safety classification and seismic categorization of SSC (as described in Section 3.2), and to certain post-accident monitoring equipment.

Equipment qualification considers all normal operating conditions in which the SSC are expected to operate including conditions arising from maintenance and testing, and also, the conditions arising from AOOs, DBAs, and internal and external hazards.

While DECAs are generally considered outside of the scope of a qualification program, guidance is provided for demonstrating with reasonable assurance that equipment credited to perform under DEC conditions will survive to perform its function. See Subsection 3.9.3.5 for consideration of a Beyond-Design Basis Earthquake (BDBE) and Subsection 3.9.4.1 for Environmental Qualification considerations.

The focus of this section is on qualification of mechanical and electrical equipment. Mechanical equipment consists of items of a facility including pumps, valves, vessels, and piping whose function is required to ensure the safe operation or safe shutdown. Electrical equipment consists of all electrical power and Instrumentation and Control (I&C) equipment, which includes all analog (non-digital) and digital I&C components. Computer-based I&C equipment is a subset of digital I&C components.

Qualification of civil structures is covered in Section 3.3.

3.9.3 Seismic

3.9.3.1 General

Seismic qualification is a subset of equipment qualification that is the verification, through testing, analysis, or other methods, of the ability of an SSC to perform its intended function during and/or following a designated earthquake. The dynamic loads of Reactor Building Vibrations (RBVs) and events caused by hydrodynamic loads are also considered. Seismic and dynamic qualification of BWRX-300 equipment and associated supports meets the requirements and recommendations of the CSA N289 series (References 3.9-2 To 3.9-6) as endorsed by CNSC REGDOC-2.5.2 (References 3.9-1), and IEC/IEEE 60980-344 (Reference 3.9-7).

The requirement for seismic qualification is based on the seismic categorization of SSC and the earthquake level they are required to withstand during and/or after the seismic event. Seismic categorization of BWRX-300 SSC is described in Section 3.2. Seismic Category A and Seismic Category B SSC are most important and have the most stringent requirements for functional integrity during and following a seismic event. Per regulatory guidance of CNSC REGDOC-2.5.2, Section 5.13.1 (Reference 3.9-1), SSC that are classified as Seismic Category A and Seismic

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Category B are seismically qualified to withstand the effects of a DBE. The site-specific DBE is defined in Subsection 3.3.1.

BWRX-300 equipment Seismic Categories are identified in Appendix 3A Table 3.12-1. Seismic Categorization of Structures is provided in Section 3.3, Table 3.3-1.

3.9.3.2 Methods for Seismic Qualification

Seismic and dynamic qualification of equipment and associated supports are accomplished by test, analysis, or a combination of testing and analysis. Seismic and dynamic qualification of equipment and associated supports designated as SC1 is accomplished by testing. Seismic and dynamic qualification of equipment and associated supports designated as SC2 may be accomplished by analysis or a combination of testing and analysis.

Qualification by actual seismic experience (also referred to as seismic qualification by similarity), as described in IEC/IEEE 60980-344 (Reference 3.9-7) and CSA N289.1 (Reference 3.9-2), is also utilized as appropriate considering the limitations identified in CSA N289.1, Annex D.3 (Reference 3.9-2).

The selection of qualification method to be used is largely a matter of engineering judgment for cases where testing is not required. When both test and analysis are defined as acceptable methods, the deciding factors considered (as applicable) for choosing between tests or analysis include magnitude and frequency of seismic and RBV dynamic loadings, environmental conditions associated with the dynamic loadings, nature of the safety category function(s), size and complexity of the equipment, dynamic characteristics of expected failure modes (structural or functional), and partial test data upon which to base the analysis.

Tests or analyses of assemblies are preferable to tests or analyses on separate components (e.g., a motor and a pump, including the coupling and other appurtenances, should be tested or analyzed as an assembly), unless deemed not practical. Equipment that has been previously qualified by means of tests and analyses equivalent to those required for the current qualification program are used if proper documentation of such tests and analyses is available.

For equipment defined as requiring test for qualification, analysis by similarity may be used if similar equipment is being or has been qualified by test.

3.9.3.2.1 Testing

Testing of BWRX-300 SSC for seismic qualification is conducted in accordance with CSA N289.4 (Reference 3.9-5) IEC/IEEE 60980-344 (Reference 3.9-7).

Seismic qualification by testing is typically used for SSC that will be performing an active function and are required to change state during or following a seismic event to perform a safety category function, while maintaining structural and/or pressure boundary integrity. Seismic testing can identify contact chatter or unauthorized change of state of contact in electrical and I&C components during seismic excitation.

The dynamic test sequence includes as applicable, vibration conditioning, exploratory resonance search, low-level earthquake loading (one-half DBE) including Reactor Building Vibrations (RBV) dynamic loads and the DBE loading including RBV dynamic loads.

Dynamic tests are performed with the equipment subjected to nominal operating service conditions. Significant, normal operating loads such as electrical, mechanical, pressure, and thermal are included. Where normal operating loads cannot be included in the dynamic tests, supplemental analysis is used to qualify the equipment for those effects. If there is any dynamic coupling due to interacting equipment, it is considered.

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For equipment located in multiple locations, the enveloping upper bound seismic condition limits are used to eliminate the need for multiple qualification tests, unless otherwise specified.

Resonance Tests

When required, exploratory resonance search tests (such as sine sweeps or random vibration) are used for equipment to help determine the method of test or analysis that would be best for qualification and/or determine the dynamic characteristics such as the resonance frequencies of the equipment, mode shapes and damping values.

Sine sweep resonance search is the preferred method and is performed by running a continuous sweep frequency search using a sinusoidal steady-state input at the lowest possible amplitude at which resonance can be determined.

Resonance searches may be performed prior to and after the seismic test to determine any shifts in frequency caused by testing.

If resonance frequencies are present, the transmissibility between the input and the location of the equipment is determined by measuring the accelerations at the equipment location and calculating the magnification between it and the input.

Floor-mounted frequency testing can be used as another method to determine the resonance or natural frequencies for equipment.

Seismic Input Motion

Dynamic load conditions are simulated by testing, using independent, random multi-frequency input or single frequency input motion (within equipment capability) over the frequency range of interest.

Acceptable justification for use of single frequency input includes, but is not limited to:

1. The characteristics of the required input motion are dominated by one frequency.
2. The anticipated response of the equipment is adequately represented by one mode.
3. The input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra envelop the corresponding response spectra of the individual modes.
4. The time phasing of the inputs in the vertical or horizontal directions will be such that a purely rectilinear resultant input is avoided.

The actual input motion used during testing, for both multi and single frequency, envelops the applicable input motion (floor, wall, response, etc.) at the location(s) of the equipment under test.

When the equipment is qualified by dynamic test, the In-Structure Response Spectra (ISRS) or time histories, developed from the results of Soil-Structure Interaction (SSI) analyses as described in Section 3.3.1.2.7, representing the in-structure seismic response of the attachment point is used in determining required response spectra of input motion used for the test.

For the case of equipment having multiple supports with different dynamic motions, the effects of the multiple support attachment points must be considered in the dynamic qualification and can be accounted for by selecting an upper bound envelope of all the individual response spectra for these locations to calculate the maximum internal responses applicable to the equipment, unless otherwise specified.

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Past testing demonstrates that Seismic Category A electrical equipment has critical damping ratios equal to or less than 5%. Hence, the required response spectra at 5% or less critical damping ratio are developed as input to the equipment base, unless identified otherwise.

Seismic Test

The preferred test method for seismic qualification is shake table testing. Seismic testing is performed in a manner that demonstrates dynamic response characteristics and acceptability of the test specimen to withstand and maintain its function as required during the expected level of shaking. Test requirements are normally specified in the form of required response spectra at a specified damping value and confirmed by a Test Response Spectra (TRS) generated from the table motion.

The seismic test for DBE produces a TRS that envelops the applicable portion of the required response spectra as defined in the test specification (typically by a factor of 1.1) per CSA N289.4 (Reference 3.9-5). The approach is to apply 10% to the acceleration of the ISRS, developed from the results of SSI analyses as described in Section 3.3.1.2.6, which meets the recommendations of IEC/IEEE 60780-323 (Reference 3.9-8).

Testing for low-level earthquake loading and RBV dynamic loads is performed to demonstrate that the low-level earthquake loads combined with RBV dynamic loads do not degrade the continued structural and functional integrity of the equipment.

Testing for DBE loading and RBV dynamic loads are performed to demonstrate that equipment would perform its intended function(s) through DBE combined with RBV dynamic loads.

For both low-level earthquake and DBE seismic test runs, the input excitation TRS is required to envelop the specified required response spectra levels in accordance with CSA N289.4 (Reference 3.9-5) and Section 9 of IEC/IEEE 60980-344, (Reference 3.9-7).

If the TRS do not meet the requirements (i.e., do not envelop the required response spectra, do not demonstrate stationarity, do not demonstrate statistical independence) for the seismic test run, the test run is documented as unacceptable, adjustments may be required, and then the test is repeated.

Alternatively, per Clause 5.1.2.2.4 of CSA N289.4 (Reference 3.9-5), for acceptance in cases where TRS does not envelop required response spectra, the following criteria are applied:

- The number of points below the required response spectra shall not exceed 5
- The points shall not fall below the required response spectra by more than 10%
- Any two points below the required response spectra shall be at least 1 octave apart
- The points adjacent to the points that fall below the required response spectra shall be at least 10% above the required response spectra

For equipment that is subjected to vibration in its in-service condition, vibrational aging to its end of life condition must be completed prior to seismic testing (both low-level earthquake and DBE load tests).

For seismic qualification, the seismic input consists of five one-half DBE amplitude events (low-level earthquakes) followed by one DBE event. Alternatively, in accordance with Annex E of IEC/IEEE 60980-344 (Reference 3.9-7), a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half DBE events may be used followed by one full DBE event; however, in this case the amplitude shall not be below the minimum of one-half the DBE input motion.

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The preferred method for seismic testing is to use triaxial, multi-frequency testing. However, if justified, biaxial and single-axis testing is acceptable.

Multi-frequency, multi-axis dynamic tests (triaxial or biaxial) are used to qualify equipment with a single resonance or multiple resonances within the frequency range of interest or if the critical resonance frequencies cannot be ascertained.

Single frequency testing is allowed if:

1. It can be demonstrated that the component is subjected to no resonances, or one predominant resonance frequency that is not in the frequency range of interest, or if the resonance frequencies are widely separated and do not interact to reduce the fragility level in the frequency range of interest, or if otherwise justified.
2. Single-axis tests can only be used if the tests are designed to conservatively reflect the dynamic event at the equipment mounting locations or if the equipment being tested can be shown to respond independently in each of the three orthogonal axes or otherwise withstand the dynamic event at its mounting location.

Equipment is tested in a functionally operable condition to allow for the monitoring of safety requirements throughout the seismic testing.

Equipment is operated at appropriate times (as necessary) to demonstrate the ability to perform its safety category function throughout the seismic testing.

For Seismic Category A and B mechanical and electrical equipment, it is defined if the equipment must perform its safety category function before, during, and after seismic events (typical for most equipment), or only before and after seismic events (applicable to some equipment such as plant status display equipment).

The equipment damping value used for dynamic qualification is established in accordance with Section 5 of IEC/IEEE 60980-344 (Reference 3.9-7).

Documentation of seismic testing is in accordance with Section 13 of IEC/IEEE 60980-344, (Reference 3.9-7) and include, at minimum, locations of accelerometers, any existing resonance frequency(s) and transmission ratios, equipment damping coefficients if there is resonance over the range of the test response spectra, test equipment used, any modifications made to test specimen, hardware interface requirements, test methods, approval signature and dates, description of test facility, summary of results, equipment seismic qualification conclusions (including RBV dynamic loads), anomalies and their resolution, test data, and justification for using single-axis or single frequency tests for all items that are tested in this manner.

3.9.3.2.2 Selection of Test Specimen

Test specimens are selected as representative samples of the production equipment and supports that are covered by the qualification program. Test specimens are manufactured using the same process that are implemented for the production units. Variations in the configuration of the equipment are analyzed with supporting test data. For example, these variations may include mass distributions that differ from one cabinet to another. From test or analysis, it is determined which mass distribution results in the maximum acceleration or frequency content, and this worst-case configuration is used as the test specimen. The test report includes a justification that this configuration envelops all other equipment configurations.

3.9.3.3 Seismic Analysis

Dynamic analysis or an equivalent static analysis is employed to qualify the equipment when analysis is chosen as the method for qualification per CSA N289.3, Section 6 (Reference 3.9-4).

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The decision on using dynamic versus static analysis is typically defined based on whether the equipment is rigid or flexible.

If the fundamental frequency of the equipment is above the input excitation frequency (cutoff frequency of required response spectra) the equipment is considered rigid.

The search for the natural frequency is done analytically, if the equipment shape can be defined mathematically, or by prototype testing.

If the equipment is determined to be a rigid body (i.e., shown to have no resonance frequency within the expected frequency range) the static analysis method is able to be applied in place of dynamic analysis.

If the equipment is determined to be flexible (i.e., with the fundamental frequency of the equipment within frequency range of the input spectra) and not simple enough for equivalent static analysis, a dynamic analysis method is applied, unless justified otherwise.

If it is determined that either dynamic or static analysis can be used, in general, the choice of the analysis is based on the expected design margin, as the static coefficient method is more conservative than the dynamic analysis method.

For static analysis, the dynamic forces on each component can be obtained by concentrating the mass at the center of gravity and multiplying the mass by the appropriate floor acceleration. The dynamic stresses are then added to the operating stresses and a determination made of the adequacy of the strength of the equipment.

A static coefficient analysis may also be used for certain equipment in lieu of the dynamic analysis. No determination of natural frequencies is made in this case. The seismic loads are determined statically by multiplying the actual distributed weight of the equipment by a static coefficient equal to 1.5 times the peak value of the required response spectra at the equipment mounting location, at a conservative and justifiable value of damping.

Both types of analyses verify integrity of the equipment is maintained under low-level earthquake loads including appropriate RBV dynamic loads in combination with normal operating loads and normal operating and DBE loads including appropriate RBV dynamic loads, unless otherwise justified.

See Section 3.3.1.3 for additional details and discussion of Seismic Analysis of Seismic Category A and B Subsystems.

3.9.3.4 Seismic Qualification by Combined Testing and Analysis

Qualification by combined testing and analysis is used as a method for qualification for complex or large equipment where it is not practical to test the entire assembly or it is too large to be tested at once, unless another method of qualification is justified.

One method of combined qualification is to use a representative prototype portion or scaled-down prototype of the assembly that is subjected to type testing. The data from the type testing is then used to develop and validate an analytical model of the prototype. The prototype analytical model is then extrapolated to represent the larger assembly and then using the results to justify qualification of the equipment based on prototype testing.

A second method of combined qualification is to mount the full assembly to a rigid floor to simulate service mounting and then a portable shaker test (or an impact or pull test if justified) is performed to excite the natural or resonance frequencies of the specimen. The amplification of resonance motion is used to determine the appropriate modal frequency and damping for a dynamic analysis of the equipment.

For equipment with multiple site configurations the combined qualification method can be applied to reduce the number of configurations to be tested. In this case, an evaluation must be performed to determine the enveloping “worst-case” configuration(s), which is then tested. Analysis is then used to justify the various configurations based on the “worst-case” configuration(s).

The combination method can be used for qualification of larger electrical equipment support assemblies containing Seismic Category A or B equipment. For this case, a test is run to determine if there are natural frequencies in the support equipment within the critical frequency range. If the support is determined to be free of natural frequencies in the critical frequency range, then it is assumed to be rigid and a static analysis is performed and calculations of transmissibility and responses to varying input accelerations are determined to see if Seismic Category A or B equipment mounted in the assembly would operate without malfunctioning.

3.9.3.5 Beyond Design Basis Earthquake

REGDOC-2.5.2 Section 7.13 (Reference 3.9-1) states that for a beyond-design-basis earthquake (BDBE), demonstration that there is a high confidence of low probability of failure (HCLPF) of the SSC that are credited to function during and after the event. This demonstration need not be seismic qualification by testing. BDBE is identified as a Checking Level Earthquake (CLE). Typically, the CLE (as discussed in Section 3.5.6.1.2) is considered a DEC. DEC for seismic events are a subset of beyond design basis seismic events that are considered in the evaluation of the facility using best-estimate methodology to keep releases of radioactive material within acceptable limits.

If determined to be useful, fragility testing per IEC/IEEE 60980-344 (Reference 3.9-7) may be used as a qualification method. Fragility testing is a form of vibration testing of an SSC to determine the point where it can no longer perform its function, whether due to electrical or mechanical malfunction, or excessive structural deformation or destruction. Where fragility testing is performed, it provides useful information about margin to failure. Knowledge of the seismic fragility of an SSC is useful in determining its seismic margin to failure and in providing determination of SSC functionality in BDBE evaluations (per CSA N289.1 (Reference 3.9-2).

Seismic PSA is used to analyze the plant response to seismic hazards as discussed in Chapter 15, Section 15.6.

3.9.3.6 Documentation

Seismic qualification documentation including identification of seismic equipment, test/analysis plans and reports, technical specifications, data sheets, engineering standards, and component specific seismic qualification parameters, and requirements for inspection, maintenance and procurement are prepared in an auditable summary report in accordance with Clause 7 of 289.4 (Reference 3.9-5).

Documentation of seismic testing is in accordance with CSA N289.4 Section 5.8 (Reference 3.9-5) and IEC/IEEE 60980-344, Section 13 (Reference 3.9-7) and include, at minimum, locations of accelerometers, any existing resonance frequency(s) and transmission ratios, equipment damping coefficients if there is resonance over the range of the test response spectra, test equipment used, any modifications made to test specimen, hardware interface requirements, test methods, approval signature and dates, description of test facility, summary of results, equipment seismic qualification conclusions (including RBV dynamic loads), anomalies and their resolution, test data, and justification for using single-axis or single frequency tests for all items that are tested in this manner.

3.9.4 Environmental Qualification

3.9.4.1 Scope

Environmental Qualification is a subset of equipment qualification specifically addressing equipment exposure to a harsh environment. In alignment with CNSC REGDOC-2.5.2 Section 7.8 (Reference 3.9-1) and CSA N290.13 (Reference 3.9-9), Environmental Qualification is established to ensure that BWRX-300 SC1 SSC can perform their FSFs during and after exposure to a harsh environment resulting from a DBA during and after which they are required to operate. Equipment whose failure due to the harsh environment could impair the ability of qualified equipment to perform safety category functions are also considered for Environmental Qualification. Equipment that is not significantly impacted by the increased stress due to the harsh environment, or for which there are not credible failure modes induced by the harsh environment preventing the equipment from performing its FSF are exempt from Environmental Qualification. The effects of normal service conditions including that of AOOs, and the impact of aging are considered in the SSC ability to perform their safety category functions.

While Environmental Qualification is not required to be established for equipment responding to DECAs as stated in CSA N290.13 (Reference 3.9-9), equipment survivability assessments are used to provide reasonable confidence that equipment will function in response to the DEC within the time span required and that instrumentation will function with reasonable accuracy per REGDOC-2.5.2 (Reference 3.9-1). IEC/IEEE 60780-323 (Reference 3.9-8) provides considerations for qualifying equipment for DECAs and guidance is provided in Annex B of CSA N290.13 (Reference 3.9-9), and CSA N290.16 (Reference 3.9-10).

3.9.4.2 Environment Parameters

A harsh environment occurs as a result of a subset of DBAs for which ambient and operational service conditions change significantly as a result of the DBAs, DBAs considered in the BWRX-300 design are discussed in Chapter 15. Environmental parameters considered when screening for a harsh environment include:

- Temperature
- Steam
- Condensing Humidity
- Pressure
- Submergence
- Radiation
- Chemistry

Table 3.9-1 lists harsh environment screening criteria for environmental parameters based on the guidance in CSA N290.13 Annex A (Reference 3.9-10).

Per CSA N290.13, (Reference 3.9-10), a mild environment is one that would at no time be significantly more severe than the environment that would occur during the normal plant operation, including during AOOs, and would not give rise to significant aging mechanisms. For equipment located in a mild environment during and after a DBA for which it is required to function, Environmental Qualification is not required.

Per the description of mild environment qualification in CNSC REGDOC-2.5.2, Section 7.8 (Reference 3.9-1), for equipment not requiring Environmental Qualification per the scope of CSA N-290.13 (Reference 3.9-9) as described herein, the environmental conditions for its expected

function would be identified in its design specification and a manufacturers certification that the equipment meets the specification would be provided.

3.9.4.3 Objectives

The objectives of Environmental Qualification of BWRX-300 SSC include:

1. Identification of SSC required to be environmentally qualified
2. Establishment of the safety category functions, performance requirements, normal service conditions, and post-accident harsh environment conditions for SSC identified as requiring qualification
3. Documentation of objective evidence verifying that the identified SSC are capable of performing credited safety category functions under the relevant harsh conditions, including consideration of age-related degradation during normal service
4. Controls and evidence to ensure that SSC are installed considering identified configuration and interface requirements
5. Controls and evidence to ensure that qualification of the equipment is preserved throughout the design life including aging and obsolescence

3.9.4.4 Requirements for Environmental Qualification

3.9.4.4.1 DBA Identification

BWRX-300 DBAs that produce a harsh environment with potential to cause common cause failures are identified and analyzed at the appropriate design phase. Documentation of the basis for classifying an accident as harsh is included.

3.9.4.4.2 Defining Normal and Accident Environmental Envelope

At the appropriate design phase an environmental envelope that includes a listing of all areas of the facility in which SSC are expected to fulfill safety category functions during and after a DBA is identified and documented. For each identified area, the ambient environmental and operational conditions are provided for normal conditions (normal operating modes and AOOs), and for DBA conditions based on the limiting parameters identified from DBA identification.

3.9.4.4.3 Identification of Equipment Requiring Harsh Environment Qualification

At the appropriate design stage, BWRX-300 equipment requiring Environmental Qualification (as described in 3.9.4.1) is identified and documented. The list also includes equipment whose failure due to the harsh environment could impair the performance of qualified equipment. Equipment that is not significantly impacted by the increased stress due to the harsh environment, or for which there are not credible failure modes induced by the harsh environment preventing the equipment from performing its safety category function is exempt from Environmental Qualification. A basis for exempting equipment from qualification (e.g., failure modes, environmental conditions, materials, etc.) will be documented.

Information documented in the list of environmentally qualified equipment includes:

- Equipment identification
- Safety category function
- Applicable DBA
- Mission time
- Normal and accident service conditions

3.9.4.4.4 *Qualified Life*

Qualified life is established for equipment determined to be susceptible to age-related degradation for the specified service conditions. The equipment included within the scope of the Environmental Qualification program is analyzed based on an expected plant life of 60 years or is subject to replacement or evaluation of the effects of aging and obsolescence on a periodic basis.

3.9.4.5 *Establishing Environmental Qualification*

Methods for demonstration that equipment is environmentally qualified include testing, analysis, by operating experience, or by a combination of these methods in accordance with CNSC REGDOC-2.5.2 (Reference 3.9-1), CSA N290.13 (Reference 3.9-9), Reg. Guide 1.89 (Reference 3.9-11), and IEC/IEEE 60780-323, (Reference 3.9-8).

3.9.4.5.1 *Qualification By Testing*

Type testing is the preferred method for demonstrating that equipment is Environmentally Qualified. A type test subjects a representative sample of equipment, including interfaces, to a series of tests, and include simulating the effects of significant aging mechanisms during normal operation. The sample is subsequently subjected to conditions that simulate DBA harsh conditions and thereby establishes the tested configuration for installed equipment service, including mounting, orientation, interfaces, conduit sealing, and expected environments. A type test demonstrates that the equipment performs the intended safety category function(s) for the required operating time before, during, and/or following the DBA, as appropriate.

Type tests are performed in accordance with applicable industry standards, such as CSA N290.13 (Reference 3.9-9) and IEC/IEEE 60780-323 (Reference 3.9-8).

A typical sequence includes, but is not limited to the following:

- Initial inspection
- Baselines functional test
- Normal radiation exposure
- Accident radiation exposure
- Accelerated thermal aging
- Other aging simulation as applicable
- Post-aging functional test
- Accident simulation
- Final inspection

3.9.4.5.2 *Qualification by Analysis*

Qualification by analysis requires the construction of a valid mathematical model of the equipment to be qualified, in which the performance characteristics of the equipment are dependent variables, and the environmental influences are the independent variables. The validity of the mathematical model is justified by test data, operating experience, vendor data, and established engineering principles that support the analytical assumptions and conclusions.

Consistent with CSA N290.13 (Reference 3.9-9), the qualification of complex equipment by analysis only is not used because of the great difficulty in developing an accurate analytical model, unless it can be justified that using only analysis is sufficient.

3.9.4.5.3 Qualification by Operating Experience

Qualification by use of operating experience requires documented data to be available confirming that the product providing the operating experience is identical or justifiably similar to the equipment to be qualified, the product providing the operating experience has operated under service conditions which equal or exceed, in severity, the service conditions and performance requirements for which the product is to be qualified, and the installed product must, in general, be removed from service and subjected to partial type testing to include the DBA environments for which the product is to be qualified. Operating experience may also provide information on limits of extrapolation, failure modes, and failure rates.

3.9.4.5.4 Combined Qualification

Equipment may be qualified by test, analysis, operating experience, or any combination of these methods. Combined qualification may be used to supplement existing test data. Partial type testing may be augmented by tests of components where size, applications, time, or other test limitations preclude the use of a full type test. Examples of combined qualification include separate effect tests with extrapolation or analysis, operating experience with extrapolation or analysis, and type tests supplemented with tests of components and analysis.

3.9.4.5.5 Aging Considerations

Significant aging mechanisms are considered in establishing Environmental Qualification for the specified service conditions and in defining the qualified life of equipment and components. An aging mechanism is significant if subsequent to manufacture, while in storage, and/or in the normal and abnormal service environment, it results in degradation of the equipment that progressively and appreciably renders the equipment vulnerable to failure to perform its safety category function under harsh environmental DBA conditions. These typically include thermal, radiation, and operation induced degradation. Age conditioning is used during qualification to simulate these effects.

Accelerated thermal aging is used to simulate the deterioration due to temperature during the normal service life of equipment. The use of the Arrhenius Equation is the recognized method.

The effects of radiation are simulated during qualification testing for equipment exposed to radiation in normal or accident conditions. Radiation qualification considers that equipment damage is a function of total integrated dose and can be influenced by dose rate, energy spectrum, and particle type. The radiation qualification includes doses from all potential radiation sources at the equipment location. The assessment of accelerated aging effects due to normal radiation exposure is performed separately from or included as part of the accident radiation exposure.

Cycle aging conservatively simulates the degradation during the required operating cycles for the equipment. The number of cycles required for equipment is based on the design specification.

For equipment that cannot meet the required cycles for the 60-year life, a shorter qualified life is established, and the effects of physical aging and obsolescence are reflected in the maintenance, surveillance, and replacement program.

Age conditioning considers sequential, simultaneous, and synergistic effects to achieve the worst state of degradation.

Age conditioning is not required for equipment with no determined aging mechanisms.

3.9.4.5.6 Environmental Margins

Margin is applied during Environmental Qualification to account for unquantified uncertainties such as normal variations in equipment production, inaccuracies in measurement and test instrumentation and reasonable errors in defining satisfactory performance. Current qualification practices do not require statistical or reliability data when establishing Environmental Qualification. Instead, conservatism and margins are intended to provide reasonable assurance that the installed equipment can perform as required.

The following margins as recommended in CSAN290.13 (Reference 3.9-9) may be applied to simulated accident conditions during qualification testing or considered when performing qualification by analysis.

The margin applicable to a specific parameter is determined based on the peak conditions as follows:

- Temperature: + 10% of peak temperature to a maximum of 8°C
- Pressure: + 10% of peak gauge pressure to a maximum of 70kPa
- Radiation: + 10% of the total integrated accident dose
- Mission Time: + 10% of the required mission time (up to the maximum)

3.9.4.6 Documentation of Environmental Qualification

Documentation is required to ensure an auditable proof of performance under DBA conditions is developed and maintained for equipment requiring Environmental Qualification. The following subsections provide a general description of the expected information. The organization or format of the documentation is not intended to be prescriptive.

3.9.4.6.1 Equipment Specifications for Environmental Qualification

Plant specific equipment specifications for Environmental Qualification are developed and include essential information about the equipment to be qualified. The following is included as applicable:

- Details of aging stressors resulting from normal environmental conditions
- Details of aging stressors resulting from normal operating conditions
- Details of in-plant configuration, including mounting
- Description of control, indication, and other auxiliary devices required for proper operation
- Functional requirements under the defined normal and accident service conditions
- Required qualified life for the equipment or maintenance intervals for specific components, or both
- Details of DBA stressors resulting from accident environmental conditions
- Details of DBA stressors resulting from accident operating conditions
- Performance requirements and acceptance criteria
- Mission time(s) for relevant safety category functions of equipment
- Provision for condition monitoring

3.9.4.6.2 Qualification Plan

Prior to starting the qualification of equipment, plans are developed detailing the qualification method. If the qualification method is by test, the qualification plan is incorporated into the test plan. The following is included:

- Equipment identification
- Equipment qualification specifications requirements for Environmental Qualification as described above
- Scope of qualification
- Documentation for traceability of equipment and of all polymeric or elastomeric material
- A description of the components of the equipment
- Qualification method selected and justification for the selection of a method if it is other than testing
- When analysis is the chosen method, a description of the analytical methods to be used
- Age conditioning limits/parameters, including qualified life objective, peak aging temperature limits, radiation dose, and condition-based qualification methods, if applicable
- Evaluation of identified synergistic effects

3.9.4.6.3 Test Report

For qualification by test, a test report is developed after the completion of testing.

A test report includes the test plan and provides a detailed summary of the testing performed and the test results to demonstrate the equipment is successfully qualified for the environmental conditions specific to the testing. As a minimum it includes:

- Approved and dated certification sheet
- Identification of equipment tested
- Identification of test specimen
- The range of types or sizes covered
- The qualification requirements
- Results of initial and final inspection
- Description of mounting configuration during testing
- The simulated aging and accident environmental conditions as a function of time
- Results of all functional tests
- A description of the test facility
- A description of the test facility's QA program
- Calibration details for test equipment
- Disposition of any anomalous test results and variance from the test plan
- Details of any maintenance performed

- A summary of the testing program
- A conclusion stating compliance/non-compliance with acceptance criteria and test plan
- Details of connections and interfaces with the tested equipment
- A determination of the qualified life of the equipment under specified service conditions

3.9.4.6.4 Analysis Report

For qualification by analysis, an analysis report is developed providing a detailed summary of the analytical method used (including identification of any software used), calculations performed, and the results to demonstrate the equipment is successfully qualified for the environmental and/or seismic/dynamic condition(s) specified by the analysis.

3.9.4.6.5 Qualification Summary Report

An Environmental Qualification summary report provides documented assurance in an auditable format that equipment requiring Environmental Qualification should function as required under the relevant service conditions for its required mission time. It establishes the basis for equipment configuration, maintenance and procurement requirements providing the means to ensure that Environmental Qualification of the equipment is maintained for the station's life. Information contained in the summary report includes:

1. Equipment identification and description including function, location, mounting and interfaces, any required enclosures/shielding consistent with qualification basis
2. The qualification basis for the equipment including methodology, documentation from testing, analysis, and other supporting documentation supporting qualification
3. An overall conclusion on the qualified status of the equipment, including any limitations on use, operating constraints, or restrictions
4. Identification of any specific maintenance, replacement, and surveillance activities necessary to ensure that the qualification of the equipment is preserved throughout its installed life
5. Identification of any specific procurement requirements necessary to ensure that replacement equipment or components are procured in a manner that is consistent with the qualification basis
6. Identification of any handling and storage requirements

3.9.5 Electromagnetic Compatibility

Accepted industry codes and standards are applied to establish an electromagnetic compatible environment applicable to electrical and I&C equipment. EMC qualification involves two elements:

1. Testing to assess susceptibility of equipment to interference levels that bound the expected electromagnetic environment
2. Testing to assess emissions of equipment to ensure that the contribution to the electromagnetic environment does not invalidate bounding interference levels applied for susceptibility testing

Susceptibility testing allows assessment of equipment immunity to Electromagnetic and Radio-Frequency Interference (EMI/RFI) and confirmation of its Surge Withstand Capability. Emissions testing provide assurance that equipment is compatible with the expected electromagnetic environment.

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Consistent with CNSC REGDOC-2.5.2 (Reference 3.9-1), EMI/RFI is addressed through recognized industry standards. NRC Reg Guide 1.180 (Reference 3.9-12) provides appropriate guidance for the EMC testing, describing methods and procedures considered acceptable for demonstrating EMC compliance based on the endorsement of IEC standards IEC 61000-2 / 4 (References 3.9-13 and 3.9-14), Military Standards MIL-STD (Reference 3.9-15) EPRI Topical Report TR-102323 (Reference 3.9-16) and IEEE Standard 627 (Reference 3.9-17) for test methods consistent with specific equipment requirements.

Chapter 2, Section 2.2.9 characterizes the site-specific electromagnetic hazards for which the design must consider and for which EMC qualification must address.

Chapters 7 and 8 describe the design of the I&C systems and the Electrical systems, respectively. As part of the design process, layout strategies are developed to ensure that the design considers interaction between SSC, and as the design is constructed, elements such as grounding and shielding are incorporated to meet the EMC/EMI standards (prior to testing).

Chapters 7, Subsections 7.3.1.3.1, 7.3.2.3.1, and 7.3.4.3.1 discuss design and quality measures for I&C systems as they relate to qualification measures that confirm I&C systems and equipment are capable of reliably performing the design basis functions for which they are credited over the range of environmental conditions postulated for the plant state and for the area in which they are located. Chapter 7, Table 7.1-1 provides System and Equipment standards to be followed in the design that ensures qualification measures are applied.

Chapter 8, Section 8.6 provides electrical system design information on grounding and EMC. Chapter 8, Section 8.1.1.2 describes how electrical systems are designed to accommodate grid disturbances. The electrical design includes considerations for the environmental conditions postulated for plant states in the areas in which components are located and credited to function.

The standards referenced provide detailed test conditions to ensure equipment is tested in the environments in which they are expected to function and provide post-installation practices for maintaining qualification including handling and storage requirements.

3.9.6 Specific Equipment Requirements

Specific equipment categories may have additional requirements not applicable generically across all qualification programs. The Electrical and I&C equipment must meet the guidance provided in CSA N289 series (References 3.9-2 through 3.9-6) and the CSA N290 series standards (Reference 3.9.18 through Reference 3.9-22).

3.9.6.1 Mechanical Equipment

Safety Class mechanical equipment, which has the sole safety category function of maintaining pressure integrity, and which is designed, fabricated, and qualified consistent with ASME Boiler and Pressure Vessel Code, Section III (Reference 3.9-23), is considered qualified as specified in CSA N290.13 (Reference 3.9-9).

Mechanical equipment can be qualified by presenting historical performance data if it is demonstrated that the equipment satisfactorily sustains dynamic loads which are equal to or greater than those specified for the equipment and that the equipment performs a function equal to or better than that for which it is specified.

For mechanical equipment where the loading under normal service is more severe than loading under DBA, then the loading under normal service must be considered in addition to the loading under DBA by test and/or analysis.

For mechanical equipment, the loading and capability under DBA conditions is analyzed in the qualification process to establish the suitability of materials, parts, and equipment needed for

safety category functions, and to verify that the design of such materials, parts, and equipment is adequate.

The qualification of mechanical equipment includes, as applicable, materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms), required operating time, non-metallic subcomponents of such equipment, the environmental conditions and process parameters for which this equipment must be qualified, non-metallic material capabilities, and the evaluation of environmental effects.

In addition, the qualification guidance provided in ASME QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants, (Reference 3.9-24), is considered for qualification of active mechanical equipment. Mechanical pipe supports of SC1 equipment that are susceptible to environmental degradation are seismically and environmentally qualified.

3.9.6.2 Electrical Equipment

Additional qualification guidance is considered for specific electrical equipment, if applicable, as follows:

- SC1 Batteries and their supporting element – IEEE 535 (Reference 3.9-25)
- SC1 Transformers IEEE 638 – (Reference 3.9-26)
- Static battery chargers and inverters – IEEE 650 (Reference 3.9-27)
- Electric penetration assemblies – IEEE 317 (Reference 3.9-28)
- SC1 Actuators – IEEE 382 (Reference 3.9-29)
- SC1 Continuous duty motors – IEEE 334 (Reference 3.9-30), as endorsed by Reg Guide 1.40 (Reference 3.9-31)
- SC1 Motor Control Centers (MCCs) – IEEE 649 (Reference 3.9-32)
- For the electrical equipment described above, excluding motors, the EMC qualification guidance provided in Reg Guide 1.180, (Reference 3.9-17) is considered

3.9.6.3 Instrumentation & Control Equipment

Additional qualification guidance is considered for specific I&C equipment, if applicable. For example, control boards, panels, and racks classified as SC1 components utilize IEEE 420, (Reference 3.9-33) for their qualification program.

Qualification of computer-based I&C systems is in accordance with CNSC REGDOC-2.5.2, (Reference 3.9-1), CSA N290.13 (Reference 3.9-9), and IEEE 7-4.3.2 (Reference 3.9-34) which is consistent with the EMC requirements specified in Reg Guide 1.180 (Reference 3.9-12) and described in Subsection 3.9.5.

When computer based I&C systems environmental type testing is performed:

1. The system under test demonstrates that it functions and performs with safety software that has been validated and verified and is representative of the software to be installed in-service.
2. The testing demonstrates performance of all safety category functions that may be impacted by environmental factors under the environmental service conditions specified in the design specification. Software algorithms, that are tested during verification and validation testing, are not required to be tested unless their outputs exercise different hardware components which may be impacted by environmental conditions.

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3. The testing exercises all portions of the system that are necessary to accomplish the safety category functions and those portions whose operation or failure could impair the safety category functions.
4. The testing confirms the response of digital interfaces and verify that the design accommodates the potential impact of environmental effects on the overall response of the system.

The testing of a complete system is preferred. When testing of a complete system is not practical, confirmation of the dynamic response to the most limiting environmental and operational conditions is based on type testing of the individual modules and analysis of the cumulative effects of environmental and operational stress on the entire system to demonstrate required safety performance.

3.9.6.4 Cables, Raceways, Supports, etc.

For qualification of SC1 cables, the qualification guidance provided in CSA N290.13, (Reference 3.9-10) and IEEE 383 (Reference 3.9-35) are considered.

Supports (hangers) that support trays or conduit that carry safety circuits are designed and analyzed to demonstrate qualification in accordance with IEEE 628 (Reference 3.9-36).

Supports used for Non-Safety Class raceway (conduit and cable tray) in Seismic Category A structures are analyzed to withstand the effects of a DBE and evaluated for seismic interaction as applicable.

SC1 connection assemblies consider the qualification guidance provided in IEEE 572, (Reference 3.9-37) as endorsed by Reg Guide 1.156, (Reference 3.9-38) for their qualification program.

3.9.6.5 Line-Mounted Equipment

Guidance in IEEE 572 (Reference 3.9-37) and IEC/IEEE 60980-344 (Reference 3.9-9.) identifies that special consideration is required for line-mounted (pipe-supported) equipment regarding seismic qualification as the most critical seismic loading condition will occur as a result of the piping or duct system.

Guidance and further clarification for special considerations for line-mounted equipment is provided in IEEE 572 (Reference 3.9-33) and IEC/IEEE 60980-344 (Reference 3.9-8) as well as IEEE 382 (Reference 3.9.10.11-29).

3.9.7 References

- 3.9-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.9-2 CSA N289.1, "General Requirements for Seismic Design and Qualification of Nuclear Power Plants," CSA Group.
- 3.9-3 CSA N289.2, "Ground Motion Determination for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.9-4 CSA N289.3, "Design Procedures for Seismic Qualification of Nuclear Power Plants," CSA Group.
- 3.9-5 CSA N289.4, "Testing procedures for seismic qualification of nuclear power plant structures, systems, and components." CSA Group.
- 3.9-6 CSA N289.5, "Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities," CSA Group.

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- 3.9-7 IEC/IEEE 60980-344, "Nuclear facilities – Equipment important to safety – Seismic qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers.
- 3.9-8 IEC/IEEE 60780-323, "Nuclear facilities – Electrical equipment important to safety – Qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers.
- 3.9-9 CSA N290.13, "Environmental qualification of equipment for nuclear power plants," CSA Group.
- 3.9-10 CSA N290.16, "Requirements for beyond design basis accidents," CSA Group.
- 3.9-11 USNRC Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants."
- 3.9-12 USNRC Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems".
- 3.9-13 IEC 61000-6-2, "Electromagnetic compatibility (EMC) – Part 6-2: Generic standards – Immunity standard for industrial environments," International Electrotechnical Commission.
- 3.9-14 IEC 61000-4, "Electromagnetic Compatibility (EMC) – Part 4: Testing," International Electrotechnical Commission.
- 3.9-15 MIL-STD-461G, "Electromagnetic Interference Characteristics of Equipment," US Department of Defense.
- 3.9-16 EPRI TR-102323, "Guidelines for Electromagnetic Interference Testing in Power Plants," Electric Power Research Institute.
- 3.9-17 IEEE 627, "Standard for Qualification of Equipment Used in Nuclear Facilities", Institute of Electrical and Electronic Engineers.
- 3.9-18 CSA N290.0, "General requirements for safety systems of nuclear power plants," CSA Group.
- 3.9-19 CSA N290.14, "Qualification of digital hardware and software for use in instrumentation and control applications for nuclear power plants," CSA Group.
- 3.9-20 CSA N290.4, "Requirements for reactor control systems of nuclear power plants," CSA Group.
- 3.9-21 CSA N290.7, "Cyber Security for Nuclear Facilities," CSA Group.
- 3.9-22 CSA N290.8, "Technical specification requirements for nuclear power plant components," CSA Group.
- 3.9-23 ASME BPVC-III, "Boiler and Pressure Vessel Code Section III - Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.
- 3.9-24 ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," American Society of Mechanical Engineers.
- 3.9-25 IEEE 535, "Standard for Qualification of Class 1E Vented Lead Acid Storage Batteries for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-26 IEEE 638, "Standard for Qualification of Class 1E Transformers for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.

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- 3.9-27 IEEE 650, "Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-28 IEEE 317, "Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-29 IEEE 382, "Standard for Qualification of Actuators for Power Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants," Institute of Electrical and Electronic Engineers.
- 3.9-30 IEEE 334, "Standard for Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-31 USNRC Regulatory Guide 1.40, "Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants."
- 3.9-32 IEEE 649, "Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-33 IEEE 420, "Standard for the Design and Qualification of Class 1E Control Boards, Panels and Racks Used in Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-34 IEEE 7-4.3.2, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-35 IEEE 383, "Standard for Qualifying Class 1E Electric Cable and Field Splices for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-36 IEEE 628, "Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-37 IEEE 572, "Standard Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
- 3.9-38 USNRC Regulatory Guide 1.156, "Qualification of Connection Assemblies for Nuclear Power Plants."

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Table 3.9-1: Harsh Environment Parameter Conditions

Parameter		Condition
Temperature		10°C above normal ambient and $\geq 50^{\circ}\text{C}$ ⁽¹⁾
Pressure		>4 kPa(g) (or 10%) increase or decrease from normal ambient pressure due to a DBA ⁽²⁾
Humidity		100% Relative Humidity or condensing steam conditions ⁽³⁾
Submergence		Any ⁽⁴⁾
Radiation	Non-electronic equipment	DBA Total integrated accident dose (TIAD) > 170 Gy (17 krad) ⁽⁵⁾
	Electronic equipment	TIAD > 10 Gy (1krad) ⁽⁶⁾
Chemistry		Significant change in chemistry of the ambient environment or operating conditions

- (1) Temperature criteria are based on 10°C as a significant increase in normal ambient temperature added to the typical 40°C ambient temperature rating of most industrial EI&C equipment.
- (2) Typically, pressure change must be coincident with other DBA stressors to be considered harsh.
- (3) If steam is present under normal conditions, it is not a harsh DBA stressor. If condensing humidity condition do not change following a DBA, it is not a harsh DBA stressor.
- (4) Submergence is not harsh if it also occurs under normal operation.
- (5) Based on the radiation threshold of the most radiation-sensitive polymer.
- (6) Based on the radiation threshold of integrated circuits.

3.10 In-Service Monitoring, Tests, Maintenance, and Inspections

3.10.1 Safety Design Bases and Requirements

Ontario Power Generation DNNP-1 Project Quality Plan identifies the controls and describe the quality requirements to be implemented throughout the development of the BWRX 300 SMR project. This Project Quality Plan supplements NEDO 11209-A (Reference 3.10-12), for the execution of GEH design activities that are associated with the BWRX-300 project. NEDO 11209-A has been approved by the U.S. Nuclear Regulatory Commission (NRC). In addition, the CSA Group (CSA) Standard N299 Series (Reference 3.10-7 Thru 3.10-9) defines a consistent set of Canadian quality assurance program requirements for the provision of items and services for nuclear power plants.

The Canadian Nuclear Safety Commission (CNSC) governs the Canadian nuclear industry regulations and has jurisdictional authority. Canadian suppliers comply with CNSC regulations. U.S. based suppliers who export to Canada may request a waiver from U.S. CFRs, RGs, and NUREG and comply with CNSC regulations. In addition, CSA Standards N299.1, N299.2, and N299.3, defines the Canadian quality assurance program requirements for the provision of items and services for nuclear power plants, Categories 1, 2, and 3, respectively.

CNSC REDOC 2.6.1 (Reference 3.10-17), Section 3, is used as guidance for establishment of inspections, tests, modeling, and monitoring programs for the DNNP BWRX-300 Nuclear Power plant. Chapter 13 provides the specific features of the programs.

CNSC REGDOC-2.5.2, Version 1 (Reference 3.10-16) and CNSC REGDOC-2.6.2 (Reference 3.10-18) provide the primary requirements for addressing In-Service Monitoring, Tests, Maintenance, and Inspections.

SSCs that have shorter service lifetimes than the plant lifetime will be identified and described in the design documentation.

Design requirements associated with In-service Monitoring, Tests, Maintenance, and Inspections involve accessibility, ALARA, aging management and easy-removable insulation for inspection, testing, and maintenance. In cases where SSCs are of safety class and cannot be designed to support the desirable testing, inspection, or monitoring schedules, one of the following approaches shall be taken:

1. Proven alternative methods, such as surveillance of reference items or use of verified and validated calculation methods, shall be specified.
2. Conservative safety margins shall be applied, or other appropriate precautions shall be taken, to compensate for possible unanticipated failures.

3.10.2 In-Service Monitoring

The BWRX-300 levels of in-service monitoring for SSC is related to the Defence-in-Depth Defense Levels (DL) that are specified in Section 3.1 and associated classifications of SSCs in Section 3.2. Specifics on In-service monitoring are developed in the other PSAR chapters.

The design provides facilities for monitoring chemical conditions of fluids and of metallic and non-metallic materials. The means for adding or modifying the chemical constituents of fluid streams is specified in Chapter 13, Subsection 13.3.2.3 programmatic requirements for in-service monitoring.

3.10.3 In-Service Testing

IST of certain ASME Boiler and Pressure Vessel Code (BPVC) Section III Division 1 (Reference 3.10-1) pumps, valves, and snubbers (dynamic restraints) as applicable is performed in

accordance with the ASME OM code. In addition, IST is performed in accordance with applicable Canadian Codes and Standards, and IAEA Safety Standards.

Pre-service test results will be documented and used as a baseline for periodic in-service testing.

The design of BWRX-300 structures, systems, and components provides access for the performance of IST to the extent practicable.

The IST Program includes periodic tests and inspections that demonstrate the operational readiness of certain SSC that perform a function in shutting down the reactor to a safe shutdown condition, maintaining a safe shutdown condition, or mitigating the consequences of an accident.

Specific required in-service tests are established in other PSAR chapters involving SSCs.

Chapter 13, Subsection 13.3.2.3, provides programmatic requirements for in-service testing.

3.10.4 In-Service Maintenance

CNSC REGDOC-2.6.2 (Reference 3.10-16) forms the regulatory bases for the requirements of the Canadian Nuclear Safety Commission (CNSC) regarding maintenance programs for nuclear power plants (NPPs). This document also provides information and guidance on how the requirements may be met. The DNNP BWRX-300 Nuclear Power plant will abide by the recommendations of CNSC REGDOC-2.6.2 which are based in part on the following publications:

- CNSC, REGDOC-2.6.1, Reliability Programs for Nuclear Power Plants (Reference 3.10-15).
- CNSC, REGDOC-2.5.2, Version 1, Design of Reactor Facilities: Nuclear Power Plants (Reference 3.10-14).
- CNSC, REGDOC-1.1.2, Licence Application Guide: Licence to Construct a Reactor Facility, Version 2 (Draft) (Reference 3.10-13).
- International Atomic Energy Agency (IAEA), TECDOC-658, Safety Related Maintenance in the Framework of the Reliability Centered Maintenance Concept, Vienna, 1992 (Reference 3.10-10).
- IAEA Safety Standards Series, No. NS-G-2.6, Maintenance, Surveillance, and In-service (Reference 3.10-11).
- CSA N286-12, Management system requirements for nuclear facilities (Reference 3.10-6).

Baseline data will be gathered during initial testing and system commissioning of SSCs.

Chapter 13, Subsection 13.3.3, provides programmatic requirements for in-service maintenance.

3.10.5 In-Service Inspection

Mechanical components and equipment including heat exchangers, pipe supports, pumps, valves, and vessels, that are classified as ASME BPVC III Division 1 Class 1, 2, or 3 are designed and provided with accessible openings for ISI and testing, to justify the operational readiness of components and equipment as set forth within ASME BPVC III- Division 1.

Components and equipment, that require inspections and testing to satisfy ASME BPVC-XI-Division 1 requirements, are examined by appropriate ISI and testing techniques, including ASME BPVC III Division 1, ASME Code OM, CNSC REGDOC-2.5.2, and CNSC REGDOC 2.6.2 required examinations, prior to the component or equipment leaving the manufacturer's facility.

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ASME BPVC-XI-2021, ASME Code OM, CNSC REGDOC-2.5.2, and CNSC REGDOC 2.6.2 inspection and testing requirements do not replace or change ASME BPVC III required examinations.

Nondestructive Examination (NDE) methods are described within ASME BPVC-V (Reference 3.10-2) and ASME BPVC-XI.

Component and equipment procurement specifications provide detailed requirements, which are to be used during the manufacturing phase and installation at the plant site.

Chapter 13, Subsection 13.3.2.3, provides programmatic requirements for ISI.

3.10.6 References

- 3.10-1 ASME BPVC-III, "Boiler and Pressure Vessel Code Section III - Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers.
- 3.10-2 ASME BPVC-V, "Section V - Non-destructive Examination," American Society of Mechanical Engineers.
- 3.10-3 ASME BPVC-XI, "Boiler and Pressure Vessel Code Section XI - Rules for In-Service Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers.
- 3.10-4 ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers.
- 3.10-5 ASME OM, "Operation and Maintenance of Nuclear Power Plants," American Society of Mechanical Engineers.
- 3.10-6 CSA N286-12, "Management System Requirements for Nuclear Facilities," CSA Group.
- 3.10-7 CSA N299.1-16, "Quality Assurance Program Requirements for the Supply of Items and Services for Nuclear Power Plants, Category 1," CSA Group.
- 3.10-8 CSA N299.2-16, "Quality Assurance Program Requirements for the Supply of Items and Services for Nuclear Power Plants, Category 2," CSA Group.
- 3.10-9 CSA N299.3-16, "Quality Assurance Program Requirements for the Supply of Items and Services for Nuclear Power Plants, Category 3," CSA Group.
- 3.10-10 IAEA TECDOC-658, "Safety Related Maintenance in the Framework of the Reliability Centered Maintenance Concept," International Atomic Energy Agency.
- 3.10-11 IAEA Safety Standards Series No. NS-G-2.6, "Maintenance, Surveillance, and In-service Inspection in Nuclear Power Plants," International Atomic Energy Agency.
- 3.10-12 NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.10-13 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 3.10-14 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.10-15 CNSC Regulatory Document REGDOC-2.6.1, "Reliability Programs for Nuclear Power Plants."
- 3.10-16 CNSC Regulatory Document REGDOC-2.6.2, "Maintenance Programs for Nuclear Power Plants."

3.11 Compliance with National and International Standards

Chapter 1, Appendix B Tables B1.11 through B1.11-3 Conformance with Applicable Regulations, codes, and standards, describes the applicable CNSC Regulatory documents, codes and standards used in the design of the OPG DNNP BWRX-300 plant. CNSC REGDOC 1.1.2 Draft Version 2 and CNSC REGDOC 2.5.2 Draft Version 2 form the basis of the Canadian regulatory requirements. The CSA Group (CSA) standards form the detailed bases of code and standard methodology to comply with the regulatory requirements and compared to the standards (both National and International) used in the BWRX-300 design. Many CSA standards refer to the use of U.S. codes in the design of Canadian Nuclear Plants. Alternative codes, standards, and methodology not addressed by CSA standards are reviewed against CNSC REGDOC requirements and justified through a design assessment process for use. Chapter 17 on Safety in Design discusses the overall design process.

As stated in Chapter 1, section 1.11, CNSC Regulatory Documents, applicable IAEA and U.S. regulatory documents, and industry codes and standards used in the OPG BWRX-300 design, grouped by Safety and Control Area (SCA), are listed in Appendix B Tables B1.11-1 through 1.11-3. These tables represent all 14 SCAs that form the bases for CNSC safety reviews. The tables list the codes and standards by the organization that represents the applicability to design type such as Mechanical, Electrical, Civil, Nuclear I&C and others. The tables clarify any specific details associated with the code and/or standard use.

The specific PSAR chapters provide prescriptive details that related to the BWRX-300 design features and their alignment with Canadian regulations including compliance with both national and international standards. Chapter 3, Safety Objectives and Design Rules for Structures, Systems and Components forms the majority of requirements for other chapters used in the design of the DNNP BWRX-300 new nuclear plant.

3.11.1 References

- 3.11-1 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 3.11-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

APPENDIX 3A – PRELIMINARY CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

3.12 Introduction

The BWRX-300 approach to classifying Structures Systems and Components (SSC) is consistent with IAEA SSR-2/1, "Safety of Nuclear Power Plants: Design" (Reference 3-12-1) and IAEA SSG-30, Safety Classification of Structures, Systems, and Components in Nuclear Power Plants," (Reference 3.12-2) and aligns with CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants," Section 7.1 (Reference 3.12-3). Classification of SSC is conducted to identify the importance of the SSC with respect to safety.

The methodology for classification of BWRX-300 SSC is discussed in Section 3.2. in accordance with:

- Safety Class (SC)
- Seismic Category
- Quality Group

Table 3.12-1 provides a preliminary list of the principal BWRX-300 components organized by system. Classification of Structures is presented in Section 3.3, Table 3.3-1.

3.12.1 References

- 3.12-1 IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design" International Atomic Energy Agency.
- 3.12-2 IAEA Safety Standards Series No. SSG-30, "Safety Classification of Structures, Systems, and Components in Nuclear Power Plants," International Atomic Energy Agency.
- 3.12-3 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 3.12-4 NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description," GE-Hitachi Nuclear Energy Americas, LLC.
- 3.12-5 ISO 9001, "Quality Management Systems - Requirements," International Organization for Standardization."
- 3.12-6 CSA N286-12, "Management System Requirements for Nuclear Facilities," CSA Group.
- 3.12-7 USNRC Regulatory Guide 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors."
- 3.12-8 USNRC NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue of Reactor Materials."
- 3.12-9 10 CFR 21, "Reporting of Defects and Noncompliance."
- 3.12-10 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."
- 3.12-11 10 CFR 20.1201, "Occupational dose limits for Adults."

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- 3.12-12 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 3.12-13 CSA N288.2, "Guidelines for Calculating the Radiological Consequences to the Public of a Release of Airborne Radioactive Material for Nuclear Reactor Accidents," CSA Group.
- 3.12-14 CSA N288.1:14, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities," CSA Group.
- 3.12-15 USNRC NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning."
- 3.12-16 USNRC IN96-39, "Estimates of Decay Heat Using ANS 5.1 Decay Heat Standard May Vary Significantly."
- 3.12-17 ANSI/ANS-5.1, "American National Standard Decay Heat Power in Light Water Reactors," American Nuclear Society.
- 3.12-18 ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers.
- 3.12-19 ASME BPVC-III APP, "Section III - Rules for Construction of Nuclear Facility Components - Appendices," American Society of Mechanical Engineers.
- 3.12-20 ASME BPVC-III Code Case N-411-1, "Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1, ERRATA SUP 13," American Society of Mechanical Engineers.

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Table 3.12-1: Preliminary BWRX-300 Classification List

Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
NUCLEAR STEAM SUPPLY SYSTEMS				
<i>Nuclear Boiler System</i>				
Reactor pressure vessel	SC1	SCCV	A	A
Main Steam (MS), Head Vent, Isolation Condenser System (ICS), Feed Water (FW), and Reactor Water Cleanup System (CUW) Reactor Isolation Valves (RIV)	SC1	SCCV	A	B
Core Support Structures: <ul style="list-style-type: none"> • Shroud • Chimney • Core Support Ring and Legs (Shroud Support) • Core Plate (and Core Plate Hardware) • Top Guide (and Top Guide Hardware) • Orifice Fuel Supports and Peripheral Fuel Supports • Control Rod Guide Tubes (CRGTS) • Non-Pressure Boundary Portion of Control Rod Drive Housings (CRDHs) 	SC1	SCCV	B	A
Internal Structures: <ul style="list-style-type: none"> • Nuclear Instrumentation In-Core Guide Tubes • Non-Pressure Boundary Portion of In-Core Housings 	SC1	SCCV	B	A
Internal Structures: <ul style="list-style-type: none"> • Chimney Head and Steam Separator Assembly • Steam Dryer Assembly • Feedwater Spargers • Head Vent Internal Piping • CUW Suction Piping • Nuclear Instrumentation In-Core Guide Tube Stabilizers • ICS Return Internal Piping 	SC3	SCCV	B	NS
Surveillance Assembly (Sample Holders)	SCN	SCCV	NA	NS
Nuclear Instrumentation Dry Tube	SC1	SCCV	A	A
Nuclear Instrumentation Housings, Flanges and Ceramic Plugs	SC1	SCCV	A	A

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Pressure Boundary Portion of Control Rod Drive Housings	SC1	SCCV	A	A
Control Rods	SC1	SCCV	NA	B
Reactor Pressure Vessel (RPV) Support - Refueling Bellows	TBD	SCCV	TBD	TBD
RPV Stabilizers	SC1	SCCV	A	A
RPV Support Skirt	SC1	SCCV	A	A
Main Steam piping from the Reactor Isolation Valve to the outboard MS Containment Isolation Valve	SC1	SCCV	B	A
Outboard MS Containment Isolation Valves	SC1	RB	B	B
RPV Level Instrumentation Sensing Line including pressure retaining parts of instrumentation located on these lines	SC1	RB	B	A
MS line piping and components from outside the CIV to the Seismic Interface Restraint	SC1	RB	B	A
MS Seismic Interface Restraint	SC1	RB	B	A
MS line piping and components from the Seismic Interface Restraint (SIR) to the Condensate and Feedwater System, Main Turbine Equipment, Moisture Separator Reheater System, Turbine Bypass System, and Main Condenser and Auxiliaries components	SC3	TB	D	NS
MS line leak detection instrumentation in Reactor Building	SC1	RB	NA	B
MS line leak detection instrumentation in Turbine Building	SC1	TB	NA	NS
RPV Head Vent piping to MSL	SC1	SCCV	B	A
RPV Head Vent piping to Quench Tank Isolation Valve	SC1	SCCV	B	A
Quench Tank Isolation Valves	SC1	SCCV	B	B
RPV Head Vent piping from Quench Tank Isolation Valve to Quench Tank	SC3	SCCV	D	NS
Quench Tank	SC3	SCCV	D	NS
Head Vent Quench Tank Vacuum Breaker	SC3	SCCV	D	NS
O-Ring Seal Leak Detection piping up to Pressure Transmitter	SC3	SCCV	B	A See Note 8
O-Ring Seal Leak Detection piping to O-Ring Seal Leak Detection Manual Isolation Valve	SC3	SCCV	B	A See Note 8

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
O-Ring Seal Leak Detection Isolation Valves	SC3	SCCV	B	B See Note 8
O-Ring Seal Leak Detection Isolation Valve piping to Quench Tank	SC3	SCCV	D	NS
Other Nuclear Boiler System (NBS) mechanical / instrumentation ASME Section III pressure boundary components on the MS Lines	SC1	RB	B	A
Other NBS mechanical / instrumentation ASME B31.1 pressure boundary components on the MS Lines	SC3	TB	D	NS
INSTRUMENTATION AND CONTROL SYSTEM				
SC1 Instrumentation and Control System	SC1	RB and CB	NA	B
<i>SC2 and 3 Instrumentation and Control System</i>				
Equipment that supports DL2 functions	SC3	RB, TB, and CB	NA	NS
Equipment that supports DL4a functions	SC2	RB, TB, and CB	NA	NS
Equipment that supports DL4b functions	SC3	TBD	NA	NS
Non-Safety Instrumentation and Control System	SCN	RB, TB, and CB	NA	NS
RADIATION MONITORING SYSTEMS				
Process Radiation and Environmental Monitoring System				
<i>Process Radiation and Environmental Monitoring System, Process Radiation Monitoring Subsystem</i>				
In-line (external) radiation monitoring equipment (supporting PAM Type E variables)	SC3	RB, TB, CB, RWB	NA	NS
Off-line (process stream) radiation monitoring equipment (supporting PAM Type E variables)	SC3	RB, TB, CB, RWB	D	NS
<i>Process Radiation and Environmental Monitoring System, Area Radiation Monitoring Subsystem</i>				
Refueling Floor radiation monitors supporting Defense Line 2 functions (supporting PAM Type E variables)	SC3	RB	NA	NS
General Area radiation monitors (supporting PAM Type E variables)	SC3	RB, TB, CB, RWB	NA	NS
<i>Process Radiation and Environmental Monitoring System, Containment Monitoring Subsystem</i>				
CIVs and inboard process piping	SC1	RB	B	B

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Containment hydrogen and oxygen monitoring equipment (including process piping outboard of CIVs) (supporting PAM Type C and F variables)	SC3	RB	D	B
Containment fission product monitoring equipment (including process piping outboard of CIVs)	SC3	RB	D	NS
Containment water level transmitters	SC3	RB	NA	NS
Containment pressure transmitters supporting Defense Line 3 functions (supporting PAM Type C and D variables)	SC1	RB	NA	B
Containment pressure transmitters supporting Defense Line 4a functions	SC2	RB	NA	NS
Containment temperature transmitters (supporting PAM Type D variables)	SC3	RB	NA	B
Containment area radiation monitors (supporting PAM Type C and E variables)	SC3	RB	NA	B
Containment relative humidity transmitters	SCN	RB	NA	NS
<i>Process Radiation and Environmental Monitoring System, Process Sampling Subsystem</i>				
Non-pressure boundary sampling equipment	SCN	RB, TB, RWB	NA	NS
Pressure boundary sampling equipment (non-contaminated)	SCN	RB, TB, RWB	D	NS
Pressure boundary sampling equipment (contaminated)	SC3	RB, TB, RWB	D	NS
CORE COOLING SYSTEMS				
Isolation Condenser System				
Steam supply, condensate return, standby gas purge piping	SC1	SCCV	A	A
Shutdown Cooling System (SDC) interface piping to containment isolation valve, A and B trains	SC1	SCCV, RB	A	A
Boron Injection System (BIS) interface piping to BIS interface valve, C train	SC1	SCCV	A	A
SDC interface piping from containment isolation valve to downstream redundant isolation valve, A and B trains	SC1	RB	A	A
ICS pools atmospheric vent piping	SC1	RB	B	A
Outer pool to inner pool cross-connect piping	SC1	RB	B	A

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Long-term ICS pool makeup piping (also referred to as flex-makeup piping)	SC3	RB	D	NS
Isolation Condensers (Inside Containment Boundary)	SC1	SCCV, RB	A	A
Isolation Condensers (Outside Containment Boundary)	SC1	RB	B	A
All condensate return valves: Subcomponents supporting pressure boundary	SC1	SCCV	A	A
Open/Close condensate return valves: Subcomponents supporting function to open and remain open	SC1	SCCV	NA	B
Open/Close condensate return valves: Subcomponents supporting function to close and remain closed	SC3	SCCV	NA	NS
Throttling condensate return valves: Subcomponents supporting function to fully open and remain fully open	SC2	SCCV	NA	NS
Throttling condensate return valves: Subcomponents supporting function to throttle, to close, and remain close	SC3	SCCV	NA	NS
Standby gas purge valves: Subcomponents supporting pressure boundary	SC1	SCCV	A	A
Standby gas purge valves: Subcomponents supporting function to close and remain closed	SC1	SCCV	NA	B
Standby gas purge valves: Subcomponents supporting function to open and remain open	SC3	SCCV	NA	NS
Containment isolation valves to SDC system, A and B trains: Subcomponents supporting pressure boundary function	SC1	RB	A	A
Containment isolation valves to SDC system, A and B trains: Subcomponents supporting function to close and remain closed	SC1	RB	NA	B

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Containment isolation valves to SDC system, A and B trains: Subcomponents supporting function to open and remain open	SC3	RB	NA	NS
Redundant isolation valves to SDC system, A and B trains: Subcomponents supporting pressure boundary function	SC1	RB	A	A
Redundant isolation valves to SDC system, A and B trains: Subcomponents supporting function to close and remain closed	SC1	RB	NA	B
Redundant isolation valves to SDC system, A and B trains: Subcomponents supporting function to open and remain open	SC3	RB	NA	NS
Outer pool to inner pool cross-connect backflow prevention devices Subcomponents supporting pressure boundary function	SC1	RB	B	A
Outer pool to inner pool cross-connect backflow prevention devices Subcomponents supporting active functions	SC1	RB	NA	B
Flow detection impulse piping and inline passive pressure boundary components	SC1	SCCV, RB	B	A
Flow detection impulse piping excess flow check valve	SC1	RB	B	B
Flow detection differential pressure instrumentation=	SC1	RB	NA	B
Wide range pool level instrumentation used for post-accident monitoring, long term (>72 hours)	SC3	RB	NA	NS
All piping installed temperature instrumentation, pool temperature instrumentation and narrow range pool level instrumentation used for Operating Limits and Conditions monitoring only	SC3	RB, SCCV	NA	NS

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Pneumatic supply tubing and components from the actuator to the control solenoid valves for the open/closed only condensate return valves, containment isolation valves and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC1	RB, SCCV	NA	A
Hydraulic supply tubing and components from the actuator to the control solenoids valves for the throttling condensate return valves	SC2	SCCV	NA	NS
Control solenoid valves for the open/closed only condensate return valves, containment isolation valves, and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC1	RB, SCCV	NA	B
Control solenoid valves for the throttling condensate return valves	SC2	SCCV	NA	NS
Pneumatic supply tubing and components from the interface point with Plant Pneumatics System to the control solenoid valves for the open/closed only condensate return valves	SC3	SCCV	NA	NS
Pneumatic supply tubing and components from the interface point with Plant Pneumatics to the control solenoid valves for the containment isolation valves and the redundant downstream isolation valves in the interface lines to SDC, Trains A and B	SC3	RB	NA	NS
Hydraulic supply tubing from the positioner to the control solenoid valves for the throttling condensate return valves	SC3	SCCV	NA	NS
REACTOR SERVICING EQUIPMENT				
<i>Refueling Equipment and Servicing</i>				
Refueling Platform	SC3	RB	NA	A See Note 8
Fuel Storage Racks	SC3	RB	NA	A See Note 8
Miscellaneous Servicing Equipment	SCN	RB	NA	NS
REACTIVITY CONTROL				
<i>Boron Injection System</i>				
Injection Pump	SC3	RB	D	NS

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Injection Pump Motor	SC3	RB	NA	NS
Storage Tank	SC3	RB	D	NS
Test Tank	SCN	RB	NAD	NS
Instrumentation – Tank Level, Solution Temperature, Discharge Pressure, Flow Rate	SC3	RB	D	NS
Piping from Tank to Pumps	SC3	RB	D	NS
Piping from Pumps to Outboard Containment Isolation Valve	SC3	RB	D	NS
Injection / Containment Isolation Valves	SC1	RB/SCCV	A	B
Containment Pipe Penetration	SC1	RB/SCCV	A	A
Piping from Containment Penetration to IC return line	SC1	SCCV	A	A
Piping and Valves with no SC function	SCN	RB	D	NS
Control Rod Drive System/High Pressure Injection				
Non-pressure retaining Fine Motor control Rod Drive (FMCRD) scram subcomponents	SC1	SCCV	NA	B
FMCRD RCPB subcomponents except flange ball check valve	SC1	SCCV	A	A
FMCRD Flange Ball Check Valve	SC1	SCCV	A	B
FMCRD Motor	SC2	SCCV	NA	NS
FMCRD separation switches	SC3	SCCV	NA	NS
FMCRD Position Indication Probe with Switches	SC3	SCCV	NA	NS
Hydraulic control unit (HCU) Nitrogen Tank	SC1	RB	B	A
HCU Scram Valve	SC1	RB	B	B
HCU accumulator	SC1	RB	B	B
HCU Scram Solenoid Valve Assembly	SC1	RB	NA	B
HCU Instrument manifold pressure boundary components	SC1	RB	B	A
ARI Valves	SC2	RB	NA	NS
HCU piping and piping between HCU and FMCRD	SC1	RB	B	A
Charging Water piping and valves (except when directly above HCU), pump discharge, drive header, and other piping not part of HCU)	SC3	RB	D	NS
Charging Water Piping and Valves (directly above HCU)	SC3	RB	D	NS

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Purge Water Piping and Valves (except when directly above HCUs)	SC3	RB	D	NS
Purge Water Piping and Valves (directly above HCUs)	SC3	RB	D	NS
Control Rod Drive (CRD) charge pumps	SC3	RB	D	NS
CRD Purge Pumps	SC3	RB	D	NS
CRD Purge FCVs	SC3	RB	D	NS
DECAY HEAT REMOVAL				
ICS Pool Cooling and Cleanup System (ICC)				
Suction Surge Tank Return Guard Pipe	SC1	RB	B	A
All other system piping and components located in RB 1650 Piping (including valves and instrumentation), Pumps/ASDs, HXs, Demineralizer, Dosing Pot	SCN	RB	D	NS
All other components located in ICS pools, including piping, anti-siphon devices, and distribution spargers)	SCN	RB	D	NS
Shutdown Cooling System				
Pump	SC3	RB	D	NS
Heat Exchanger	SC3	RB	D	NS
Leak Detection Equipment supporting Safety Category 1 functions	SC1	RB	C	B
Leak Detection Equipment supporting Safety Category 2 functions	SC2	RB	D	NS
Decay Heat Removal Piping/Valves/ etc.	SC3	RB	D	NS
Overboard Piping/Valves/etc.	SC3	RB/TB	D	NS
Reactor Water Cleanup System				
Heat Exchanger	SC3	TB	D	NS
RB flow element supporting Safety Category 1 and 2 functions	SC1	RB	B	A
RB leak detection instrumentation supporting Safety Category 1 functions	SC1	RB	NA	B
RB leak detection instrumentation supporting Safety Category 2 functions	SC2	RB	NA	B See Note 8
TB flow elements supporting Safety Category 1 and 2 functions	SC1	TB	C	NS See Note 7

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
TB leak detection instrumentation supporting Safety Category 1 functions	SC1	TB	NA	NS See Note 7
TB leak detection instrumentation supporting Safety Category 2 functions	SC2	TB	NA	NS
Piping/Valves/ etc. from RIV to outboard containment isolation valve	SC1	SCCV/RB	B	B
Piping/Valves/ etc. outboard of outer containment isolation valve	SC3	RB/TB	D	NS
Pressure Reduction Station	SC3	TB	D	NS
Fuel Pool Cooling and Cleanup System (FPC)				
General System Piping and Valves	SC3	RB	D	NS
Off-Normal Makeup Piping and Valves	SC3	RB	D	NS
Surge Tanks	SC3	RB	D	NS
Pumps	SC3	RB	D	NS
Filter Elements	SC3	RB	D	NS
Deep Mixed Bed Demineralizers and Service Piping	SC3	RB	D	NS
Heat Exchangers	SC3	RB	D	NS
NUCLEAR FUEL				
Nuclear Fuel Supply	SC1	SCCV, RB	NA	A
RADIOACTIVE WASTE MANAGEMENT SYSTEMS				
Liquid Waste Management System (LWM)				
LWM Equipment	SC3	RB, RWB, TB	D	NS
LWM containment penetration & locked closed isolation valves relied upon for passive pressure integrity in Defense Line 3.	SC1	SCCV	B	A
Solid Waste Management System (SWM)				
SWM Equipment	SC3	RWB	D	NS
Spent Resin Tank	SC3	RWB	D	NS
Sludge Tank	SC3	RWB	D	NS
Offgas System (OGS)				
TB Piping and Valves	SC3	TB	D	NS
Offgas Recombiner	SC3	TB	D	NS
Cooler Condenser	SC3	TB	D	NS
Moisture Separator	SC3	TB	D	NS
Refrigeration Dryers	SC3	TB	D	NS

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Gas Analyzers	SC3	TB	D	NS
RWB Piping and Valves	SC3	RWB	D	NS
Offgas Reheater	SC3	RWB	D	NS
Charcoal Vault / Adsorber Tanks	SC3	RWB	D	NS
Offgas HEPA Filter	SC3	RWB	D	NS
POWER CYCLE SYSTEMS				
Condensate and Feedwater Heating System				
All passive components from the Seismic Restraint near the RB wall to the FW Reactor Isolation Valves	SC1	RB	B	A
Containment isolation valves and system isolation valves for SDC and OLNC.	SC1	RB, SCCV	B	B
Differential Pressure Measurement for Feedwater Leak Detection	SC1	TB	B	NS See Note 7
Components supporting the detection of loss of feedwater	SC1	TB	B	NS See Note 7
System components in the FW flow path from the Condenser interface to the Seismic Restraint near the RB wall	SC3	TB	D	NS
System components in the FW Heater drain path to the condenser	SC3	TB	D	NS
All other system equipment	SCN	TB	D	NS
Condensate Filters and Demineralizers System				
Filters, demineralizers, bypass lines, valves, and related components	SC3	All	D	NS
All other system equipment	SCN	All	D	NS
Main Turbine Equipment				
Main Turbine Equipment and Subsystem	SC3	TB	D	NS
Non-Return Valves	SC3	TB	D	NS
Moisture Separator Reheater System				
Moisture Separator Reheater and associated components supporting drains to Feedwater Heaters	SC3	TB	D	NS
Components supporting steam supply to MSR (Tube and Shell) and the LP Turbines	SC3	TB	D	NS
All other system components	SCN	TB	D	NS
Turbine Bypass System				
Components supporting Turbine bypass	SC3	TB	D	NS

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
All other system equipment	SCN	TB	D or NA	NS
<i>Generator and Exciter</i>				
Generator and Exciter System	SC3	TB	NA	NS
Neutral Grounding Transformer	SCN	TB	NA	NS
Neutral Grounding Resistor	SCN	TB	NA	NS
Automatic Voltage Regulator Cabinet	SC3	TB	NA	NS
Excitation Cabinet	SC3	TB	NA	NS
<i>Main Condenser and Auxiliaries</i>				
Components relied upon for measuring main condenser vacuum (pressure) in support of Defense Line 4a functions.	SC2	TB	D	NS
Components relied upon for measuring main condenser vacuum (pressure) in support of Defense Line 2 functions.	SC3	TB	D	NS
All components associated with: The requirement for MCA to provide the heat sink to condense reactor steam or drainage from the FW heaters and other steam supply users.	SC3	TB	D	NS
All components associated with: The requirement for MCA to provide a means to draw a vacuum and remove non-condensable gases from the condenser shell.	SC3	TB	D	NS
All remaining components not associated with the functions above.	SCN	TB	D	NS
<i>Circulating Water System</i>				
All components associated with: The requirement for CWS to reject heat from the MCA to the environment through the NHS.	SC3	TB, OO	D	NS
All components associated with: The requirement for CWS to reject heat from PCW to the environment through the NHS.	SC3	TB, OO	D	NS
All remaining components not associated with the functions above	SCN	TB, OO	D	NS
<i>STATION AUXILIARY SYSTEMS</i>				
<i>Chilled Water Equipment</i>				
Components supporting HVAC for post-shutdown I&C equipment	SC3	RB, CB	D	NS
Piping and valves inside containment that support containment cooling	SC3	SCCV, RB	D	NS

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Containment penetration, containment isolation valves, and piping between the CIVs	SC1	SCCV, RB	B	A/B
Air-Cooled chillers, expansion tanks, chiller pumps, and air separators	SC3	RWB	D	NS
Glycol Auto Fill Unit, and Chemical Bypass Unit	SCN	RWB	D	NS
Components support HVAC for non-safety equipment	SCN	ALL	D	NS
Plant Cooling Water System				
Components associated with makeup water supply to the surge tanks and ICS Pools and cleanup heat exchangers.	SCN	ALL	D	NS
All other system equipment	SC3	ALL	D	NS
Plant Pneumatics System				
Containment Penetrations & Isolation Valves	SC1	SCCV, RB	B	A/B
All other system equipment	SC3	ALL	D	NS
Hydrogen Water Chemistry				
All system equipment	SCN	TB	D	NS
Zinc Injection Passivation				
All system equipment	SCN	TB	D	NS
STATION ELECTRICAL SYSTEMS				
SC1 Electrical Distribution System				
All System Equipment	SC1	RB	NA	A
Standby Electrical Distribution System				
SC2 Components	SC2	RB, CB	NA	NS
SC3 Components	SC3	ALL	NA	NS
Normal Electrical Distribution System				
SC3 Components	SC3	ALL	NA	NS
All System Equipment	SCN	ALL	NA	NS
POWER TRANSMISSION SYSTEM				
Switchyard				
All system equipment	SCN	Switchyard	NA	NS
CONTAINMENT AND ENVIRONMENT CONTROL SYSTEMS				
Primary Containment				

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
Steel-plate Composite Containment Vessel, including all hatches and seals (such as containment closure head and airlocks) relied upon for passive pressure integrity in Defense Line 3.	SC1	SCCV, RB	B	A
All Containment Penetrations	SC1	SCCV	B	A
Refueling Bellows Seal	TBD	SCCV	TBD	TBD
LRT piping and locked closed containment isolation valves relied upon for passive pressure integrity in Defense Line 3.	SC1	RB	B	A
Passive Containment Cooling System	SC1	SCCV, RB	B	A
PCCS Containment Isolation Valves	SC1	RB	B	B
Containment Inerting System				
Containment Pipe Penetrations	SC1	SCCV	B	A
CIVs, Rupture Disc, Check Valve, and Associated Piping	SC1	RB	B	A/B
Sparger Piping	SC3	RB	D	NS
All other system equipment and piping	SC3	RG, RWB, OO	D	NS
Containment Cooling System				
Drain valves	SCN	SCCV	D	NS
All other system equipment	SC3	SCCV	D	NS
STRUCTURE AND SERVICING SYSTEMS				
Cranes, Hoists, and Elevators				
All system equipment	SCN	ALL	NA	NS
Heating Ventilation and Cooling System				
MCR Emergency HVAC	TBD	CB	NA	NS
SCR Emergency HVAC	TBD	RB	NA	NS
RB DCIS Rooms and SCR Fan Coil Units (FCU)	TBD	RB	NA	NS
Defense Line 2 FCUs	SC3	CB	NA	NS
Defense Line 4a FCUs	SC2	CB	NA	NS
RB Refueling Floor Isolation Dampers	SC3	RB	NA	NS
All other system equipment	SCN	ALL	NA	NS
Fire Protection System (FPS)				
System components that support DL2 or DL4b functions (Piping, valves and sprinklers)	SC3	ALL	D	TBD

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Principal Component	Safety Class (Notes 1, 5)	Location (Note 2)	Quality Group (Notes 3, 5)	Seismic Category (Notes 4, 5, 7)
All other system equipment	SCN	RB and CB	D	TBD
<i>Equipment and Floor Drain System</i>				
Piping and valves and supports forming part of the containment boundary	SC1	RB	B	A/B
Drain piping and valves, including supports.	SC3	ALL	D	NS
All other general equipment and floor drain system equipment	SC3	ALL	D or NA	NS
Oily waste sump system and other non-radioactive subsystems	SCN	TB	NA	NS
<i>Water, Gas, and Chemical Pads</i>				
Components required to provide standby diesel fuel oil storage and transfer	SC3	ALL	D	NS
All other system equipment	SCN	ALL	D	NS

NOTES:

1. SC determination and methodology is discussed in Subsections 3.2.1 and 3.2.2.
2. Location Codes:
 - a. SCCV: Containment Vessel
 - b. RB: Reactor Building
 - c. TB: Turbine Building
 - d. CB: Control Building
 - e. RWB: Radwaste Building
 - f. OO: Outdoors On-site
 - g. OL: Any Other Location
 - h. ALL: All locations
3. Quality group classifications is discussed in Subsection 3.2.4.
4. Seismic categories are discussed in Subsection 3.2.3. Any items classified as NS are subject to evaluations for Seismic Interaction as discussed in Subsection 3.2.3.1.
5. Structures, systems and components required to be designed in accordance with Radioactive Waste Management requirements from RG 1.143 for Category RW-IIa, shall meet the guidance of NRC Regulatory Guide 1.143, as applied to radioactive waste management systems, with regard to quality, seismic, and quality group requirements.
6. Other abbreviations.
 - a. TBD: To Be Determined – classification information is to be provided later in the BWRX-300 design process
 - b. NA: Not Applicable
7. Components classified as SC1 may be assigned to a Seismic Class lower than A or B provided they are of a fail-safe design such that the failure of those component(s) does not adversely affect the ability to achieve the safety function.
8. Although these components are not SC1, they are seismically qualified because they are credited with monitoring leakage of reactor coolant under the scope of Regulatory Guide 1.45 or are related to handling and storage of used nuclear fuel.

APPENDIX 3B – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF SEISMIC CATEGORY STRUCTURES

3.13 Introduction

This appendix describes the major computer programs used in the analysis and design of the BWRX-300 Seismic Category structures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 3.12-4) that complies with ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 3.12-18) and CSA N286.7-16, "Quality Assurance of Analytical, Scientific, and Design Computer Programs" (Reference 3.12-12).

GEH maintains an ISO 9001:2015, "Quality Management Systems - Requirements," International Organization for Standardization" (Reference 3.12-5) Certificate of Approval by U.S. Lloyd's Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12, "Management System Requirements for Nuclear Facilities" (Reference 3.12-6).

3.13.1 ACS SASSI v4

Description: ACS SASSI is a finite element computer code on the Microsoft Windows PC platforms for performing 3D dynamic soil-structure interaction (SSI) analysis to analyze the effect of seismic ground motions on structures. The analysis is performed in the frequency domain using linear or equivalent-linear material properties for the structure and soil.

Validation: The software is approved for production use under GEH procedure on engineering software for design and analysis software.

Extent of Application: ACS SASSI is used to perform seismic and static SSI and structure-soil-structure-interaction (SSSI) analyses, as applicable.

3.13.2 ANSYS v17

Description: ANSYS, INC. Multiphysics computer program. ANSYS is a general-purpose large-scale finite element analysis computer program and has interactive capabilities. Finite element analysis is a numerical method for analyzing structure, thermal, fluid flow and other physical problems. The analysis method is based on displacement formulation of the finite element method. Typical applications include finding stress, deformation, thermal analysis, and modal analysis with user inputs of geometrical dimensions, element type, material properties, boundary conditions, and loadings.

Validation: The software is approved for production use under GEH procedure on engineering software for design and analysis software.

Extent of Application: This program is used to model the structure and the hydrodynamics within the BWR and perform structural analysis for applicable loads.

3.13.3 Ansys LS-Dyna v2021

Description: Ansys LS-DYNA is an explicit simulation program capable of simulating the response of materials to short periods of severe loading. Its many elements, contact formulations, material models and other controls can be used to simulate complex models with control over all the details of the problem. Ansys LS-DYNA applications include explosion/penetration, impact analysis, and non-linear explicit structural analysis.

Validation: This software is not approved for production use under GEH procedure on engineering software for design and analysis software and requires output verification in accordance with the design process.

Extent of Application: Ansys LS-DYNA is used to analyze BWRX-300 structures for effects of blast loading and aircraft impact.

3.13.4 SSI-StressCoord v1

Description: The STRESS_POST program is an auxiliary program to post-process the ACS SASSI NQA V4 STRESS result binary database. The STRESS_POST program includes an ensemble of STRESS database processing functionalities which were customized for the GEH engineers for application to the BWRX-300 SMR seismic SSI analysis projects. The STRESS_POST customized program is based on specific implementations incorporated in the ACS SASSI NQA V4 User Interface (UI) capabilities, such as the CTVEC and the CTCCV commands, and existing STRESS binary database verification tools used in-house during the development over years of the STRESS module.

Validation: The software is approved for production use under GEH procedure on engineering software for design and analysis software.

Extent of Application: This STRESS_POST Program is used for post-processing the ACS SASSI STRESS binary databases for Integrated RB Walls and Floors in batch mode.

3.13.5 GT STRUDL

3.13.5.1 Description

GT STRUDL® is structural engineering software offering a complete design solution, including 3D CAD modeling and 64-bit high-performance computation solvers into all versions. GT STRUDL includes all the tools necessary to analyze a broad range of structural engineering and finite element analysis problems, including linear and non-linear static and dynamic analysis.

3.13.5.2 Validation

This software is not approved for production use under GEH procedure on engineering software for design and analysis software and requires output verification in accordance with the design process.

3.13.5.3 Extent of Application

GT STRUDL is used to for the structural analysis and design of non-Seismic Category A structures.

APPENDIX 3C – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF MECHANICAL STRUCTURES, SYSTEMS AND COMPONENTS

3.14 Introduction

As discussed in Subsection 3.6.1.1, this appendix describes the major computer programs used in the analysis of mechanical SSC.. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description” (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd’s Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH’s capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

3.14.1 ANSYS v17

Description: ANSYS, INC. Multiphysics computer program. ANSYS is a general-purpose large-scale finite element analysis computer program and has interactive capabilities. Finite element analysis is a numerical method for analyzing structure, thermal, fluid flow and other physical problems. The analysis method is based on displacement formulation of the finite element method. Typical applications include finding stress, deformation, thermal analysis, and modal analysis with user inputs of geometrical dimensions, element type, material properties, boundary conditions, and loadings.

Validation: The software is approved for production use under GEH procedure on engineering software for design and analysis software.

Extent of Application: This program is used to model the structure and the hydrodynamics within the BWR and perform structural analysis for applicable loads.

3.14.2 PBLE v1

Description: Steam Dryer Analysis

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: PBLE calculates the acoustic loads on a steam dryer based on measurements of pressure along the main steam lines or pressures measured directly on the face of the steam dryer. The loads are then used in a finite element model to calculate the stresses in the dryer.

3.14.3 SIMCENTER 3D Acoustics v2022

Description: Used for modeling dryer acoustic loads and instrumentation diagnostics. Simcenter 3D is a unified, scalable, open and extensible environment for 3D CAE with connections to design,

1D simulation, test, and data management. Fast and accurate solvers power structural, acoustics, flow, thermal, motion, and composites analyses, as well as optimization and multi-physics simulation.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent Of Application: SIMCENTER Finite elements acoustic software will be used to model and calculate acoustic wave propagation in fluid (steam, water) mediums.

3.14.4 GT STRUDL

Description: GT STRUDL® is structural engineering software offering a complete design solution, including 3D CAD modeling and 64-bit high-performance computation solvers into all versions. GT STRUDL includes all the tools necessary to analyze a broad range of structural engineering and finite element analysis problems, including linear and non-linear static and dynamic analysis.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: GT STRUDL will be used to perform structural analysis and qualification of supports.

3.14.5 HyperMesh

Description: HyperMesh is the market-leading, multi-disciplinary finite element pre-processor which manages the generation of the largest, most complex models, starting with the import of a CAD geometry to exporting a ready-to-run solver file. With its advanced geometry and meshing capabilities, HyperMesh provides an environment for rapid model generation.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: HyperMesh is a tool which will be used to generate mechanical models for complicated mechanical components. This tool will serve as a pre-processor to build mesh models, no calculations get performed with Hypermesh.

3.14.6 ERSIN v3

Description: Piping Analysis Software. Secondary Response Spectra for control panels, equipment racks, etc.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: ERSIN is used to generate secondary response spectra for pipe and floor mounted equipment. Example applications include control panels, equipment racks, Main Steam Isolation Valves (MSIVs), Safety Relief Valves (SRVs), Hydraulic Control Units (HCUs), et cetera. ERSIN03P software has three input options: 1) card decks, 2) SAP software decks, and 3) PISYS software decks. ERSIN03P can be used with SAP version 4G07P (Ref. 5-1) and PISYS version 08P (Ref. 5-2) structure/piping models only. If a card input is used, a mass normalized mode shape is required. ERSIN03P is not applicable for axisymmetric analyses using a Fourier Decomposition technique.

3.14.7 RINEX Computer Program

Description: RINEX is a computer code used to interpolate and extrapolate amplified response spectra used in the response spectrum method of dynamic analysis. RINEX is also used to generate response spectra with non-constant model damping. The non-constant model damping

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analysis option can calculate spectral acceleration at the discrete eigenvalues of a dynamic system using either the strain energy weighted modal damping or the ASME BPVC-III Code Case N-411-1, "Alternative Damping Values for Response Spectra Analysis of Class 1, 2, and 3 Piping, Section III, Division 1, ERRATA SUP 13" (Reference 3.12-20) damping values.

Validation: Hand calculations and test cases analyzed are used to demonstrate the program's applicability and validity.

Extent of Application: This program is used to generate multiple damping spectra for piping.

3.14.8 PDA (Civil)

Description: Pipe Dynamic Analysis (PDA) Pipe Whip Restraint Analysis

Validation: This software is not approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software and requires output verification in accordance with the CP-03-100 Design Process.

Extent of Application: GEH in-house program for calculating pipe whip response under postulated break conditions. Determines response for a standard configuration which utilizes U-type pipe whip restraint.

3.14.9 PIPESTRESS

Description: PIPESTRESS (developed under a Quality Assurance Program compliant with the ASME NQA-1 (Reference 3.12-18) standard along with 10 CFR 21, "Reporting of Defects and Noncompliance" (Reference 3.12-9) and 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" (Reference 3.12-10)) is a pipe stress and flexibility analysis program, used for the evaluation of structural response and stress levels of piping systems against the requirements of industry codes and standards.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The plant layout, isometric drawings, P&ID, PFD, etc. will be used to build the piping model in PIPESTRESS, then PIPESTRESS will calculate the displacement, force/moment and stress. This software has the piping information, pipe routing & system information for BWRX-300 & some equipment information.

3.14.10 FLOMASTER v2021.1

Description: Uses simulation to offer reliable & accurate solvers and solutions for fluids engineering

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: Simcenter Flomaster is a unique thermo-fluid system simulation software tool used to simulate thermo-fluid systems; facilitating upfront engineering to reduce cost and lead times in product development and maintenance. It has an extensive library of component models, pre-populated with reliable performance data, Flomaster allows fluid system design to start before CAD data is available and component suppliers have been selected.

3.14.11 Ansys LS-Dyna v2021

Description: Ansys LS-DYNA is an explicit simulation program capable of simulating the response of materials to short periods of severe loading. Its many elements, contact formulations, material models and other controls can be used to simulate complex models with control over all

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the details of the problem. Ansys LS-DYNA applications include explosion/penetration, impact analysis, and non-linear explicit structural analysis.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: Ansys LS-DYNA will be used to analyze BWRX-300 structures for effects of blast loading and aircraft impact.

3.14.12 3KeyMaster v2021 (ICE/Plant Integration Engineering/Systems Engineering)

Description: Plant-wide physics-based simulation supporting engineering design options, confirmation, and future reactor operator training full scope simulator (FSS) in accordance with ANS Std 3.5.

Validation: This software is not approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software and requires output verification in accordance with the CP-03-100 Design Process.

Extent of Application: 3KeyMaster is used to generate plant layout schematics & run test simulations for new plant setups through variable/parameter manipulation for OPG.

APPENDIX 3D – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF ELECTRICAL STRUCTURES, SYSTEMS AND COMPONENTS

3.15 Introduction

This appendix describes the major computer programs used in the analysis of electrical SSC. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd's Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH's capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

3.15.1 ETAP v2021.1 (ICE Systems/I&C Tech)

Description: Electrical Transient Analyzer Program (ETAP) is an electrical network modeling and simulation software tool used by power systems engineers to create an "electrical digital twin" and analyze electrical power system dynamics, transients and protection.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: ETAP is the Global Market and Technology Leader of power systems solutions for a broad spectrum of sectors including Generation, Transmission, Distribution, Transportation, Industrial, and Commercial. The most comprehensive and integrated model-driven solutions for design, simulation, analysis, optimization, monitoring, operation, and automation of electrical power systems.

3.15.2 LDRA (I&C Tech/ICE Systems)

Description: Liverpool Data Research Associates is a provider of software analysis, and test and requirements traceability tools for the Public and Private sectors and a pioneer in static and dynamic software analysis.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: LDRA is a tool used to perform unit/module testing on software functions and components. It allows us to create and store test cases so we can perform regression testing, and it also allows us to execute the test cases on the target hardware (in this case an ARM Cortex-A9 processor).

3.15.3 Quartus II (I&C Tech/ICE Systems)

Description: Tools that provide FPGA compiler, simulation, and programming capabilities.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

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Extent of Application: Quartus is a tool used to develop applications for programmable logic devices such as PLDs and FPGAs. Applications in this case means the logic that the device implements. For example, it could be logic that provides a 2 out of 3 votes, it could be something that processes digital communications such as our fibre optic links, etc. Included in the software is something called timing analysis, which is a methodology for ensuring the logic inside the device meets timing characteristics. It also includes support for a simulator. The simulator allows engineers to evaluate the functionality of their logic by specifying input and examining how the logic reacts (e.g., verify the correctness of the design). The simulator does not require a physical device.

APPENDIX 3E – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSES STRUCTURES, SYSTEMS AND COMPONENTS – NUCLEAR FUELS

3.16 Introduction

This appendix describes the major computer programs used in the analysis of nuclear fuels. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description” (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd’s Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH’s capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

3.16.1 EPRI: Acube v11

Description: Advanced cutset upper bound estimator

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use ACUBE to post-process result cutsets using a Binary Decision Diagram method which will provide a more accurate point estimate of the results. ACUBE is a post-processing software that analyzes minimal cutsets and returns an estimate of the probability for a given top event using the BDD method. The BDD method is more accurate estimation than the approximation calculations used in baseline results. The software can be used with manual inputs but typically is used with intermediate quantification software such as FRANX or PRAQuant.

3.16.2 EPRI: CAFTA v11

Description: CAFTA is an integrated tool to perform Probabilistic Risk Analysis, incorporating linking event tree/fault tree methodology.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The CAFTA software will be qualified to complete all designed functions within the software. The use of the CAFTA software will be acceptable for use as is. Note that the testing will not cover every possible variation or combination of use for the software but it will validate the software operates as intended for within the standard operating configuration of the software.

3.16.3 EPRI: MAAP v5

Description: The Modular Accident Analysis Program (MAAP) Version 5 - an Electric Power Research Institute (EPRI) owned and licenced computer software - is a fast-running computer code that simulates the response of light water and heavy water moderated nuclear power plants for both current and Advanced Light Water Reactor (ALWR) designs. It can simulate Loss-Of-Coolant Accident (LOCA) and non-LOCA transients for Probabilistic Risk Assessment (PRA) applications as well as severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines (SAMGs).

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use MAAP to analyze reactor thermal-hydraulic and containment response to transients as well as severe accident sequence progressions. MAAP is used to predict the timing of key events, evaluate the influence of mitigative systems, evaluate effectiveness of operator actions, predict magnitude and timing of fission product releases, and investigate uncertainties in severe accident phenomena.

3.16.4 EPRI: PRAQuant v11

Description: Accident Sequence Quantification. In performing a fault tree based analysis it is often necessary to solve the fault tree several times, using different subtrees, boundary conditions, truncations or other assumptions about the model. These solutions can be performed manually in the CAFTA software, but it is often difficult to track and document the numerous results. PRAQuant is a general tool to configure several fault tree analysis solutions in advance, and to track the completion and results from each run.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use PRAQuant in the processing of the combined hazard model to generate a combined hazard cutset output. PRAQuant is a processing software to configure several fault tree analysis solutions and track the completion and results from each run. The software is capable of defining specific criteria to be applied in each fault tree analysis solution (e.g., flag files, recovery rules, output file name, truncation, etc.) and processes the supplied inputs into a format that a quantification engine (e.g., FTREX) is capable of processing. Once the quantification engine generates an output cutset file, the software can interface with QRecover to apply recovery rules before saving the final output to a defined directory.

3.16.5 FURST (Core & Fuel)

Description: Static & dynamic modeling

Validation: The software is approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software.

Extent of Application: Mechanical design of core internals loads, deflections, and stress analysis for X300

3.16.6 GTRAC v1

Description: Post-processing TRACG graphics file to edit desired output

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: GTRAC01P is a computer program that accepts binary graphics files generated by compatible versions of TRACG04P as input, and outputs user requested portions of those results into ASCII and CEDAR formats suitable for further post-processing. The data quantities residing on a TRACG graphics file are referred to as labels. An input file is used to request desired data using the corresponding label names in accordance with the structure defined in the TRACG User's Manual. If the labels on the graphics file are unknown, GTRAC01P can provide a listing of labels present on the file without actually outputting any label data, or users can use wildcard and pattern matching to request any labels that match a provided pattern. Some additional data is available on the graphics file, including a short description of the data set, and the units associated with data.

3.16.7 MACCS v4

Description: The MELCOR Accident Consequence Code Systems (MACCS) code, and its successor code, MACCS2, are based on the straight-line Gaussian plume model was developed originally for the Nuclear Regulatory Commission (USNRC). MACCS2 evaluates doses and health risks from the accidental atmospheric releases of radio nuclides. The principal phenomena considered in MACCS2 are atmospheric transport and deposition under time-variant meteorology, short-term and long-term mitigative actions and exposure pathways, deterministic and stochastic health effects, and economic costs.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: MACCS will be used as part of the licensing basis events analysis in radiological consequences.

3.16.8 MCNPX v6

Description: Monte Carlo N-Particle Transport is a general-purpose, continuous-energy, generalized-geometry, time-dependent, Monte Carlo radiation transport code designed to track many particle types over broad ranges of energies and is developed by Los Alamos National Laboratory.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: MCNP will be used for performing criticality and shielding analyses. MCNP can be used in several transport modes: neutron only, photon only, electron only, combined neutron/photon transport where the photons are produced by neutron interactions, neutron/photon/electron, photon/electron, or electron/photon. The neutron energy regime is from 10⁻¹¹ MeV to 20 MeV for all isotopes and up to 150 MeV for some isotopes, the photon energy regime is from 1 keV to 100 GeV, and the electron energy regime is from 1 KeV to 1 GeV. The capability to calculate keff eigenvalues for fissile systems is also a standard feature.

3.16.9 ORIGEN v1

Description: ORIGEN is a one-group depletion and radioactive decay computer code. ORIGEN is used to calculate the radionuclide composition and other related properties of nuclear materials (irradiated fuel isotope inventory).

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: ORIGEN is used for calculating core inventories of isotopes, and sometime for performing activation analyses of various materials or components.

3.16.10 PANAC v11

Description: PANAC (PANACEA) is the computer program used for the detailed nuclear calculations of the BWR Core. It is a steady-state, three-dimensional, one and one half energy group, diffusion theory computer program with coupled nuclear and thermal-hydraulic representation of the reactor Core.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: The BWR Core Simulator (PANAC11A/P) is a steady-state, three-dimensional coupled nuclear-thermalhydraulic computer program representing a BWR core. An automated plant heat balance option is used for modeling of the external flow loop. Provisions are made for fuel cycle and thermal limits calculations. The program is used for detailed three-dimensional design and operational calculations of BWR neutron flux and power distributions and thermal performance as a function of control rod position, refueling pattern, coolant flow, reactor pressure, and other operational and design variables. A special power exposure iteration option is available for target exposure distribution and cycle length predictions. PANAC11A/P includes the effect of Doppler broadening as a function of moderator density, exposure, control and moderator density history for a given fuel type. The nuclear model is based on coarse-mesh nodal, improved 1-1/2 group (quasi-two group), static diffusion theory. The diffusion equations are solved using the fast energy group. Resonance energy neutronic effects are included in the model by relating the resonance fluxes to the fast energy flux. The thermal flux is represented by an asymptotic expansion using a slowing down source from the epithermal region. A spectral history reactivity model and control blade history reactivity model are included. Control blade history local peaking effects are also incorporated in the nuclear model. A pin power reconstruction model is implemented to account for the effect of flux gradients across the nodes on the local peaking distribution. Neutronic parameters used by PANAC11A/P are obtained from the two-dimensional lattice physics code (TGBLA06) and parametrically fitted as a function of moderator density, exposure, control and moderator density history for a given fuel type.

3.16.11 PRIME v3

Description: The PRIME03P computer program is used to calculate the thermal/mechanical response of nuclear fuel to time varying power histories.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: PRIME03P is used for steady-state and transient licensing analysis of UO₂ and (U,Gd)O₂ fuel with (and without) additive material. PRIME03P is used for steady-state and transient licensing analysis as well as qualification cases of Recrystallized Annealed Zircaloy-2 cladding. Additionally, PRIME03P may be used with Stress-Relieved Annealed Zircaloy-4 cladding of either 70 % or 30 % cold work for qualification cases, but not for licensing analysis.

3.16.12 RAMP: GALE v3.2

Description: The Gaseous and Liquid Effluents (GALE) series of codes consists of four codes that calculate the gaseous and liquid effluent releases from pressurized-water reactors (PWRs) and boiling-water reactors (BWRs)

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: GALE uses a combination of input data and hardwired parameters to calculate the source term of radionuclides generated by a nuclear power plant during routine operation. Parameters that vary from plant to plant are treated as "inputs"; GALE asks the

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operator for input values on each run. Hardwired parameters are plant characteristics that are assumed to be the same for all reactors.

3.16.13 RAMP: HABIT v2.2

Description: HABIT v2.2 is a suite of computer codes to assist in evaluating Light Water Reactor (LWR) control room habitability in the event of accidental spills of toxic chemicals or the accidental release of radionuclides, including noble gas.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: HABIT v2.2 also uses a heavy-gas dispersion model, unifies the input screen of EXTRAN, DEGADIS, and SLAB, and incorporates Bitter Mc-Quaid calculation to determine which model needs to run and plot the concentration versus time outputs.

3.16.14 RAMP: DandD v2.1

Description: A code for screening analyses for licence termination and decommissioning.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The DandD software automates the definition and development of the scenarios, exposure pathways, models, mathematical formulations, assumptions, and justifications of parameter selections documented in Volumes 1 and 3 of USNRC NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning" (Reference 3.12-15).

3.16.15 RAMP: GENII v2.10

Description: GENII Version 2.10 is now part of the Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) at the U.S. Nuclear Regulatory Commission.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: GENII is a documented set of programs for calculating radiation dose and risk from radionuclides released to the environment. Although the code was initially developed for the U.S. Environmental Protection Agency, regulators and decision makers in other federal agencies, including several outside the U.S., employ this state-of-the-art, technically peer reviewed system to analyze hazards and design controls to prevent or mitigate potential accidents.

3.16.16 RAMP: MILDOS v4

Description: Radiological dose commitment calculation code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The MILDOS-AREA computer code calculates the radiological dose commitments received by individuals and the general population within an 80-km radius of an operating uranium recovery facility. In addition, air and ground concentrations of radionuclides are estimated for individual locations, as well as for a generalized population grid. Extra-regional population doses resulting from transport of radon and export of agricultural produce are also estimated.

3.16.17 RAMP: NRC-RADTRAN v6.02.1

Description: Risk & Consequence analysis code

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The USNRC Radioactive Material Transport (NRC-RADTRAN) computer code is used for risk and consequence analysis of radioactive material transportation. A variety of radioactive material is transported annually within this country and internationally. The shipments are carried out by overland modes (mainly truck and rail), marine vessels, and aircraft. Transportation workers and persons residing near or sharing transportation links with these shipments may be exposed to radiation from radioactive material packages during routine transport operations; exposures may also occur as a result of accidents. Risks and consequences associated with such exposures are the focus of the NRC-RADTRAN code.

3.16.18 RAMP: PIMAL v4.1.0

Description: GUI with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The PIMAL code is a graphical user interface with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes. It allows users to easily generate quantitative figures of merit regarding positioning arms and legs in difference geometries. PIMAL software is considered an efficient and accurate tool for performing dosimetry calculations for radiation workers and exposed members of the public.

3.16.19 RAMP: TurboFRMAC v2021 11.0.2

Description: Radiological Hazard evaluation code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The Turbo FRMAC analysis tool performs complex calculations to quickly evaluate radiological hazards during an emergency response by assessing impacts to the public, workers, and the food supply. Turbo FRMAC can be used to evaluate the hazard from a wide variety of radiological incidents, such as:

- Radiological Dispersal Devices (RDDs)
- Nuclear Power Plant Emergencies
- Fuel Handling Accidents
- Transportation Accidents
- Nuclear Detonations

Turbo FRMAC calculations are based on methods established by the Federal Radiological Monitoring and Assessment Center (FRMAC).

3.16.20 RAMP: VARSKIN v1.0

Description: Occupational Dose Analysis Code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: VARSKIN+ is used to calculate occupational dose to the skin resulting from exposure to radiation emitted from hot particles or other contamination on or near the skin. These assessments are required by 10 CFR 20.1201(c), "Occupational dose limits for Adults" {Reference 3.12-11), which states that the assigned shallow dose equivalent is to the part of the body receiving the highest exposure over a contiguous 10 cm² of skin at a tissue depth of 0.007 centimeters (7 mg/cm²).

3.16.21 SAP4G07P v7

Description: SAP4G07P has been tested for a range of applications for static and dynamic analyses of structural and piping systems. SAP4G07P is generated in FORTRAN.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: SAP4G07P has been tested for a range of applications for static and dynamic analyses of structural and piping systems. SAP4G07P is generated in FORTRAN and has been compiled and run on Windows 7 (32 bit), Windows 7 (64 bit), and Windows 2003 and 2012 servers.

3.16.22 SCALE v6

Description: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design. Scale6.1 (KENO/ORIGEN-ARP/S).

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: SCALE (KENOVI) is a Monte Carlo program for solving the neutron transport equation for an eigenvalue problem. The code implements the Monte Carlo process for neutron, photon, electron, or coupled transport involving all these particles, and computes the eigenvalue for neutron-multiplying systems. KENOVI uses the pointwise (i.e., continuous) cross-section data, and all reactions in a given cross-section evaluation (e.g., ENDF/B-VII.0) are considered.

3.16.23 TGBLA v6

Description: LANCR will replace TGBLA. Calculates lattice parameters for fuel bundles and the output is used by PANACEA to model the behavior of the fuel in the core

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: TGBLA06 is a lattice design computer program for conventional BWRs, which have the following lattices: 7x7, 8x8, 9x9, or 10x10. Water rods, including large central water rods and approximations for centered and offset water boxes, may be introduced into cells of the 2D mesh, which TGBLA06 solves. The 8x8 lattice can have up to four cells per water rod; the 9x9 lattice can have up to 3.5 cells per water rod; the 10x10 lattice can have up to four cells per water rod. Lattices with vanishing rods, thick-thin channels, or some water cross designs such as 8x8 and 10x10 water cross lattices, are qualified. TGBLA06 is qualified for water box designs where the water box is simulated by the use of nine water rods. Although TGBLA06 is capable of analyzing 11x11 and 12x12 lattices, MOX fuel and other design configurations, it has not been qualified for them. TGBLA06 solves 2D diffusion equations with diffusion parameters corrected by transport theory to provide the multiplication factor, the fission density distribution, the neutron balance, and the homogenized cross sections. Also, TGBLA06 performs burnup calculations for generating input to the BWR 3D simulator. In addition, TGBLA06 generates the rod-by-rod neutron cross sections, gamma smeared power distributions and flux discontinuity factors. The ring-by-ring gamma source distribution in gadolinium rods is not correct and should not be used.

3.16.24 TRACG v4

Description: TRACG is a GEH version of the Transient Reactor Analysis Code representing a best-estimate code for the analysis of BWR transients. It is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: TRACG04 is a computer program applicable for the calculation of thermal-hydraulic parameters and reactor power during BWR transients. TRACG04 is intended to be used as a 'best-estimate' system computer code, with capabilities for three-dimensional hydrodynamic calculations in the vessel components, and one-dimensional calculations in the other components. A full two-fluid representation supplemented by air and boron models is employed for the characterization of two-phase flow, allowing application to transients where thermal non-equilibrium and counter-current flow between phases is significant. TRACG04 has point, 1-D, and 3-D neutron kinetics models for simulating the feedback effects of moderator density, fuel temperature, boron, and control blade movement on the core power. TRACG04 has a control system model capable of simulating the BWR feedback control system. TRACG04 is capable of modeling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modeling the BWR, TRACG04 is also applicable to experimental test facilities constructed from components representative of a BWR.

3.16.25 SEISM v5

Description: The SEISM program can be used for the non-linear response prediction of structural system with spring, damper, friction & stop element, under dynamic loads. The program employs the component element method and can account for impact and friction forces effect. SEISM program performs calculations in double precision.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: SEISM can be used for the non-linear time history response prediction of structural systems with spring, damper, friction and stop elements under dynamic loads. The program employs the component element method and can account for impact and friction force effects. When running SEISM, the user can select to run any of its four modules (CRTFI, SEPRE, SEISM, SEPST) individually or combined within a single session. Output of one module may be passed to and used as input to the next module.

3.16.26 DECAY v1

Description: DECAY01A calculates the decay heat power fraction after certain operation period and exposure of a fissile core.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: DECAY01A is an Engineering Computer Code developed by GE Hitachi Nuclear Energy (GEH) as a method to determine the decay heat (shutdown power) for BWR fuel. The code was created in response to USNRC IN96-39, "Estimates of Decay Heat Using ANS 5.1 Decay Heat Standard May Vary Significantly" (Reference 3.12-16) that brought attention to the extreme variation in decay heat calculations throughout the country. This was due to either overly conservative assumptions or a misapplication of the ANS Decay Heat Standards. The DECAY01A code has therefore gone to great lengths to assure both the validity and applicability of its calculations. DECAY01A works as a function of both the ANSI/ANS-5.1-1979, "American National Standard Decay Heat Power in Light Water Reactors" (Reference 3.12-17) or ANSI/ANS-5.1-1994 (Reference 3.12-17) decay heat standards used for domestic and advanced reactor

designs respectively. These standards set forth values of decay heat from fission products of ²³⁵U, ²³⁹Pu, ²³⁸U and ²⁴¹Pu; and decay heat from actinides ²³⁹U and ²³⁹Np. DECAY01A also includes the decay heat contribution from other Actinides (in addition to ²³⁹U and ²³⁹Np which are specified in the Standard) as well as from Activation Products. In addition to the decay heat, DECAY01A evaluates the one-sigma uncertainty in the decay heat and adds a user-specified multiple of this uncertainty (usually 2 sigma) to the decay heat power.

3.16.27 GTRAC v1

Description: Post-processing TRACG graphics file to edit desired output

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: GTRAC01P is a computer program that accepts binary graphics files generated by compatible versions of TRACG04P as input, and outputs user requested portions of those results into ASCII and CEDAR formats suitable for further post-processing. The data quantities residing on a TRACG graphics file are referred to as labels. An input file is used to request desired data using the corresponding label names in accordance with the structure defined in the TRACG User's Manual. If the labels on the graphics file are unknown, GTRAC01P can provide a listing of labels present on the file without actually outputting any label data, or users can use wildcard and pattern matching to request any labels that match a provided pattern. Some additional data is available on the graphics file, including a short description of the data set, and the units associated with data.

APPENDIX 3F – COMPUTER PROGRAMS USED IN ENVIRONMENTAL AND RADIOLOGICAL ANALYSES SUPPORTING THE DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3.17 Introduction

This appendix describes the major computer programs used in deterministic and probabilistic safety analyses. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description” (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd’s Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH’s capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

3.17.1 ADDAM Version 1.4.2

Description: The ADDAM (Atmospheric Dispersion and Dose Analysis Method) computer code computes the statistical distribution of radiation doses to an individual or population after the airborne release of radioactive material into the environment. See Chapter 15, Subsection 15.5.1.2.5 for a description.

Validation

Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application

See Chapter 15, Subsection 15.5. for extent of application.

3.17.2 DECAY v1

Description: DECAY01A calculates the decay heat power fraction after certain operation period and exposure of a fissile core.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: DECAY01A is an Engineering Computer Code developed by GE Hitachi Nuclear Energy (GEH) as a method to determine the decay heat (shutdown power) for BWR fuel. The code was created in response to USNRC IN96-39 (Reference 3.12-16) that brought attention to the extreme variation in decay heat calculations throughout the country. This was due to either overly conservative assumptions or a misapplication of the ANS Decay Heat Standards. The DECAY01A code has therefore gone to great lengths to assure both the validity and applicability of its calculations. DECAY01A works as a function of both the ANSI/ANS-5.1-1979 (Reference

3.12-17) or ANSI/ANS-5.1-1994 (Reference 3.12-17) decay heat standards used for domestic and advanced reactor designs respectively. These standards set forth values of decay heat from fission products of ²³⁵U, ²³⁹Pu, ²³⁸U and ²⁴¹Pu; and decay heat from actinides ²³⁹U and ²³⁹Np. DECAY01A also includes the decay heat contribution from other Actinides (in addition to ²³⁹U and ²³⁹Np which are specified in the Standard) as well as from Activation Products. In addition to the decay heat, DECAY01A evaluates the one-sigma uncertainty in the decay heat and adds a user-specified multiple of this uncertainty (usually 2 sigma) to the decay heat power.

3.17.3 RADTRAD (Analytical Methods/ Radiological Analysis)

Description: RADTRAD uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario. It also provides the inventory, decay chain, and dose conversion factor tables needed for the dose calculation.

Validation: The software is approved for production use under GEH procedure CP-23-400, Engineering Software for Design and Analysis Software.

Extent of Application: The RADTRAD code is used for calculating accident doses, calculating transport of fission products inside the plant after an accident, performing filter loading calculations for post-accident.

3.17.4 RAMP: GALE v3.2

Description: The Gaseous and Liquid Effluents (GALE) series of codes consists of four codes that calculate the gaseous and liquid effluent releases from pressurized-water reactors (PWRs) and boiling-water reactors (BWRs)

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: GALE uses a combination of input data and hardwired parameters to calculate the source term of radionuclides generated by a nuclear power plant during routine operation. Parameters that vary from plant to plant are treated as "inputs"; GALE asks the operator for input values on each run. Hardwired parameters are plant characteristics that are assumed to be the same for all reactors.

3.17.5 RAMP: HABIT v2.2

Description: HABIT v2.2 is a suite of computer codes to assist in evaluating Light Water Reactor (LWR) control room habitability in the event of accidental spills of toxic chemicals or the accidental release of radionuclides, including noble gas.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: HABIT v2.2 also uses a heavy-gas dispersion model, unifies the input screen of EXTRAN, DEGADIS, and SLAB, and incorporates Bitter Mc-Quaid calculation to determine which model needs to run and plot the concentration versus time outputs.

3.17.6 RAMP: DandD v2.1

Description: A code for screening analyses for licence termination and decommissioning.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The DandD software automates the definition and development of the scenarios, exposure pathways, models, mathematical formulations, assumptions, and

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justifications of parameter selections documented in Volumes 1 and 3 of NUREG/CR-5512 (Reference 3.12-15).

3.17.7 RAMP: GENII v2.10 (Analytical Methods/Radiological Analysis)

Description: GENII Version 2.10 is now part of the Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) at the U.S. Nuclear Regulatory Commission.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: GENII is a documented set of programs for calculating radiation dose and risk from radionuclides released to the environment. Although the code was initially developed for the U.S. Environmental Protection Agency, regulators and decision makers in other federal agencies, including several outside the U.S., employ this state-of-the-art, technically peer reviewed system to analyze hazards and design controls to prevent or mitigate potential accidents.

3.17.8 RAMP: MILDOS v4

Description: Radiological dose commitment calculation code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The MILDOS-AREA computer code calculates the radiological dose commitments received by individuals and the general population within an 80-km radius of an operating uranium recovery facility. In addition, air and ground concentrations of radionuclides are estimated for individual locations, as well as for a generalized population grid. Extra-regional population doses resulting from transport of radon and export of agricultural produce are also estimated.

3.17.9 RAMP: NRC-RADTRAN v6.02.1

Description: Risk & Consequence analysis code

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The NRC Radioactive Material Transport (NRC-RADTRAN) computer code is used for risk and consequence analysis of radioactive material transportation. A variety of radioactive material is transported annually within this country and internationally. The shipments are carried out by overland modes (mainly truck and rail), marine vessels, and aircraft. Transportation workers and persons residing near or sharing transportation links with these shipments may be exposed to radiation from radioactive material packages during routine transport operations; exposures may also occur as a result of accidents. Risks and consequences associated with such exposures are the focus of the NRC-RADTRAN code.

3.17.10 RAMP: PIMAL v4.1.0

Description: GUI with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The PIMAL code is a graphical user interface with pre-processor and post-processor capabilities which assists users in developing MCNP input decks and running the codes. It allows users to easily generate quantitative figures of merit regarding positioning arms

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and legs in difference geometries. PIMAL software is considered an efficient and accurate tool for performing dosimetry calculations for radiation workers and exposed members of the public.

3.17.11 RAMP: TurboFRMAC v2021 11.0.2

Description: Radiological Hazard evaluation code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: The Turbo FRMAC analysis tool performs complex calculations to quickly evaluate radiological hazards during an emergency response by assessing impacts to the public, workers, and the food supply. Turbo FRMAC can be used to evaluate the hazard from a wide variety of radiological incidents, such as:

- Radiological Dispersal Devices (RDDs)
- Nuclear Power Plant Emergencies
- Fuel Handling Accidents
- Transportation Accidents
- Nuclear Detonations

Turbo FRMAC calculations are based on methods established by the Federal Radiological Monitoring and Assessment Center (FRMAC).

3.17.12 RAMP: VARSKIN v1.0

Description: Occupational Dose Analysis Code

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: VARSKIN+ is used to calculate occupational dose to the skin resulting from exposure to radiation emitted from hot particles or other contamination on or near the skin. These assessments are required by 10 CFR 20.1201(c) {Reference 3.12-11), which states that the assigned shallow dose equivalent is to the part of the body receiving the highest exposure over a contiguous 10 cm² of skin at a tissue depth of 0.007 centimeters (7 mg/cm²).

3.17.13 SAP4G07P v7

Description: SAP4G07P has been tested for a range of applications for static and dynamic analyses of structural and piping systems. SAP4G07P is generated in FORTRAN.

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: SAP4G07P has been tested for a range of applications for static and dynamic analyses of structural and piping systems. SAP4G07P is generated in FORTRAN and has been compiled and run on Windows 7 (32 bit), Windows 7 (64 bit), and Windows 2003 and 2012 servers.

3.17.14 SCALE v6

Description: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design. Scale6.1 (KENO/ORIGEN-ARP/S).

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: SCALE (KENOVI) is a Monte Carlo program for solving the neutron transport equation for an eigenvalue problem. The code implements the Monte Carlo process for neutron, photon, electron, or coupled transport involving all these particles, and computes the eigenvalue for neutron-multiplying systems. KENOVI uses the pointwise (i.e., continuous) cross-section data, and all reactions in a given cross-section evaluation (e.g., ENDF/B-VII.0) are considered.

3.17.15 TGBLA v6

Description: LANCR will replace TGBLA. Calculates lattice parameters for fuel bundles and the output is used by PANACEA to model the behavior of the fuel in the core

Validation: Validation of this tool is in compliance with the OPG project quality plan.

Extent of Application: TGBLA06 is a lattice design computer program for conventional BWRs, which have the following lattices: 7x7, 8x8, 9x9, or 10x10. Water rods, including large central water rods and approximations for centered and offset water boxes, may be introduced into cells of the 2D mesh, which TGBLA06 solves. The 8x8 lattice can have up to four cells per water rod; the 9x9 lattice can have up to 3.5 cells per water rod; the 10x10 lattice can have up to four cells per water rod. Lattices with vanishing rods, thick-thin channels, or some water cross designs such as 8x8 and 10x10 water cross lattices, are qualified. TGBLA06 is qualified for water box designs where the water box is simulated by the use of nine water rods. Although TGBLA06 is capable of analyzing 11x11 and 12x12 lattices, MOX fuel and other design configurations, it has not been qualified for them. TGBLA06 solves 2D diffusion equations with diffusion parameters corrected by transport theory to provide the multiplication factor, the fission density distribution, the neutron balance, and the homogenized cross sections. Also, TGBLA06 performs burnup calculations for generating input to the BWR 3D simulator. In addition, TGBLA06 generates the rod-by-rod neutron cross sections, gamma smeared power distributions and flux discontinuity factors. The ring-by-ring gamma source distribution in gadolinium rods is not correct and should not be used.

3.17.16 TRACG v4

Description: TRACG is a GEH version of the Transient Reactor Analysis Code representing a best-estimate code for the analysis of BWR transients. It is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: TRACG04 is a computer program applicable for the calculation of thermal-hydraulic parameters and reactor power during BWR transients. TRACG04 is intended to be used as a 'best-estimate' system computer code, with capabilities for three-dimensional hydrodynamic calculations in the vessel components, and one-dimensional calculations in the other components. A full two-fluid representation supplemented by air and boron models is employed for the characterization of two-phase flow, allowing application to transients where thermal non-equilibrium and counter-current flow between phases is significant. TRACG04 has point, 1-D, and 3-D neutron kinetics models for simulating the feedback effects of moderator density, fuel temperature, boron, and control blade movement on the core power. TRACG04 has a control system model capable of simulating the BWR feedback control system. TRACG04 is capable of modeling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modeling the BWR, TRACG04 is also applicable to experimental test facilities constructed from components representative of a BWR.

3.17.17 **IMPACT**

Description: IMPACT is a customizable tool that allows the user to assess the transport and fate of contaminants through a user-specified environment.

Validation: The software is not qualified under the engineering software process and the output of the software will be verified with each use per the design process.

Extent of Application: IMPACT performs the calculations for CSA N288.1:14, "Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities", R2019 (Reference 3.12-14). The code calculates the doses from routine effluent emission from a plant that are the results of normal operation.

APPENDIX 3G – COMPUTER PROGRAMS USED IN THE DESIGN OF COMPONENTS, SYSTEMS AND STRUCTURES IN SAFETY ANALYSES (PRA AND DETERMINISTIC)

3.18 Introduction

This appendix describes the major computer programs used in the analysis of the safety-related components, equipment, and structures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description” (Reference 3.12-4) that complies with ASME NQA-1 Quality program (Reference 3.12-18) and CSA N286.7-16 (Reference 3.12-12).

GEH maintains an ISO 9001:2015 (Reference 3.12-5) Certificate of Approval by U.S. Lloyd’s Registrar QA (Identify Number: 10068327), with the following scope of approval applicable to:

- Design, Engineering, Procurement, and Servicing of Nuclear Power Plants, Related Systems and Components
- Design and Manufacturer of Nuclear Fuel
- Design and Development of Associated Software

The GEH design control measures are presented in Appendix A to reflect GEH’s capabilities to meet the management system and high energy reactor facilities requirements described in CSA N286-12 (Reference 3.12-6).

3.18.1 EPRI: Acube v11

Description: Advanced cutset upper bound estimator

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use ACUBE to post-process result cutsets using a Binary Decision Diagram method which will provide a more accurate point estimate of the results. ACUBE is a post-processing software that analyzes minimal cutsets and returns an estimate of the probability for a given top event using the BDD method. The BDD method is more accurate estimation than the approximation calculations used in baseline results. The software can be used with manual inputs but typically is used with intermediate quantification software such as FRANX or PRAQuant.

3.18.2 EPRI: CAFTA v11

Description: CAFTA is an integrated tool to perform Probabilistic Risk Analysis, incorporating linking event tree/fault tree methodology.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The CAFTA software will be qualified to complete all designed functions within the software. The use of the CAFTA software will be acceptable for use as is. Note that the testing will not cover every possible variation or combination of use for the software but it will validate the software operates as intended for within the standard operating configuration of the software.

3.18.3 EPRI: FRANX v11

Description: Development of PRA Hazards models (Fire, Flood, High Winds, Seismic, etc.)

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use FRANX in the development of the Internal Fire, Internal Flood, Seismic, and High Winds hazard analyses. Specifically, FRANX will be used to build hazard specific scenarios and generate one-top models for later combination into an integrated hazard model. The FRANX software is a tool for analyzing external event risk. This tool is used to manage and develop the scenarios, calculate the probabilistic impact on core damage, and generate one-top solution models.

3.18.4 EPRI: FTREx v1.8

Description: FTREX reads a fault tree that consists of Boolean equations for system failure and generates cut sets that are minimal combinations of component failures that cause system failure.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: This software will have all functionality qualified and be valid for use with the necessary interfacing software (e.g., FRANX, CAFTA, PRAQuant) or independently of those software. The software must be accessible from the interfacing software locations as well as have permission to read and write files to a temp directory and a defined output file directory.

3.18.5 EPRI: HRA Calculator

Description: Supports development of PRA Human Reliability Analyses

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use the HRA Calculator to develop the human reliability analysis, calculate the human error probabilities, and develop a dependency analysis for the credited operator actions. The HRA Calculator provides a step by step process for developing the HRA applying one of the following methods: CBDTM, HCR/ORE, ASEP, SPAR-H, THERP.

3.18.6 EPRI: MAAP v5

Description: The Modular Accident Analysis Program (MAAP) Version 5 - an Electric Power Research Institute (EPRI) owned and licenced computer software - is a fast-running computer code that simulates the response of light water and heavy water moderated nuclear power plants for both current and Advanced Light Water Reactor (ALWR) designs. It can simulate Loss-Of-Coolant Accident (LOCA) and non-LOCA transients for Probabilistic Risk Assessment (PRA) applications as well as severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines (SAMGs).

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use MAAP to analyze reactor thermal-hydraulic and containment response to transients as well as severe accident sequence progressions. MAAP is used to predict the timing of key events, evaluate the influence of mitigative systems, evaluate effectiveness of operator actions, predict magnitude and timing of fission product releases, and investigate uncertainties in severe accident phenomena.

3.18.7 EPRI: PRAQuant v11

Description: Accident Sequence Quantification. In performing a fault tree based analysis it is often necessary to solve the fault tree several times, using different subtrees, boundary conditions, truncations or other assumptions about the model. These solutions can be performed manually in the CAFTA software, but it is often difficult to track and document the numerous results. PRAQuant is a general tool to configure several fault tree analysis solutions in advance, and to track the completion and results from each run.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use PRAQuant in the processing of the combined hazard model to generate a combined hazard cutset output. PRAQuant is a processing software to configure several fault tree analysis solutions and track the completion and results from each run. The software is capable of defining specific criteria to be applied in each fault tree analysis solution (e.g., flag files, recovery rules, output file name, truncation, etc.) and processes the supplied inputs into a format that a quantification engine (e.g., FTREX) is capable of processing. Once the quantification engine generates an output cutset file, the software can interface with QRecover to apply recovery rules before saving the final output to a defined directory.

3.18.8 ActivePoint HMI/CIMPLICITY 11

Description: Digital user interface design and display software by GE Power that runs using GE Digital CIMPLICITY HMI/SCADA automation platform.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The HFE team is using the software to design the BWRX-300 digital user interfaces. The scope of the interfaces is all display screens run by the DCIS, and any other platforms that can communicate directly with CIMPLICITY.

3.18.9 Control ST – ToolboxST Tool

Description: GE Power's ControlST* software suite provides the foundation for the Mark* Vle Control System in a wide range of applications, including control, safety integrity level, monitoring, and protection of assets. ToolboxST is one of the tools within ControlST, used for process configuration and diagnostics software for process, SIL, excitation and power conversion

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: For BWRX-300, the HFE team is using ToolboxST to provide early dynamic features and testing capability for the digital user interfaces designed using ActivePoint HMI/CIMPLICITY. The tool allows emulation of "live" screen features without the need for a plant simulation model driving the software. This allows early usability testing of digital user interfaces, as part of the HFE design testing and evaluation set of activities. The software is not used in production.

3.18.10 EPRI: SysImp v11

Description: Analysis of PRA Importance Measures. SysImp is a software tool used to calculate the importance of basic events, or collections of those events, in a risk model. SysImp is designed for risk models where components, equipment trains, and systems are represented by groups of basic events.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use SysImp to preform risk importance sensitivities, calculations, and grouping system importance. SysImp allows for deriving insights from risk importance rankings, estimating total plant risk given a specific change, and collective risk importance measures.

3.18.11 EPRI: UNCERT v11

Description: PRA Uncertainty Propagation analysis tool. Uncertainty Evaluation Tool (UNCERT). UNCERT can read the cut set or sequence data created from CAFTA and calculate the uncertainty of the cut set result.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project will use UNCERT to perform the parametric uncertainty calculations on the output cut sets. The UNCERT software will take a defined input (e.g., cut set file and associated CAFTA RR database) and perform the uncertainty analysis utilizing either a Monte Carlo or Latin Hypercube sampling method. The output will calculate the metrics for the cut set using that defined method.

3.18.12 GOTHIC v8

Description: GOTHIC is a procured software from Zachry Nuclear Engineering, Inc. for design, licensing, safety and operating analysis of nuclear power plant containments, confinement buildings and system components.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: GOTHIC is used to perform a sensitivity analysis for the passive containment cooling system while developing the design.

3.18.13 MACCS v4

Description: The MELCOR Accident Consequence Code Systems (MACCS) code, and its successor code, MACCS2, are based on the straight-line Gaussian plume model was developed originally for the Nuclear Regulatory Commission (NRC). MACCS2 evaluates doses and health risks from the accidental atmospheric releases of radio nuclides. The principal phenomena considered in MACCS2 are atmospheric transport and deposition under time-variant meteorology, short-term and long-term mitigative actions and exposure pathways, deterministic and stochastic health effects, and economic costs.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: MACCS will be used as part of the licensing basis events analysis in radiological consequences.

3.18.14 RAMP: NRC-RADTRAN v6.02.1

Description: Risk & Consequence analysis code

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The NRC Radioactive Material Transport (NRC-RADTRAN) computer code is used for risk and consequence analysis of radioactive material transportation. A variety of radioactive material is transported annually within this country and internationally. The shipments are carried out by overland modes (mainly truck and rail), marine vessels, and aircraft. Transportation workers and persons residing near or sharing transportation links with these shipments may be exposed to radiation from radioactive material packages during routine transport operations; exposures may also occur as a result of accidents. Risks and consequences associated with such exposures are the focus of the NRC-RADTRAN code.

3.18.15 SCALE v6

Description: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design. Scale6.1 (KENO/ORIGEN-ARP/S).

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: SCALE (KENOVI) is a Monte Carlo program for solving the neutron transport equation for an eigenvalue problem. The code implements the Monte Carlo process for neutron, photon, electron, or coupled transport involving all these particles, and computes the eigenvalue for neutron-multiplying systems. KENOVI uses the pointwise (i.e., continuous) cross-section data, and all reactions in a given cross-section evaluation (e.g., ENDF/B-VII.0) are considered.

3.18.16 TRACG v4

Description: TRACG is a GEH version of the Transient Reactor Analysis Code representing a best-estimate code for the analysis of BWR transients. It is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: TRACG04 is a computer program applicable for the calculation of thermal-hydraulic parameters and reactor power during BWR transients. TRACG04 is intended to be used as a 'best-estimate' system computer code, with capabilities for three-dimensional hydrodynamic calculations in the vessel components, and one-dimensional calculations in the other components. A full two-fluid representation supplemented by air and boron models is employed for the characterization of two-phase flow, allowing application to transients where thermal non-equilibrium and counter-current flow between phases is significant. TRACG04 has point, 1-D, and 3-D neutron kinetics models for simulating the feedback effects of moderator density, fuel temperature, boron, and control blade movement on the core power. TRACG04 has a control system model capable of simulating the BWR feedback control system. TRACG04 is capable of modeling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modeling the BWR, TRACG04 is also applicable to experimental test facilities constructed from components representative of a BWR.

3.18.17 VTR.LMP

Description: Package of functions and data frames supporting VTR LMP applications. This package was developed using open-source code R. Currently only functions on a Mac platform.

Validation: The software qualification process is being followed and verification and validation is in progress.

Extent of Application: The BWRX-300 project currently does not use this code package; however, developmental work is in progress to explore the application of this software to BWRX-

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300. The VTR.LMP R code package contains the processing commands necessary for gathering the inputs and running them through the LMP code package functions. The final licensing basis events are processed in this code package for use with the Frequency-Consequence plot.

Note: There is a developmental X300.LMP that would be the starting point for future applications of this code package.



HITACHI

GE Hitachi Nuclear Energy

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September 30, 2022

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**Ontario Power Generation Inc.
Darlington New Nuclear Project
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 4
Reactor**

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
0	All	Initial Release

ACRONYM LIST

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CNSC	Canadian Nuclear Safety Commission
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRD	Control Rod Drive
CST	Condensate Storage Tank
DBA	Design Basis Accident
DEC	Design Extension Conditions
DL	Defense Line
DPS	Diverse Protection System
DSA	Deterministic Safety Analysis
ERICP	Emergency Rod Insertion Control Panel
ESBWR	Economic Simplified Boiling Water Reactor
FMCRD	Fine Motion Control Rod Drive
GEH	GE Hitachi Nuclear Energy
GNF	Global Nuclear Fuel
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
ISI	In-Service Inspection
IST	In-Service Testing
KKM	Kernkraftwerk Mühleberg (BWR/4 in Switzerland)
LHGR	Linear Heat Generation Rate
LPRM	Local Power Range Monitor
MFLCPR	Maximum Fraction Limiting Critical Power Ratio
MCPR	Minimum Critical Power Ratio
MFLPD	Maximum Fraction Limiting Power Density

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Acronym	Explanation
MLHGR	Maximum Linear Heat Generation Rate
NBS	Nuclear Boiler System
OLMCPR	Operating Limit Minimum Critical Power Ratio
PA	Postulated Accident
PIE	Postulated Initiating Event
PRNM	Power Range Neutron Monitoring System
RC&IS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RPV	Reactor Pressure Vessel
SAFDL	Specified Acceptable Fuel Design Limits
SCRRI	Selected Control Rod Rapid Insertion
SDC	Shutdown Cooling System
TRACG	Transient Reactor Analysis Code General Electric
USNRC	U.S. Nuclear Regulatory Commission
Δ CPR/ICPR	Delta Critical Power Ratio Over Initial Critical Power Ratio

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4.0 REACTOR

Chapter 4 describes the components of the nuclear fuel, reactor, and reactor core, including the fuel rods, fuel assemblies, reactivity control system, nuclear design, and thermal-hydraulic design including requirements pertaining to safety.

The reactor assembly consists of the Reactor Pressure Vessel (RPV), pressure-containing appurtenances including Control Rod Drive (CRD) housings, in-core instrumentation housings, and reactor internal components (Figure 4.1-1). A summary of the important design and performance characteristics of the reactor and plant is given in Table 1.5-1.

A brief overview of the BWRX-300 reactor is provided in Section 4.1. The reactor materials, internal components, specifications, controls on welding, non-destructive examination, fabrication, and materials are described in Chapter 5, Section 5.2. The reactor internal components are described in Chapter 5, Subsection 5.4.2. The RPV design is described in Chapter 5, Section 5.4.

The fuel assembly and control rods are described in Section 4.2 including the design basis requirements, analytical methods, and evaluation results. Aspects pertaining to the nuclear design of the core, including the reference BWRX-300 design used for safety analysis, are described in Section 4.3. The thermal-hydraulic design basis requirements and important methodologies are described Section 4.4, and thermal-hydraulic stability is discussed in Section 4.8. The core monitoring function is described in Section 4.7. A description of the CRD system and associated requirements is provided in Section 4.6.

The design bases and functional requirements applied in the nuclear design of the fuel, core and reactivity control system comply with CNSC REGDOC-2.5.2 (Reference 4.1-1), Section 8.1.

4.1 Summary Description

Chapter 4 describes the components of the nuclear fuel, reactor, and reactor core, including the fuel rods, fuel assemblies, reactivity control system, nuclear design, and thermal-hydraulic design including requirements pertaining to safety.

The reactor assembly consists of the RPV, pressure-containing appurtenances including CRD housings, in-core instrumentation housings, and reactor internal components (Figure 4.1-1). A summary of the important design and performance characteristics of the reactor and plant is given in Table 1.5-1.

4.1.1 Reactor Pressure Vessel

The BWRX-300 RPV is a vertical, cylindrical pressure vessel fabricated with forged rings or rolled plate welded together with a removable top head by use of a head flange, seals, and bolting. The RPV also includes penetrations, nozzles, and reactor internals support.

The increased RPV height, relative to forced circulation Boiling Water Reactors (BWRs), is achieved by a “chimney” in the space that extends from the top of the core (top guide) to the entrance to the chimney head and steam separator assembly. The natural circulation flow resulting from the tall RPV results in adequate thermal margins during power operation and off-normal conditions as described in Section 4.3.

The RPV design and description are provided in Chapter 5, Section 5.4.

4.1.2 Reactor Internal Components

The major reactor internal components are described Chapter 5, Section 5.4, including:

- Core components (control rods and nuclear instrumentation)
- Core support structures (shroud, shroud support, top guide, core plate, control rod guide tubes, control rod drive housings, fuel support castings, and orificed/peripheral fuel supports)
- Chimney
- Chimney head and steam separator assembly
- Steam dryer assembly

Except for the Zircaloy in the reactor core, these reactor internals are stress corrosion-resistant stainless steels or other high alloy material. The fuel assemblies (including fuel rods and channels), control rods, chimney head and steam separator assembly, chimney assembly, steam dryers and in-core instrumentation assemblies are removable when the reactor vessel is opened for refueling or maintenance.

4.1.3 Reactor Core

The reactor core is comprised of 240 fuel assemblies arranged to form an upright cylinder. Additionally, movable control elements are inserted or withdrawn for reactivity control. The fuel assemblies are comprised of hermetically sealed fuel rods in a square array along with upper & lower tie plates, water rods, fuel rod spacers, fuel channel and connecting components. The fuel assemblies are supported by the reactor internals. The reactor internals also direct the flow of the coolant into the fuel assembly past the fuel rods. The coolant is light water and serves as both the working fluid and moderator. Boiling within the fuel assembly results in a two-phase mixture that exits the top of the fuel assembly that then enters the steam separators after traversing the axial extent of the chimney. The resulting dry steam at the exit of the steam dryers is then sent to the turbine to drive the turbine generator. The RPV and major components are depicted in Figure 4.1-1.

4.1.4 Fuel Assembly and Control Rod Assembly

A BWRX-300 GNF2 fuel assembly consists of 92 fuel rods and two large central water rods that occupy eight (8) fuel rod locations contained in a 10x10 array (i.e., 100 lattice locations). Fourteen fuel rod locations are occupied by part length fuel rods.

The fuel rod consists of uranium dioxide in the form of cylindrical pellets contained in Zircaloy tubing. The tubing is plugged, sealed, and welded at the ends to encapsulate the fuel. Fuel rods are pressurized internally with helium during fabrication to reduce clad creepdown and promote heat transfer.

Key attributes of the GNF2 fuel assembly are summarized in Table 4.1-1.

The control rods are cruciform shaped and reside in the intra-assembly gap. As such, the control rod geometric envelope is separated from that of the fuel assembly. A control rod is associated with four (4) fuel assemblies that, together, constitute the fundamental BWR cell.

The design of the fuel and control rods is described in Section 4.2.

The CRD mechanisms used to withdraw and insert the control rods are of the fine motion type providing fine reactivity control, fast scram and redundant insertion means. The CRD system is described in Section 4.6

4.1.5 Nuclear Instrumentation

The performance of the core is monitored by fixed neutron detectors within the core. The in-core nuclear instrumentation provides input to automatic reactor core control functions. The BWRX-300 nuclear instrumentation consists of Power Range Neutron Monitoring System (PRNM), Gamma Thermometers (GTs), and Wide Range Neutron Monitors.

The PRNM provides neutron monitoring power signals to the SC1 I&C protection systems. The PRNM also provides signals for post-accident monitoring purposes and, through isolated one-way optical data, links to the core monitoring three-dimensional power distribution program and to the control rod blocking systems.

GTs are in-core devices that convert local gamma flux to an electrical signal that supplies information required to calibrate the Local Power Range Monitors (LPRM) in the PRNM system. They also provide input to the SC2 I&C protection systems for various trip and monitoring functions.

The Wide Range Neutron Monitor system is a redundant pair of industrial computers with a real time operating system that monitors the fixed neutron detectors in the core. The detectors are distributed radially in the core at fixed heights. Each detector is sensitive to neutrons from below criticality to approximately 20% thermal power. The core monitoring function design is provided in Sections 4.7 and Chapter 7, Section 7.3.

4.1.6 Analysis Techniques

The analytical techniques employed in core design are comprised of the computer codes summarized in Table 4.1-2 and engineering design practices. The computer codes in Table 4.1-2 are further described in Chapter 3, Appendix 3G.

4.1.7 References

- 4.1-1 CNSC Regulatory Document REGDOG-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

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Table 4.1-1: GNF2 Fuel Bundle Key Attributes

	GNF2
Fuel Bundle	
Total number of fuel rods	92
Number of full-length rods	78
Number of partial length rods	14 total, Two Lengths
Number of part length rods (long)	8
Number of part length rods (short)	6
Lattice Array	Figure 4.1-2
Representative Assembly weight (kgU)	186
Fuel Rod	
Cladding material	Zr-2 with zirconium inner liner
Assembly active fuel length (mm)	3810
Fuel pellet Material	UO ₂
Fuel Rod Fill Gas	He
Water Rod	
Number of Water Rods	2
Water Rod Material	Zr-2
Spacer	
Spacer Structure	Grid-type with flow diverters
Spacer Material	Alloy X-750
Number of spacers	8

Table 4.1-2: Analytical Techniques in Core Design

Analysis	Technique	Computer Code
Fuel rod design	Numerical solutions of 1D steady state and transient heat transfer and finite element mechanical analysis	PRIME03P
Fuel performance characteristics (temperature, internal pressure, and clad stress, etc.)		
Nuclear design Cross-sections and group constants	Lattice physics	TGBLA06
X-Y and X-Y-Z power distribution, reactivity coefficients	Steady state coupled nuclear thermal hydraulics Quasi 2 group diffusion theory	PANAC11
Axial power distributions, control rod worth		
Fuel rod power		
Thermal-hydraulic design steady state	Multi-dimensional, two-fluid model thermal hydraulics 3-D reactor kinetics	TRACG

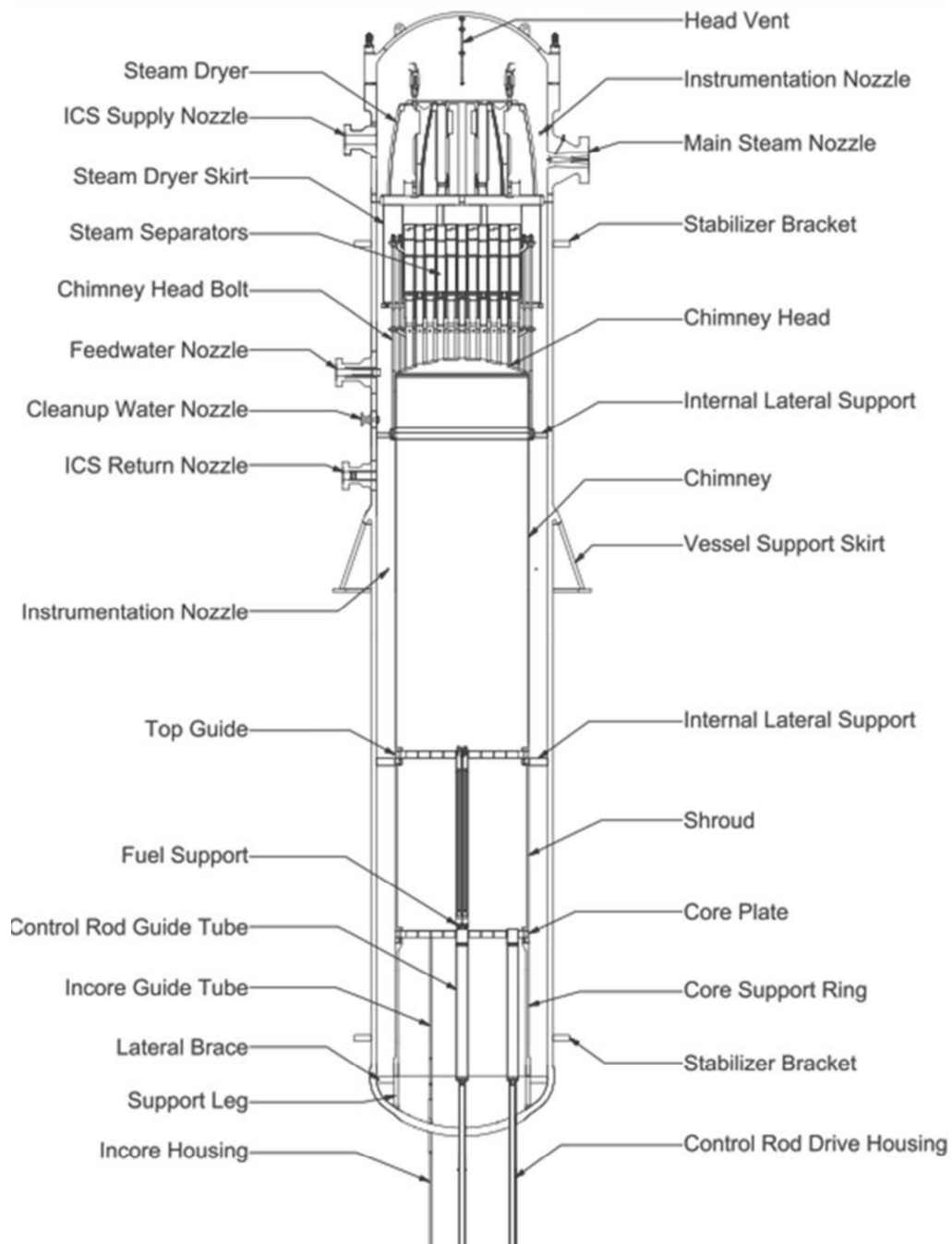
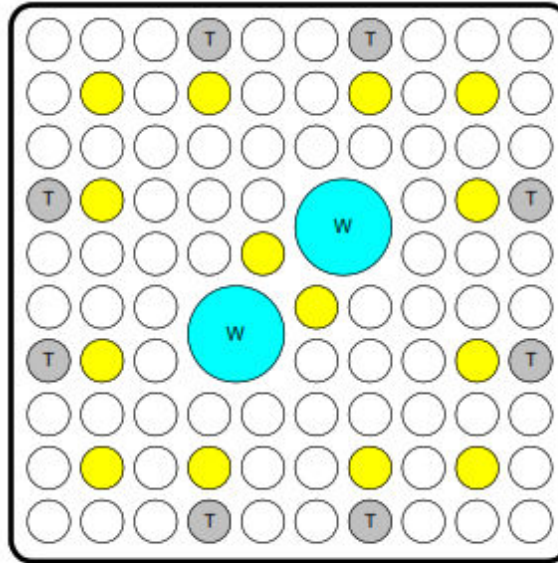
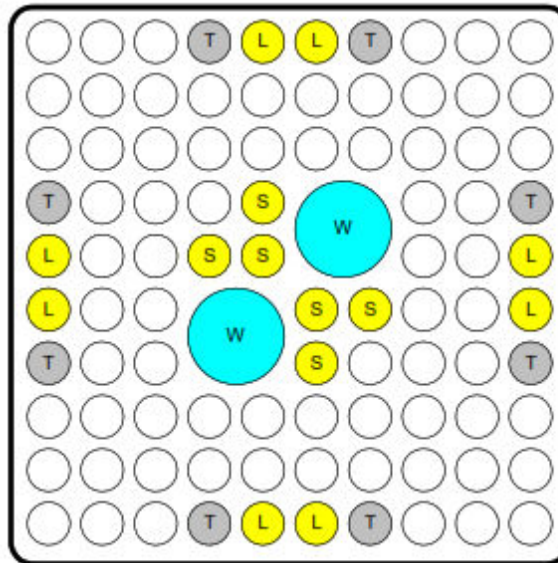


Figure 4.1-1: BWRX-300 Reactor Pressure Vessel and Internals

GE14



GNF2








-  water rod
-  PLR - 2133.6 mm
-  PLR - 2590.8 mm
-  PLR - 1371.6 mm
-  tie rod

Figure 4.1-2: GNF2 and GE14 Lattice Array

4.2 Fuel System Design

The fuel system consists of the fuel assembly and the reactivity control assembly (i.e., the control rods). The fuel assembly is comprised of a fuel bundle, a channel that surrounds the fuel bundle, and a channel fastener that attaches the channel to the bundle. The fuel bundle is comprised of fuel rods (some of which contain burnable neutron absorbers), upper and lower tie plates, water rods, spacers, springs, and assembly fittings.

The BWRX-300 reactor core is fueled with commercially available BWR fuel currently in production for forced circulation BWRs (e.g., GE BWR/4-6 and Advanced Boiling Water Reactor (ABWR)). The reference fuel design for reactor licensing is GNF2 (Figure 4.2-1). GNF2 is the third evolution of the Global Nuclear Fuel (GNF) 10x10 fuel design and was first deployed in batch quantities in 2007

GNF2 fuel conforms with the requirements stipulated in Licencing Topic Report “General Electric Standard Application for Reactor Fuel” (Reference 4.2-1). GEH BWR fuel designs have been developed to conform to the New Fuel Licencing requirements stipulated in GESTAR as per (Reference 4.2-1) NEDC-33270P. GESTAR is the framework for implementing U.S. regulatory requirements and consistent with international standards as evidenced by the multiple jurisdictions in which GNF fuel has been deployed. Furthermore, the requirements in GESTAR are evaluated in (Reference 4.2-2) which shows that the fuel meets the requirements in the CNSC REGDOC-2.5.2. (Reference 4.1-1).

4.2.1 Design Bases

4.2.1.1 Fuel Assembly Design Bases

The fuel assembly design bases are established to satisfy the performance and safety criteria presented in CNSC REGDOC-2.5-2 (Reference 4.1.-1).

The fuel assembly is designed to:

- Support self-sustaining fission chain reaction, thereby producing energy in the form of heat
- Maintain its integrity to retain fission products generated in the fuel during normal operation and operational transients or during Anticipated Operational Occurrence AOO conditions

The fuel rod design considers all applicable effects such as fuel density changes, fission gas release, clad creep, and other physical properties which vary with burnup. The integrity of the fuel rods is achieved by designing to prevent excessive fuel temperatures, internal gas pressure due to fission gas releases, cladding stresses, strains, and strain fatigue. The fuel rods are designed so that the conservative design bases of the limiting set of events envelop the lifetime operating conditions of the fuel. For each design basis, the performance of the limiting fuel rod, with appropriate consideration for uncertainties, does not exceed the limits specified by the design basis (as discussed in Section 4.2). The detailed fuel design also establishes such parameters as pellet size and density, clad/pellet diametral gap, gas plenum size, and helium pre-pressurization level.

The fuel assembly structure integrity is assured by setting limits on stresses and deformations due to various loads and by preventing the assembly structure from interfering with the functioning of other components. The fuel assembly is designed to withstand the following loads:

- Non-operational loads occurring in shipping and handling
- Normal and abnormal loads occurring in startup testing, normal operation, AOOs
- Abnormal loads occurring in infrequent events and accidents

Chapter 4 establishes the fuel design criteria. Specifically, Section 4.2 identifies fuel damage criteria. Section 4.3 establishes fuel criteria for axial offset anomaly. Section 4.4 provides specific thermal-hydraulic criteria.

4.2.1.2 Control Rods Design Bases

The control rods are designed to control the fission chain reaction. The rods, along with the control rod drive system (Section 4.6), provide stable and automatic control of reactor core power, spatial instabilities, and local power density during normal operation. The control rods also shut down the reactor and maintain the core subcritical.

The control rod design meets the following acceptance criteria:

- Control rod stresses, strains, and cumulative fatigue are evaluated to not exceed the ultimate stress or strain limit of the material, structure, or welded connection
- The control rod design is evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses
- Control rod materials are shown to be compatible with the reactor environment
- Control rod reactivity worth is included in the plant core analyses

The following bases are established for the above acceptance criteria.

Stress, Strain, and Fatigue

The control rod design is evaluated to assure that it does not fail because of loads due to shipping, handling, normal operation, including the effects of AOOs, Postulated Accidents (PAs) and Design Extension Conditions (DECs). To ensure that the control rods do not fail, these loads must not exceed the ultimate stress and strain limit of the material, structure, or welded connection. Fatigue must not exceed a fatigue usage factor of 1.0.

The loads evaluated include those from normal operational transients (scram and rod maneuvering), pressure differentials, thermal gradients, flow, and system induced vibration, and irradiation growth in addition to the lateral and vertical loads expected for each condition. Fatigue usage is based upon the cumulative effect of the cyclic loadings. The analyses include corrosion and crud deposition as a function of time, as appropriate.

Conservatism is included in the analyses by including margin to the limit or by assuming loads greater than expected for each condition. Higher loads can be incorporated into the analyses by increasing the load itself or by statistically considering the uncertainties in the value of the load.

Control Rod Insertion

The control rod design is evaluated to assure that it can be inserted during normal operations including the effects of AOOs, PAs and DECs. These evaluations include a combination of analyses of the geometrical clearance and actual testing. The analyses consider the effects of manufacturing tolerances, swelling and irradiation growth.

Control Rod Material

The external control rod materials must be capable of withstanding the reactor coolant environment for the life of the control rod. Effects of crud deposition, crevices, stress corrosion and irradiation upon the material must be included in the control rod design and core evaluations. Irradiation effects to be considered include material hardening and absorber depletion and swelling.

Reactivity

The reactivity worth of the control rod design is determined by the initial amount and type of absorber material and irradiation depletion. Scram time insertion performance must also be included in the plant core analyses including the effects of normal operations, AOOs, PAs and DECAs.

4.2.2 Fuel Assembly and Control Rods Description

4.2.2.1 Fuel Assembly Description

The reference GNF2 fuel assembly components are shown on Figure 4.2-1, and consists of a fuel bundle, a channel that surrounds the fuel bundle, and a channel fastener that attaches the channel to the bundle. The fuel rods and water rods are spaced and supported by upper and lower tie plates and intermediate spacers. The lower tie plate has the function of supporting the fuel assembly in the reactor. The upper tie plate has a handle for transferring the fuel bundle from one location to another.

The identifying fuel assembly serial number is engraved on the top of the handle with no two assemblies bearing the same serial number. This unique numbering supports the programmatic requirements for nuclear material accountancy required in CNSC REGDOC-2.13.1, Safeguards and Nuclear Material Accountancy, Section 7 (Reference 4.2-12). A projection from one side of the handle ensures proper orientation of the assembly in the core.

The materials used in the GNF2 fuel assemblies are listed in Table 4.2-1. Water chemistry controls used to minimize adverse effects on materials are described in Chapter 9A, Sections 9A.8, 9A.9, and 9A.10 and Chapter 13, Subsection 13.3.2.

4.2.2.2 Fuel Bundle

The GNF2 fuel bundle is comprised of fuel rods that contain a cladding tube that house the UO_2 fuel pellets with some UO_2 pellets containing gadolinia. Each fuel rod is hermetically sealed with welded upper and lower end plugs. The cladding and end plug material is Zircaloy-2 with a zirconium liner for pellet-clad interaction resistance. All fuel rods are inerted with helium before sealing.

4.2.2.2.1 Normal Full-Length Rods

There are 70 normal full-length rod locations in the GNF2 fuel bundle and reside in holes in the upper and lower tie plates. An expansion spring is installed onto the upper end plug that interacts with the upper tie plate exerts a downward force maintaining the axial position of the fuel rod while accommodating irradiation growth. The active fuel length is 381 cm. The plenum also contains a retention spring that protects the column of fuel pellets during transportation from the manufacturing facility to the reactor site.

4.2.2.2.2 Tie Rods

Tie rods are normal full-length rods except both the upper and lower end plugs are threaded. The lower end plug is screwed into matching threads that are machined into the lower tie plate. The tie rod upper end plug protrudes through the upper tie plate and is secured with nuts. There are eight (8) tie rods located symmetrically along the bundle periphery.

4.2.2.2.3 *Gadolinia (Burnable Absorber) Rods*

Gadolinia rods are essentially normal full-length rods with normal fuel pellets except sections of the fuel column contain pellets with Gd₂O₃ homogeneously blended with the UO₂ powder. The resultant pellets that function as a burnable neutron absorber controlling excess reactivity in the fresh fuel. The Gd₂O₃ concentration is effectively depleted by the end of the first bundle operation cycle and is cycle length dependent.

4.2.2.2.4 *Part Length Rods*

Partial length fuel rods were introduced to reduce the two-phase pressure drop for improved stability (i.e., lower decay ratios as described in Section 4.8) and retained in all subsequent fuel products.

Additionally, partial length fuel rods improve nuclear efficiency by better matching the axial hydrogen-to-fissile uranium ratio (H/U) ratio in BWR fuel with axially varying moderator density. Partial length fuel rods also reduce core reactivity in the cold condition and increase cold shutdown margins.

GNF2 has eight (8) long-partial length fuel rods and six (6) short-partial length fuel rods that have fuel column lengths that are approximately two thirds (2/3) and one third (1/3) of 381 cm (150-inch) active fuel length, respectively. The two lengths are used for fuel efficiency to better match the axial H/U ratio axially as the moderator density decreases. The long partial length fuel rods in GNF2 are located adjacent to the intra-assembly bypass gap (i.e., bundle periphery) and the short partial length fuel rods are located adjacent to the central water rods for improved reactivity margins.

4.2.2.2.5 *Water Rods*

The GNF2 assembly is designed with two large circular water rods that are centrally located and occupy 8 fuel rod lattice positions (i.e., each water rod occupies 4 fuel rod locations).

These holes allow water to be channeled from the bottom end of the bundle to the upper part of the bundle, without boiling, for improved neutron moderation which increase moderation and results in improved reactivity and improves cold shutdown margins. The water rods are hollow Zircaloy tubes with several holes around the circumference near each end allowing coolant flow through. The water rod flow is controlled by the number and diameter of the inlet holes at the lower end. Both water rods are screwed into the lower tie plate, and one is designed to position the eight spacers axially. Tabbed water rods that position the spacers are designed with a D-shaped upper end plug and with spacer positioning tabs. These tabs are welded to the tube exterior above and below each spacer location (see Figure 4.2-1). The tabbed water rod (spacer-positioning) is prevented from rotating by the engagement of its D-shaped upper end plug with a D-shaped hole in the upper tie plate. The non-tabbed water rod has a round shank upper end plug and no spacer-positioning tabs. An expansion spring is located between the water rod shoulder and upper tie plate allowing for differential axial expansion.

The single-phase (i.e., liquid) water that flows through the water rod increases neutron moderation and further thermalizes the neutron energy spectrum, improving nuclear efficiency. Water rods in the fuel assemblies also function to increase moderation for improved reactivity and improved cold shutdown margins.

4.2.2.2.6 Spacers

The GNF2 fuel design uses 8 spacers with integral springs made of Alloy X-750. The integral spring construction results in significantly fewer parts. The spacer spring force prevents fretting wear on the fuel rods due to fuel rod vibration. Flow diversion devices added to the top of the spacer improves liquid droplet deposition onto the surface of the fuel rods in the two-phase flow region.

The GNF2 bundle design has a non-uniform axial spacing of the 8 spacers optimizing critical power performance. The spacers axial separation in the lower region of the bundle prevents excessive fuel rod bow during operation. The spacers are positioned closer to each other in the upper region of the bundle for increased liquid droplet deposition in the annular flow regime. The GNF2 spacer exhibits low hydraulic resistance (i.e., pressure drop). Alloy X-750 is resistant to hydrogen pickup and maintains strength and ductility through the operating lifetime and while in storage.

4.2.2.2.7 Upper Tie Plate

The type-304 stainless steel upper tie plate supports the weight of the fuel assembly and positions the rod ends laterally during operation and handling. The upper tie plate has an internally threaded corner post fabricated to accept the channel fastener bolt. The upper tie plate is synergistically designed with two large central water rods and the partial length fuel rods maximizing flow area and minimizing two-phase pressure-drop.

4.2.2.2.8 Debris Filter Lower Tie Plate

The type-304 stainless steel lower tie plate, in conjunction with the upper tie plate, supports the weight of the fuel assembly and positions the rod ends laterally during operation and handling. Figure 2.7 of NEDC-33270, GNF2 Advantage Generic Compliance with NEDE-24011-PA (GESTAR II), (Reference 4.2-1) is an assembly consisting of a machined casting, non-removable debris filter cartridge, and a welded cover plate. The debris filter component is specifically designed to maximize filtration efficacy and the hydraulic resistance (i.e., pressure drop) is explicitly modeled in core performance and safety analyses.

4.2.2.2.9 Channel and Channel Fastener

The BWR fuel channel (Figure 4.2-1) performs the following functions:

1. Forms the fuel bundle coolant flow path outer periphery
2. Provides a surface for control rod guidance in the reactor core
3. Provides structural lateral stiffness to the fuel bundle
4. Controls, in conjunction with the lower tie plate, coolant bypass flow at the channel/lower tie plate interface

The channel is open at the bottom and makes a sliding seal fit over the lower tie plate. At the top of the channel, two opposite corners have welded clips. These clips support the weight of the channel on the upper tie plate posts. One of the clips has a hole for attaching the channel fastener to the bundle. The channel design incorporates a uniform thickness bottom end. The remainder of the channel has corners, sidewalls and sidewall grooves at the control blade roller, and symmetric locations providing sufficient strength in the regions of highest stress while minimizing material that absorbs neutrons.

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The overall stiffness of the fuel assembly is almost entirely a function of the channel stiffness. The GNF2 channel stiffness and the fuel assembly stiffness is essentially the same as previous fuel assemblies. The NSF (GNF proprietary zirconium-based alloy channel material) that is standard for GNF2 was introduced in reload quantities in 2016 following U.S. Nuclear Regulatory Commission (USNRC) approval. The design mitigates excessive channel deformation in-service and minimizes channel/control blade interference. The technical basis for replacing Zircaloy-2 and Zircaloy-4 with the NSF channel material is provided in NEDE-33798 (Reference 4.2-3).

The GNF2 channel fastener assembly consists of a stainless-steel casting (or fastener guard), a one-piece Alloy X-750 leaf spring with two active leaves at right-angles, a spring lock washer, and a fastener bolt attaching the fuel channel to the upper tie plate.

The channel fastener casting (or guard) fits over the top of the channel and bolts through the channel clip into the upper tie plate. The fastener guard serves as a reaction support for the leaf springs, provides a captive housing and lead-in for the fastener spring, and protects the springs from being overstressed. The fastener bolt attaches the channel to the bundle and remains captive in the casting, even if the fastener bolt were to fail.

4.2.2.3 Control Rods

The control rods (Figure 4.2-2) perform dual functions of power distribution shaping and reactivity control. Power distribution in the core is controlled during operation of the reactor by manipulation of selected patterns of control rods. These rods are positioned to counterbalance steam voids in the top of the core and effect significant power flattening. These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all control rods be available for either reactor “scram” (i.e., prompt shutdown) or reactivity control. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, electro-hydraulically actuated drive mechanisms that allow either electric motor controlled axial positioning for reactivity regulation or hydraulic scram insertion. The design of the rod-to-drive connection permits each control rod to be coupled or uncoupled from its drive without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and remain operable for tests with the reactor vessel open.

The core reactivity control requirements are met by use of the combined effects of the movable control rods, supplementary burnable neutron absorber (i.e., Gadolinia Rods in the fuel bundle described in Subsection 4.2.2.2.3), and the reactor coolant natural flow.

The control rod main structure consists of a top handle, an absorber section, and a bottom connector assembled into a cruciform shape. The top handle contains a grapple opening for handling. The absorber section is an array of stainless-steel tubes filled with boron carbide powder capsules or a combination of boron carbide powder capsules and hafnium rods. The connector is positioned on the bottom of the control rod for coupling to the control rod drive. While being inserted into the core, the control rod is restricted to the cruciform envelope created by the fuel bundles. Handle pads guide the control rod along the channels and connector rollers guide the control rod within the guide tube as the control rod is inserted and withdrawn from the core. Configuration of the control rod is shown in Figure 4.2-3.

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The BWRX-300 employs the Ultra-HD control rod, which is based on the Ultra-HD control rod designed for the BWR/2 through BWR/6. That design has been approved by the USNRC and applied to operating plants. The GEH Ultra-HD control rod is a derivative of prior designs incorporating the full-length hafnium rods in outer edge, high depletion tube locations. A detailed description is given in (Reference 4.2-4), NEDE-33284 Supplement 1P-A, "Marathon-Ultra Control Rod Assembly."

4.2.3 Fuel Design Evaluation

The GNF2 fuel design is the result of over 50 years of BWR fuel design, fabrication, and operational experience. The BWR fuel design process, comprised of engineering methods, analyses and test, is mature and generically applicable to the BWRX-300. The methods applied in the design of BWR fuel are documented in various Licencing Topical Reports that were submitted to the USNRC as needed. In addition, GNF developed a fuel licensing framework with the USNRC, and other regulatory authorities, called GESTAR, NEDE-24011-P-A-31 (Reference 4.4-8). The GESTAR licensing framework defines the generic requirements and approved methods for designing BWR fuel.

Most of the fuel rod thermal mechanical design analyses are performed using the PRIME fuel rod thermal-mechanical methodology. The method, qualification and application of the PRIME methodology are described in NEDC-33256P-A, NEDC-33457P-A, NEDC-33458P-A and NEDC-33840P-A (References 4.2-5, through 4.2-8). The PRIME methodology is approved for application to AOO transients (Reference 4.2-8, NEDC-33840P-A). This methodology is applicable to the analysis of the fuel rod response for all transient events and has been qualified for performing fuel analysis for BWRX using US NQA-1 standard, which is equivalent to Canadian standard CSA N286.7. PRIME analyses are performed for the following conditions:

1. For each analysis, fuel rod input parameters are based on either the most unfavorable manufacturing tolerances (i.e., 'worst tolerance' analyses) or statistical distributions of the input values. Calculations are performed providing either a 'worst tolerance' or statistically bounding tolerance limit for the resulting output parameter(s)
2. Operating conditions are postulated that cover the conditions anticipated during normal operating conditions and AOOs

Fuel rod design evaluations establish an upper bound power history envelope for the different fuel rod types. These power histories are used for fuel rod thermal mechanical design analyses evaluating the fuel rod design features and demonstrating conformance to the design criteria. These power histories are applied as a design constraint in the development of any BWR core design, including the reference BWRX-300 core described in 4.3.

The PRIME analysis assumes that during the fuel rod operating lifetime, the fuel rod (axial) node with the highest power operates on the limiting power-exposure envelope.

4.2.3.1 Worst Tolerance Analyses

The analyses apply worst tolerance assumptions to the cladding circumferential strain during an AOO. In this case, the important PRIME inputs are all biased to the fabrication tolerance extreme in the direction that produces the most limiting circumferential strain. The biases are discussed in detail in NEDC-33258P-A (Reference 4.2-7).

4.2.3.2 Statistical Analyses

For PRIME analyses are performed using standard error propagation statistical methods. The statistical analysis procedure is presented in NEDC-33258P-A (Reference 4.2-7).

4.2.3.3 Fuel Lift and Seismic and Dynamic Load Analysis

The fuel lift and seismic and dynamic load analyses are completed prior to fuel release for transportation. The fuel lift seismic and dynamic load analysis evaluates whether these loads are sufficient to unseat the fuel assembly from the lower tie plate, and if so, the maximum vertical lift distance and maximum fuel dynamic acceleration resulting from reseating. The acceptance criteria for this analysis are:

1. Vertical fuel lift is less than the engagement between the fuel support and the lower tie plate
2. Peak horizontal and vertical accelerations are less than the fuel demonstrated acceleration capabilities

4.2.3.4 Cladding Strain

The cladding strain analysis is performed using the PRIME code and the worst tolerance methodology discussed previously. For each fuel rod type, as described in Subsection 4.2.2.2, the cladding strain is calculated at various exposure points where transient overpower is assumed relative to the limiting power history. The magnitude of the overpower event is increased until the cladding strain approaches, but does not exceed, the limits described in NEDC-33258P-A (Reference 4.2-7) at the most limiting exposure to establish the mechanical overpower. Mechanical overpowers may also be defined as a function of exposure, ensuring that cladding strain does not exceed the limits described in NEDC-33258P-A (Reference 4.2-7) at any exposure point.

4.2.3.5 Fuel Rod Internal Pressure

The fuel rod internal pressure analysis is performed using the PRIME code and the statistical methodology discussed previously. The fuel rod internal pressure nominal value and standard deviation are determined at various fuel rod exposure points. At each of these exposure points, the nominal values and standard deviation of the fuel rod internal pressure that would cause the cladding to creep outward at a rate equal to the fuel pellet irradiation swelling rate (the critical pressure) are calculated. A design ratio is defined from the rod internal pressure and the critical pressure with their standard deviations. This design ratio limit of no more than 1.0 at a 95% confidence level, ensures that the fuel rod cladding does not creep out at a rate greater than the fuel pellet irradiation swelling rate.

4.2.3.6 Fuel Pellet Temperature

The fuel pellet temperature analysis is performed statistically using the PRIME code. For each fuel rod type the fuel pellet centerline temperature is statistically calculated at various exposure points assuming an overpower relative to the limiting power history. The lower 95% value for margin to fuel melting is calculated based on these statistical distributions. The overpower event magnitude is further increased until the lower 95% melt margin approaches zero but remains positive at the most limiting exposure point. The result from this analysis establishes the thermal overpower discussed below. Thermal overpowers may also be defined as a function of exposure, ensuring that lower 95% melt margin remains positive at all exposure points.

4.2.3.7 Cladding Fatigue Analysis

The cladding fatigue analysis is performed statistically using the PRIME code. Variations in power, coolant pressure and coolant temperature are superimposed on the limiting power history for calculating the cladding fatigue.

Cladding strain cycles are analyzed in NEDC-33270P (Reference 4.2-1) using the fuel duty cycles shown in NEDC-33258P-A (Reference 4.2-7). The duty cycles represent conservative assumptions regarding power changes anticipated during normal reactor operation and AOOs, planned surveillance testing, normal control blade maneuvers, shutdowns and load following. Fatigue analyses explicitly account for daily load following using the BWRX-300 plant duty cycles specification, with 347 cycles per year. The analyses of GNF2 fuel show that the cladding fatigue capability accommodates the expected fatigue duty for the projected life and operation of the fuel with sufficient margin (Reference 4.2-11, NEDC-33941P).

4.2.3.8 Cladding Creep Collapse

Cladding creep collapse occurs when excessive in-reactor fuel pellet densification causes fuel column axial gaps. This phenomenon has been observed in some pressurized water reactor fuel rods. The cladding creep collapse analysis consists of a detailed finite element mechanics analysis of the cladding. This evaluation is described in NEDC-33139P-A (Reference 4.2-9).

4.2.3.9 Fuel Rod Stress Analysis

The fuel rod stress analysis is performed using the Monte Carlo statistical methodology and addresses local fuel rod stress concerns, such as the stresses at spacer contact points, that are not directly addressed by the PRIME code. Results from PRIME analyses are used to generate inputs for the stress analysis. The cladding stress analysis is described in NEDC-33270-P (Reference 4.2-1).

4.2.3.10 Thermal and Mechanical Overpowers

Analyses are performed to establish the values of the maximum overpower magnitudes that do not result in violation of the cladding circumferential strain criterion or the incipient fuel centerline melting criterion. As part of the core design and AOO analysis, the calculated core mechanical and thermal overpowers defined in NEDC-33258P-A (Reference 4.2-7) and NEDC-33840P-A (Reference 4.2-8) are compared with the thermal overpower and mechanical overpower criteria and confirm conformance to these criteria. PRIME transient analyses performed per NEDC-33840-P-A (Reference 4.2-8) may also use worst-tolerance and statistical assumptions, consistent with those used in determination of thermal overpower and mechanical overpower criteria. This method confirms compliance by comparing worst-tolerance strain directly to strain limits and lower 95% melt margin criterion.

4.2.3.11 Fretting Wear

Testing assures that the mechanical design features, particularly those related to spacers and tie plates, do not result in significant vibration and consequent fretting wear, particularly at spacer–fuel rod contact points. The GNF2 fuel design vibration response has been compared to a reference design that has demonstrated satisfactory performance through discharge exposure.

4.2.3.12 Water Rods

Analyses are performed to determine component stresses at the bounding load conditions and compared to applicable criteria, such as yield and ultimate stresses. The load conditions consider shipping and handling loads, seismically induced bending moment, and the pressure differential across the water rod. The design is also evaluated using finite element analysis to determine the critical buckling load to ensure axial loads resulting from differential growth of water rods and other fuel assembly components are adequate.

4.2.3.13 Tie Plates

Tie plate adequacy is demonstrated by detailed finite element analysis and mechanical testing for bounding fuel handling and seismic load conditions.

4.2.3.14 Spacers

Tests are performed to demonstrate that the spacer design can withstand design basis loading without any significant deformation. The bounding load condition is seismic loading. Tests are conducted to demonstrate spacer fatigue capability and compliance with load limits and to demonstrate that a coolable geometry is maintained by showing minimal deformation at the combined load condition. Fretting wear is addressed by performing flow induced vibration tests and evaluating the results relative to spacer designs that have demonstrated acceptable performance. The results of flow induced vibration testing for GNF2 are summarized in NEDC-33270P (Reference 4.2-1).

4.2.3.15 Channel

Channel adequacy relative to applicable design criteria is confirmed by performing the following evaluations:

- Calculating elastic stress and deflection caused by channel wall pressure difference
- Calculating thermal stresses due to the various temperature gradients that the channel is subjected to during normal operation and handling
- Calculating fatigue and stress rupture considering the combined effect of pressure temperature cycling and hold time
- Calculating elastic-plastic and creep of channel wall permanent deflection
- Calculating channel stress due to control rod contact
- Analyzing channel/lower tie plate differential thermal expansion

The GNF2 Fuel Assembly Mechanical Design Report for BWRX-300 (Reference 4.2-10) provides additional discussion addressing the design features that preclude excessive channel bowing from preventing control blade insertion.

4.2.3.16 Conclusion

The BWRX-300 reactor is designed to accept conventional BWR fuel and GNF2 has been established as the reference fuel design. Fuel mechanical and fuel rod thermal mechanical evaluations, summarized in NEDC-33270P (Reference 4.2-1), demonstrate compliance with regulatory requirements and are applicable to GNF2 operation in the BWRX-300. Detailed technical information pertaining to fuel assembly mechanical (NEDC-33940P) and fuel rod thermal-mechanical performance (NEDC-33941P) is provided in (References 4.2-10 and 4.2-11).

4.2.4 Control Rods Design Evaluation

The control rod design is evaluated against the acceptance criteria and bases described in Subsection 4.2.1.2. Design compliance with these criteria constitutes the basis for acceptance and approval of the design. The control rods for the BWRX-300 are based on the design used in the operational BWR fleet. This well understood fleet operational data is utilized in the methods to design, evaluate, and analyze the control rods.

4.2.4.1 Scram

The dynamic loads on the control rods are bounded by the Fine Motion Control Rod Drive (FMCRD) imposed loads (scram loads) in the vertical direction. The BWRX-300 inoperative buffer loads are the highest vertical loads experienced by the control rod due to the high terminal velocity. The control rod is evaluated using a dynamic analysis in (Reference 4.2-4). The resultant loads are evaluated using the material properties and geometry for the area subject to the load. The effective stress is determined using distortion energy theory. The limit is the material ultimate stress or strain.

4.2.4.2 Seismic

Fuel channel deflections which result from seismic and Loss-of-Coolant Accident events impose lateral loads on the control rods (Reference 4.2-4). The BWR/2 through BWR/6, ABWR and BWRX-300 have similar channel lengths and deflections.

The seismic analysis is performed by evaluating the strain in the Ultra-HD control rod absorber section when deflected. During a seismic event, it is assumed the seismic deflections could be added to any preexisting channel bow. The absorber section strain has been analyzed for channel deflections due to seismic and channel bow deflections when deflected by a bounded value and found to be acceptable, (Reference 4.2-4).

Testing was performed on the ABWR Marathon control rod to confirm seismic scram capability. The ABWR Marathon control rod was tested at fuel channel oscillation amplitude amplitudes representing SSE conditions. The scram times were found to be acceptable, and the control rod was not damaged. The BWRX-300 channels will experience reduced creep bulge as the channel wall pressure differential (i.e., the pressure differential across the channel wall) is low due to natural circulation flow. Since the Ultra-HD geometry and bending stiffness are nearly identical to the tested Marathon, the ABWR Marathon control rod seismic scram capability testing is conservative in relation to the BWRX-300 conditions.

4.2.4.3 Stuck Rod

Compression caused by a stuck rod at the time of scram is controlled by the FMCRD. Assuming the FMCRD exerts the same compression loads, the control rod does not buckle if the entire load is applied to one wing, (Reference 4.2-4).

4.2.4.4 Absorber Irradiated Related Loads

The absorber containment licenced in NEDE-33284 (Reference 4.2-4) is applicable to the BWRX-300 Ultra HD control rod. The same methodology is used for the BWRX-300 Ultra HD control rod in (Reference 4.2-4). The absorber tube and B4C capsule design accommodates irradiation-induced swelling of boron carbide, such that a clearance exists between the inner capsule and outer absorber tube up to the maximum local 10B depletion. The analysis conservatively assumes worst-case dimensions plus a bounding boron carbide irradiation swelling rate. As such, all stresses on the outer absorber tube associated with boron carbide swelling are eliminated, which drastically reduces the likelihood of Irradiation-Assisted Stress Corrosion Cracking relative to previous BWR control rod designs. Absorber tube stresses due to helium gas generation and moisture vapor heating are considered. The resulting stresses due to helium pressure are small compared to tube material strengths. Therefore, the Ultra HD absorber containment design is adequate for the nuclear design life of the control rod.

4.2.4.5 Load Combinations and Fatigue

The BWRX-300 Ultra-HD control rod is designed to withstand load combinations including AOOs and fatigue loads associated with those combinations. Absorber tube loads are evaluated during a scram, in a cell with severe channel bow near end of control rod life when absorber burn-up helium gas generation is highest. Absorber tube loads are evaluated during a seismic event near the end of control rod life when absorber burn up helium gas generation is highest. Absorber section to connector welds and absorber section to handle loads are evaluated during a scram when the absorber helium gas build up is highest. Per NEDE-33284 (Reference 4.2-4), the BWRX-300 Ultra-HD control rod does not exceed the ultimate stress or strain limit of the material. Based on the reactor cycles, the combined loads are then evaluated for the cumulative effect of the cyclic loadings in (Reference 4.2-4). The fatigue usage is evaluated against a limit of 1.0.

4.2.4.6 Handling Loads

The BWRX-300 Ultra-HD control rod is designed to accommodate three times the weight of the control rod, (Reference 4.2-4).

4.2.4.7 Hydraulics

Inspection experience over 60 years has shown the control rod is not damaged by the vibrations or cavitations set up by coolant velocities and velocity distributions in the bypass region between fuel channels.

4.2.4.8 Materials

The absorber tubes are made from high purity 304, stainless steel. These tubes are loaded with either, 1) capsules containing B4C, or 2) Hafnium rods. Further description is provided in NEDE-33284 (Reference 4.2-4).

Materials selected for use in the Ultra-HD control rod components are chosen to minimize the component end of life radioactivity to reduce personnel exposure during handling on-site, and for final offsite shipping and disposal. All Ultra-HD control rod materials are low weight percent cobalt. The average niobium content for the handle and absorber section, less boron carbide and hafnium has a low weight percent.

Materials with low activation characteristics are specified for use in the control rod. Depleted control rods that are discharged are categorized as high-level radiological waste.

4.2.4.9 Nuclear Performance

The nuclear lifetime of the initial BWRX-300 control rod type is established as a 10 percent reduction in reactivity worth ($\Delta k/k$) relative to the initial reactivity worth in any quarter axial segment, (Reference 4.2-4).

4.2.4.10 Mechanical Compatibility

Similar to the control rods supplied for the ABWR and BWR/2 through BWR/6, the BWR X-300 Ultra-HD control rod is designed to be compatible with core and reactor internal interfaces.

The Ultra-HD control rod is designed to be compatible with the Control Rod Guide Tube cylindrical boundary, to provide a seat with the guide tube base during FMCRD removal, to provide lower guide rollers for smooth transitions, and to have clearance with the orificed fuel support for insertion and withdrawal from the core.

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The control rod coupling socket provides a compatible interface with the FMCRD. The coupling engages the FMCRD by rotating. With the FMCRD, Control Rod Drive Housing (CRDH), and Control Rod Guide Tube positively assembled, any orientation of the cruciform control rod between the fuel assemblies is a coupled position, and rotation to an uncoupled position is not possible during reactor operation. The four lobes of the FMCRD coupling spud are in line with the four wings of the control rod in the coupled position.

The control rod is designed to permit coupling and uncoupling of the control rod drive from below the vessel for FMCRD servicing without necessitating the removal of the reactor vessel head. The control rod is also designed to allow uncoupling and coupling from above the vessel using control rod handling tools.

The control rod is positively coupled to the FMCRD and is designed to remain coupled during all scrams and loading conditions, including inoperative buffer scram loads. The control rod withstands the loads induced by the FMCRD without exceeding the structural design criteria.

The control rod is dimensionally compatible with the fuel assemblies (unirradiated and irradiated). The control rod is guided, rotationally restrained, and laterally supported by the adjacent fuel assemblies. The control rod is designed and constructed to establish and maintain the alignment of the control rod drive line (Control Rod Drive Mechanisms, Control Rod Guide Tube, and the fuel assemblies) so that control rod insertion and withdrawal is predictable. The top of the active absorber of a fully withdrawn control rod is below the Bottom of the Active Fuel. Absorber gap requirements are placed on the control rod in the operating condition to be compatible with the core nuclear design requirements.

4.2.5 Inspection and Testing

4.2.5.1 Quality Assurance Program and Quality Control

The Quality Assurance Program applicable to the fabrication of fuel and control rod assemblies for the BWRX-300 is summarized in Chapter 17.

The program provides for control over activities affecting product quality, starting with design and development, continuing through procurement, materials handling, fabrication, testing, and inspection, storage, and transportation. The program also provides for training of personnel and for the auditing of activities affecting product quality through a formal auditing plan.

The mechanical drawings and product, process, and material specifications identify the inspections performed.

The following subsections describe the general quality control inspections.

4.2.5.2 Fuel System Components and Parts

The characteristics inspected depend on the component parts. The quality control includes dimensional and visual examinations, review of supplier documentation, material certification, and nondestructive examination. Once supplier processes have been qualified, changes are subjected to GNF review and approval. Suppliers are periodically audited to assure that the process and Quality programs at the manufacturing facilities meet the requirements for GE GNF's Quality program.

The material used in the BWRX-300 core is ultimately accepted and release by Quality Control.

4.2.5.2.1 Pellet

Pellet Inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Additional visual inspections are performed for cracks, chips, and surface conditions according to approved standards. Density is determined using qualified equipment with known verification and calibration standards based. Chemical analyses are performed on a specified sample basis throughout pellet production.

4.2.5.2.2 Fuel Rod Inspection

Fuel rod inspections consists of the following nondestructive examination techniques and methods, as applicable:

- Each rod is leak tested at the assembly level using a calibrated mass spectrometer, with helium being the detectable gas
- Fuel rod welds are inspected by ultrasonic test in accordance with a qualified technique and GE specifications
- Fuel rods are visually inspected for scratches, dents, and rod bow
- Dimension control on newly fabricated fuel rods is maintained by limiting the allowed variation on the dimensions of the supplied or fabricated components
- Reworked rods are dimensionally inspected prior to final release to verify acceptable length
- Fuel rods are inspected by scanning to verify proper enrichment, burnable absorber poison concentration, and zone length. Scanning also checks for the presence of misplaced pellets or gaps in the fuel column
- Traceability of rods and associated rod components is established by quality control

4.2.5.2.3 Assemblies

Components for each bundle are visually inspected prior to use in manufacturing.

Bundles are in-process verified during assembly and then final inspected prior to packing into the fresh fuel shipping container.

Shipping containers are licenced and refurbished prior to each use to assure compliance to the Safety Analysis Report.

4.2.5.2.4 Other Inspections

As part of the routine inspection operation, the following inspections are performed:

1. Tool and gauge inspection and control, including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on serialized tools. Complete records are kept of calibration and conditions of tools.
2. Audits are performed of inspection activities and records to confirm that prescribed methods are followed and that records are correct and properly maintained.
3. Surveillance inspection, where appropriate, and audits of outside contractors are performed to confirm conformance with specified requirements.

4.2.5.2.5 Process Control

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are implemented.

The uranium dioxide powder is kept in sealed containers. The contents are fully identified both by descriptive tagging and unique barcode numbers. Isotopic content is confirmed by analysis.

Powder mixing (enrichment blending or homogenization) is performed to a system generated blend plan. The production control system verifies that only containers in the active blend plan are moved to the process station. The production control system captures the actual weight of each addition to the blend and will abort the blend plan if the weight is out of tolerance. Personnel are assigned roles based upon their work area after qualification/certification; unqualified personnel cannot perform a task as the production software system will prevent the transaction.

Finished pellets are placed on trays identified with a unique barcode identifier that maintains traceability in the production control system. Samples from each pellet lot are tested for isotopic content and impurity levels based upon a sampling plan prior to release of the pellets to rod loading.

Loading of pellets into the cladding is controlled by the production control system which receives the nuclear design from engineering. Each rod is verified to have the correct mass, enrichment, burnable absorber content, and zone length of fuel when the pellets are loaded into the cladding. Traceability is maintained to the pellet trays used to load each zone. Each cladding tube is barcoded, and this barcode is used by the production control system for traceability. An alpha-numeric serial number is placed on the lower plug; this serial number is linked to the barcode identification at the lower end plug ("first") weld process. The upper end plug is inserted and then welded (in an inert gas atmosphere) to seal the tube; during the upper end plug welding the rod is pressurized with Helium as specified by the fuel rod design.

4.2.5.3 Control Rod and Components

All raw material and purchased component parts are fully inspected and accepted for control rod manufacture. Inspection includes dimensional and visual examinations, review of supplier documentation, material certification, and nondestructive examination. Supplier processes are qualified and cannot be changed without GEH review and approval. Suppliers are periodically audited to assure that the processes and quality programs at the manufacturing facilities meet the requirements for GEH's quality program.

All material used in the BWRX-300 control rods is ultimately accepted and released by GEH supplier quality control.

4.2.5.3.1 Control Rod Absorber Materials

BWRX-300 control rods use both boron carbide powder and solid hafnium rods as neutron absorbers. Prior to use, boron carbide powder is inspected for chemical content, impurities, water content, isotopic content, and grain size. Several grain sizes are then mixed and vibratory compacted inside stainless steel capsules to achieve the specification-required average density. Similarly, hafnium rods are inspected for chemical content, impurities, and dimensions prior to use. An independent quality overcheck is performed to ensure the correct loading of all absorber materials into each control rod assembly.

4.2.5.3.2 Absorber Tube Inspection

The manufacture of drawn absorber tubes used in control rod assemblies is controlled to ensure defect-free tubes. Supplier raw material production, welding, drawing and heat treatment process, as well contacting material, and inspection processes, are controlled and approved by GEH. Absorber tubes are inspected both visually and using ultrasonic inspection. Traceability of all absorber tubes is maintained by raw material and tube manufacturing lot.

4.2.5.3.3 Control Rod Assemblies

The laser weld process used to join absorber tubes into the cruciform control rod assembly is tightly controlled. Periodic weld samples and process tracking are performed to ensure specification weld penetration requirements are met, and to ensure weld quality. Both visual and eddy current inspection is used to ensure continuous welds, with no holes or gaps in finished control rod assemblies. A helium leak check is used to further ensure the integrity of finished control rods.

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- 4.2-10 NEDC-33940P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Assembly Mechanical Design Report," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.2-11 NEDC-33941P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Assembly Thermal-Mechanical Design Report," GE-Hitachi Nuclear Energy Americas, LLC.
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Table 4.2-1: GNF2 Fuel Assembly Material

Component	Material (Reference 4.2-9, NEDC-33139P-A)
Fuel Assembly	
Fuel bundle spacers	Alloy X-750
Water Rods	Recrystallized Zircaloy-2
Tie Plates	Austenitic Stainless Steel
Fuel Rod	
Cladding	Recrystallized Zircaloy-2
Barrier	Recrystallized Zirconium
Pellet	UO ₂ and UO ₂ - Gd ₂ O ₃
Channel and Channel Fastener	
Channel	NSF (GNF Proprietary Zirconium-Based Alloy)
Guard Material	Austenitic Stainless Steel
Spring, cap screw and lock-washer material	Alloy X-750
	XM-19 SST

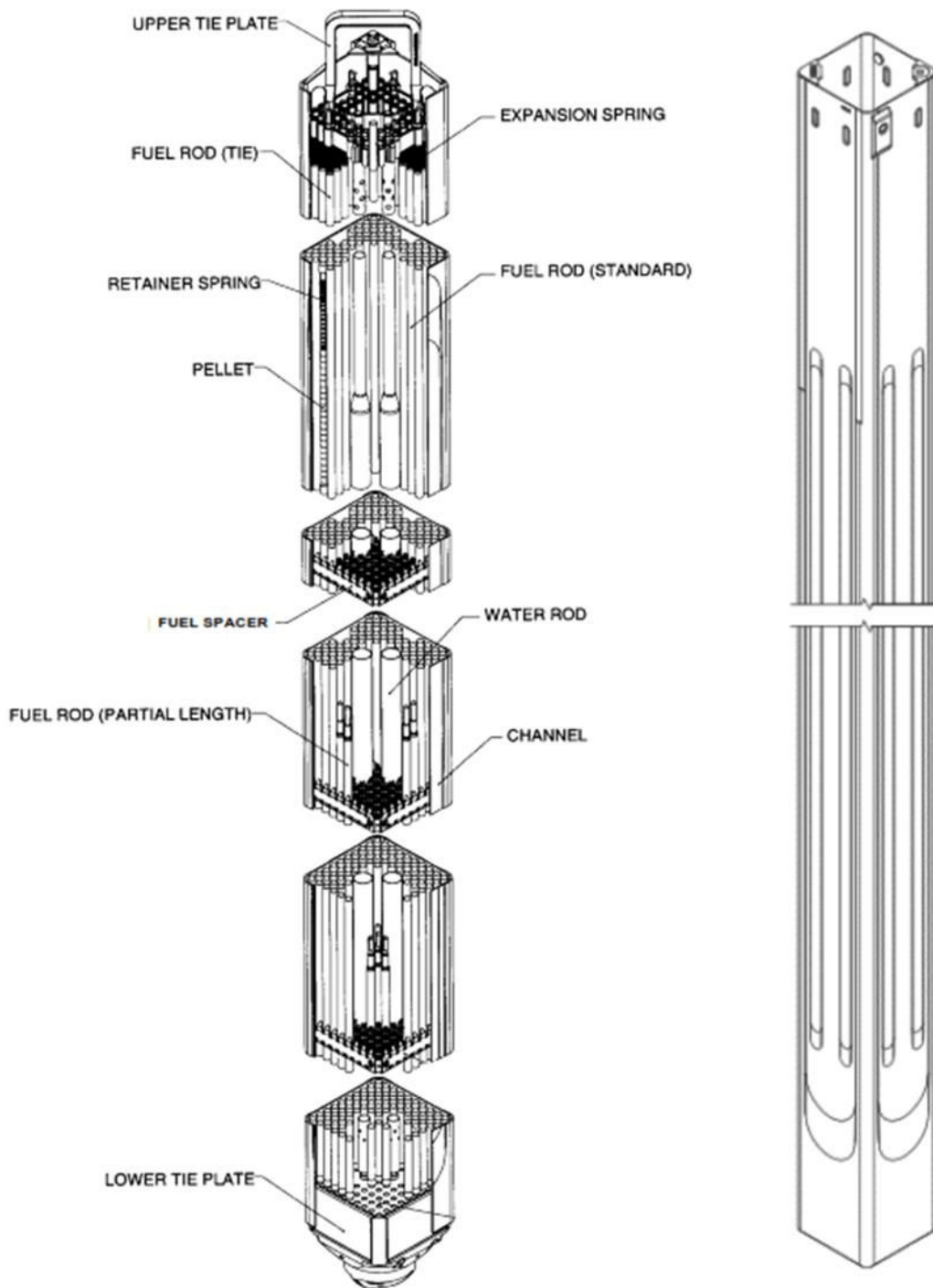


Figure 4.2-1: Fuel Assembly

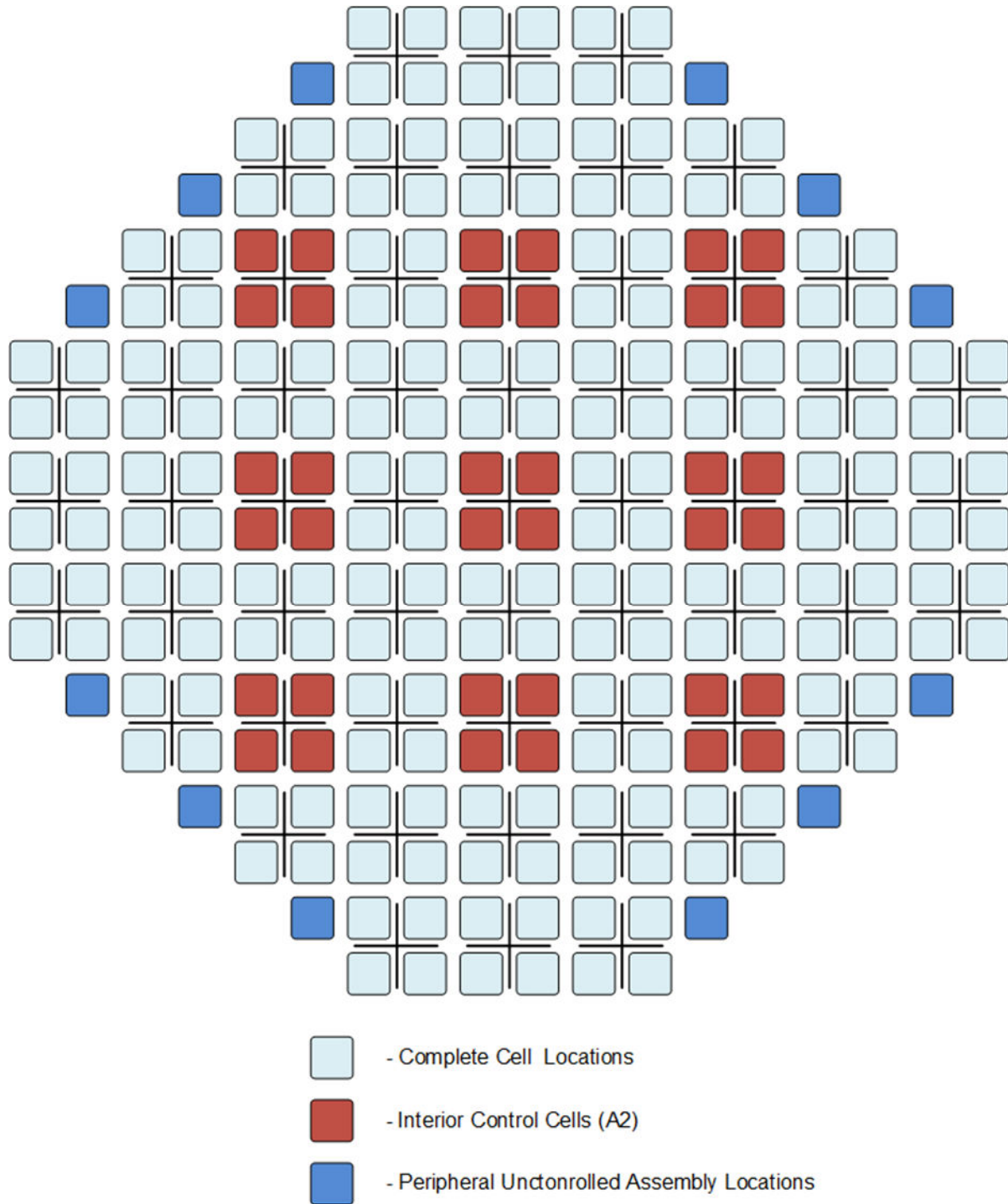


Figure 4.2-2: Ultra-HD-Control Rods Distributed Throughout Core

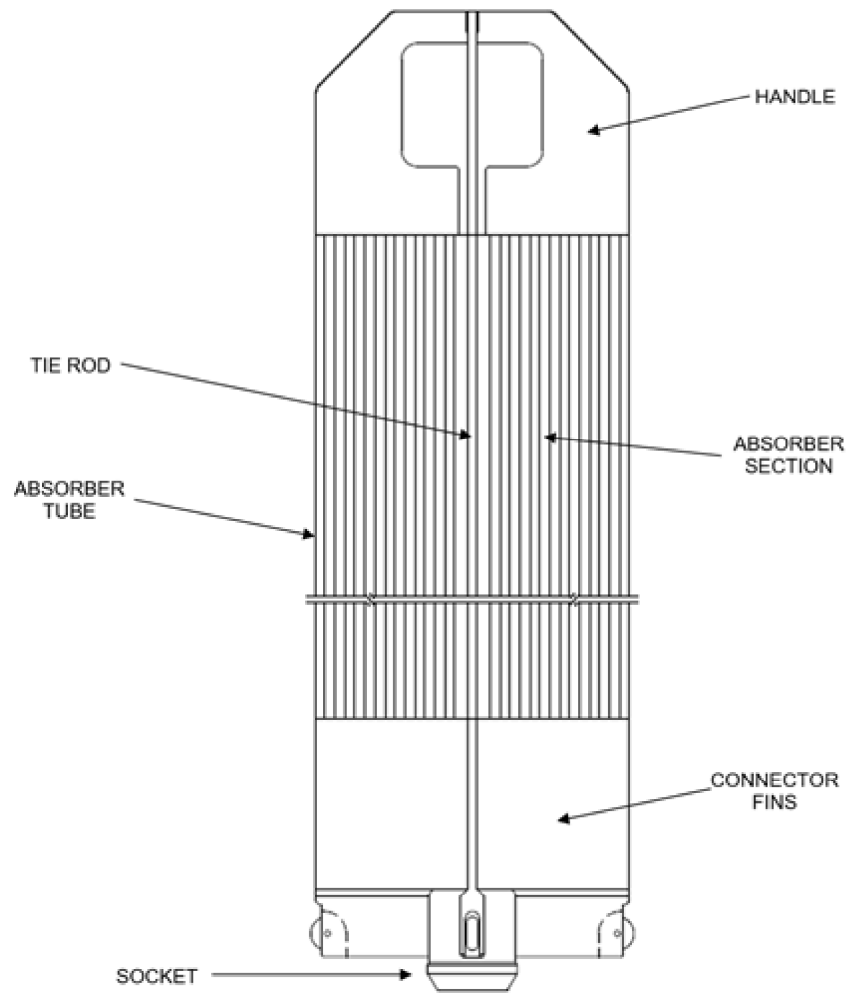


Figure 4.2-3 Ultra-HD-Ultra Control Rod

4.3 Nuclear Design

This section describes the design bases and functional requirements used in the nuclear design of the fuel, core, and reactivity control system that complies with the requirements of CNSC REGDOC-2.5.2, Revision 1, Section 8.1 (Reference 4.1-1). The Nuclear Design Report for BWRX-300 Equilibrium Cycle, NEDC-33985P (Reference 4.3-1) provides detailed results for the reference equilibrium cycle.

4.3.1 Design Bases

The design bases require the plant to operate while meeting all safety requirements:

1. The reactivity bases prevent an uncontrolled positive reactivity excursion and stipulate subcriticality requirements
2. The overpower bases ensure the core will operate within fuel integrity limits

4.3.1.1 Reactivity Feedback Bases

Reactivity coefficients representing the differential changes in reactivity produced by differential changes in core conditions are used for calculating stability parameters and evaluating the response of the core to external disturbances. The base initial condition of the system and the postulated initiating event determine which of the several defined coefficients are significant in evaluating the response of the reactor. The coefficients of interest are the Doppler, moderator temperature, and the moderator void reactivity coefficient. Also associated with the BWR is a power reactivity coefficient. The power coefficient is a combination of the Doppler and moderator void reactivity coefficients in the power operating range and is not explicitly evaluated.

The fuel reactivity acceptance criteria are established in GESTAR (Reference 4.4-8) and each of the following fuel parameters must be negative throughout the life of the core:

- Doppler reactivity coefficient for all operating conditions
- Core moderator void reactivity coefficient resulting from boiling in the active flow channels for any operating conditions
- Moderator temperature coefficient for temperatures equal to or greater than hot standby
- Power coefficient, as determined by calculating the reactivity change resulting from an incremental power change from a steady-state base power level for all operating power levels above hot standby
- Net prompt reactivity feedback originating from prompt heating of the moderator and fuel for a super prompt critical reactivity insertion accident (e.g., control rod drop accident)

The Doppler coefficient, the moderator void coefficient and the moderator temperature coefficient of reactivity are negative for power operating conditions, thereby assuring negative reactivity feedback characteristics.

4.3.1.2 Control Requirements (Shutdown Margins)

It is required that the core can be shut down and remain subcritical with the minimum margin as specified in the plant Operating Limits and Conditions (Chapter 16), with the highest worth control rod, or rod pair associated with the common Hydraulic Control Unit (HCU) postulated stuck in the full out position.

4.3.1.3 Control Requirements (Overpower Bases)

It is required that the core operate within an absolute power and power distribution envelope to ensure fuel integrity is maintained. This is controlled through two nuclear design basis parameters: the Maximum Linear Heat Generation Rate (MLHGR) Limit and Minimum Critical Power Ratio (MCPR). The MCPR and Linear Heat Generation Rate (LHGR) limit are determined with 95% confidence that the fuel does not exceed Operational Limits and Conditions during AOOs. These constraints must then be met under normal operating conditions.

Explicit Maximum LHGR Limit and MCPR parameter definitions are provided as follows:

Maximum Linear Heat Generation Rate Limit: The LHGR limit is the maximum allowable linear heat generation for each fuel rod in the bundle. The LHGR operating limit is bundle type-dependent and is a function of gadolinia content and exposure. The LHGR is monitored, and the fuel is not operated at MLHGR values greater than those found acceptable by the safety analysis under normal operating conditions. Under AOO conditions, including the maximum overpower condition, the calculated overpower is confirmed to neither cause fuel melting nor exceed the stress and strain limits as discussed in Subsection 4.2.3.

Minimum Critical Power Ratio: MCPR is the minimum CPR allowed for a given bundle type to avoid boiling transition. The CPR is a function of several important parameters: bundle power, bundle flow, rod power peaking distribution, and bundle mechanical design. The plant Operating Limit Minimum Critical Power Ratio (OLMCPR) is established by considering the limiting AOOs for each operating cycle. The OLMCPR is determined to avoid boiling transition for 99.9% of the rods during the limiting analyzed AOO transient discussed in Chapter 15, Subsection 15.5.3.

4.3.2 Core Nuclear Design Description

The core is light-water moderated and fueled with low enriched uranium dioxide fuel assemblies. The use of light water as a moderator produces a neutron energy spectrum where fissions are caused principally by thermal neutrons. At normal operating conditions, the moderator boils, producing a spatially variable distribution of steam voids in the core. The void reactivity feedback effect is an inherent safety feature of the BWR system. Any system change that increases reactor power, either locally or core-wide, produces additional steam voids and reduces power.

The reactor core is arranged as an upright cylinder containing fuel assemblies located within the core shroud. The coolant flows upward through the core. The reactor core includes fuel assemblies, control rods, and nuclear instrumentation. The arrangement of fuel assemblies/core loading map is the same as the forced flow Kernkraftwerk Mühleberg Nuclear Power Plant (KKM) BWR in Switzerland and is displayed in Figure 4.3-1.

The core nuclear design of any BWR core for an operating cycle is comprised of the following elements:

- Core arrangement, lattice type and fuel product line that combine to establish the detailed geometry of the reactor core
- The energy utilization plan that defines the energy requirements
- Core coolant hydraulics (e.g., core flow, pressure, and inlet temperature)
- Nuclear design (i.e., enrichment and burnable poison distribution) of the fresh fuel
- Core loading pattern of fresh and exposed fuel
- Planned control rod patterns during power operation

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An equilibrium cycle is a reactor state in which the same fresh fuel assemblies are loaded into the same locations, the exposed fuel is shuffled to the same locations and the cycle is depleted in the same way until an equilibrium state is achieved producing essentially identical results (e.g., energy generated, power distributions, reactivity margins, etc.) cycle after cycle. An equilibrium cycle is the best representation of the core over the life of the reactor. A reference equilibrium core as shown in Figure 4.3-2 has been developed for safety analysis to demonstrate that the BWRX-300 conforms to all regulatory requirements with high confidence. The reference BWRX-300 equilibrium cycle has been selected to be an annual cycle (i.e., 12-month refueling interval) with high, albeit normal, discharge exposure. Alternate refueling intervals (e.g., 18, or 24 months) may be applied to the BWRX-300 and conforms to all regulatory requirements.

The reference equilibrium cycle is loaded with multiple fresh nuclear fuel bundle types of various enrichments and gadolinia burnable poison designs that satisfy a multitude of requirements and objectives, including to optimize the core burnup while maintain core performance. The core loading pattern, operating control rod patterns, and core performance results are described in the Nuclear Design Report, NEDC-33985P (Reference 4.3-1) and the nuclear design of the fresh fuel assemblies is provided Fuel Bundle Information Report (Reference 4.3-6). The BWRX-300 core configuration, including a representative in-core instrumentation design and rod patterns, are depicted in Figure 4.3-1 and Figure 4.3-3 respectively. Select nuclear design information and core performance results are depicted in Figure 4.3-4 through Figure 4.3-9.

4.3.2.1 Refueling Interval

BWRs operate on distinct refueling intervals. At the end of a normal operating cycle, after shutdown, the reactor is disassembled, then generally the lowest reactivity fuel is discharged and new reload fuel is inserted and the fuel is shuffled into the core configuration for the upcoming cycle. The reactor is then reassembled, and the startup of the next cycle can commence. This activity, from shutdown of any operating cycle to startup of the next, is termed the refueling outage. Inspections and maintenance are also typically performed during the refueling outage on intervals specific to each piece of equipment or component.

For BWRs, there is not a specific limit on the length of an operating cycle, or planned refueling interval; however, requirements governing inspection and surveillance frequency must be satisfied and the safety analyses that constitute the safety basis must encompass the planned operating cycle length. Cycle specific operating limits are established to span the planned operating cycle length and provision is made for cycle extension (e.g., power coast down) as specified by the reactor owner.

Any planned operating cycle length must:

1. Support requirements governing inspections of the reactor internals & equipment
2. Reside within evaluated space (i.e., the technical inputs to the safety analyses must envelope the planned operating cycle length)

The cycle specific operating limits are derived from AOO analysis of the actual core design. Any refueling interval that results in acceptable thermal operating limits that conform to the Specified Acceptable Fuel Design Limits (SAFDL) can be supported.

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BWRs operate on approximately 12-, 18-, or 24-month refueling intervals; however, operating cycles longer than 24 months have been conducted as well as intermediate lengths. The reference BWRX-300 equilibrium core design to support safety and performance evaluations was established to be a 12-month cycle because it illustrates the highest degree of operational flexibility associated with excess thermal margins (e.g., Figure 4.3-7 shows ~30% excess thermal margin compared to a normal design target of MFLPD < 0.9), where MFLPD is the Maximum Fraction Limiting Power Density, which is equivalent to the maximum ratio of any LHGR in the core compared to its exposure dependent, LHGR limit).

The safety analyses that are generally affected by increasing the operating cycle length, along with areas that are part of the operating envelope, are denoted in Table 4.3-1.

4.3.3 Core Nuclear Analytical Methods

The analytical methods used in the design and analysis of the BWRX-300 core during all states of normal operation are summarized below and described in detail in NEDC-33939 "Steady State Nuclear Methods" (Reference 4.3-2).

4.3.3.1 Steady-State

The principal tools used in the steady-state nuclear core analysis are the three-dimensional (3-D) BWR Core Simulator PANAC11 and the two-dimensional lattice physics code TGBLA06. The BWR Core Simulator is a coupled nuclear-thermal-hydraulic computer program representing the BWR core exclusive of the external flow loop.

Natural circulation flow is determined by the Transient Reactor Analysis Code General Electric (TRACG) described in NEDE-32176P (Reference 4.3-3). The associated core flow is used as an input to the core simulator.

The simulator computes core reactivity, power distributions, exposure, and reactor thermal-hydraulic characteristics, with spatially varying voids, control rods, and burnable poisons as described in the nuclear libraries and other constitutive state variables. The simulator is used to calculate reactivity variations through the cycle, shutdown margins and compliance with thermal limits (i.e., LHGR Limit and MCPR).

PANAC11 and the associated nuclear libraries produced by TGBLA06 have undergone extensive validation by comparing calculated results with alternate methods, end-of-cycle gamma scan data, and operating reactor data. PANAC11 is a USNRC approved method used for production core design, licensing analysis, and core exposure tracking for the BWR fleet. The exposure tracking process provides the opportunity for continuous comparison and validation of the nuclear methods against operating data.

Validation of the adequacy of PANAC11 and associated compliance with CNSC guidance is demonstrated in NEDC-33939P (Reference 4.3-2).

4.3.3.2 Reactivity Coefficient

The lattice physics and 3D core simulator are used in computing the change in neutron multiplication (i.e., reactivity inserted) caused by a change in state (e.g., change in fuel temperature, in-channel void fraction, etc.) when determining the reactivity coefficients.

4.3.4 Core Nuclear Design Evaluation

4.3.4.1 Reactivity Coefficient

The safety analysis methods are based on system and core models that include an explicit representation of core space-time kinetics. Therefore, the reactivity coefficients are not directly used in the safety analysis methods but are useful in the general understanding and discussion of core response to perturbations.

The reactivity coefficients evaluation is performed as part of new fuel design development assuring consistency with safety objectives. The GNF2 results are documented in NEDC-33270P, GNF2 Advantage Generic Compliance with GESTAR II, which concludes that all the criteria defined in GESTAR II have been met for the GNF2 fuel design (Reference 4.2-1).

4.3.4.2 Reactivity Variation

The excess reactivity needed to deliver the target cycle energy while maintaining rated thermal power is controlled by the control rod system supplemented by fuel rods containing a burnable absorber. When applied to any specific fuel cycle, these integral fuel burnable absorber rods are used to provide partial control of the excess reactivity available during power operation. The burnable absorber lowers the reactivity of fresh fuel and is designed to be largely depleted by the end of the first cycle of operation. Control rods are used during the cycle to compensate for the remaining hot excess reactivity and reactivity changes due to burnup. Control rods may also be used to control the power distribution. The burnable absorber design is established such that the remaining hot excess reactivity is consistent with target control patterns during steady-state power generation. The hot excess reactivity and associated control rod patterns for the reference BWRX-300 equilibrium cycle are documented in the Nuclear Design Report for BWRX-300 Equilibrium Cycle. Examples of typical control rod patterns are presented in Figure 4.3-3.

4.3.4.2.1 Core Reactivity Effects

Control Reactivity

Neutron absorbing control rods are the primary means to control reactivity in transient and accident analyses. During events that result in relatively fast positive reactivity feedback, control rods are inserted rapidly using stored hydraulic energy. This is referred to as a scram. During events that result in relatively slow positive reactivity feedback and do not require a reactor “scram”, the FMCRDs that are operated using electric motors can be used for slower control rod insertion, (see Chapter 4, Subsection 4.5.1). Reactivity feedback mechanism is key for some Postulated Initiating Event groups (see Table 15.2-1) in which rods are withdrawn in error. Control reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response in TRACG.

Doppler Reactivity

Doppler reactivity is a reactivity feedback mechanism in BWR transient, and accident analyses associated with changes in fuel temperature. Doppler reactivity is negative with an increase in fuel temperature and becomes more important as the fuel temperature continues to increase. In Deterministic Safety Analysis (DSA) where reactivity feedback is important, doppler reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response in TRACG.

Void Reactivity

Void reactivity is an important reactivity feedback mechanism in BWR transient and accident analyses. The void reactivity feedback is always negative and is typically stronger (more negative void coefficient) at the end of an operating cycle. It is also stronger for reload cores versus an initial reactor core that includes only fresh fuel. This reactivity feedback mechanism is the dominant feedback for some Postulated Initiating Event groups. Table 15.2-1 focuses on events initiated from conditions of normal power operation. In the DSA, where it is important, void reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response analyzed using TRACG, the primary DSA computer code (see Subsection 15.5.1.2) that complies with CSA 286.7 on computer code quality assurance.

Moderator Temperature Reactivity

The moderator temperature coefficient of reactivity is defined as the change in reactivity produced by a unit change in moderator temperature. The value of this coefficient is important during the startup of a BWR. During power operation, the coefficient is not important, because the moderator is boiling, and primarily remains at the saturation temperature corresponding to the operating pressure. In addition, moderator density changes caused by boiling are much larger than changes from moderator temperature changes and therefore, mask any effects. Moderator temperature feedback is accounted for in the DSA analysis of startup conditions when coolant temperature is key to the event response, and there is no significant voiding in the core. Once core boiling/voiding begins, void reactivity feedback becomes dominant.

Boron Reactivity

For the BWRX-300, boron reactivity insertion (see Chapter 15, Section 15B), is only needed for long-term shutdown in very low probability events, where the hydraulic, and electric motors fail to insert a sufficient number of control rods.

Xenon Reactivity

Xenon reactivity feedback is not typically accounted for during events in DSA because the rate of change of reactivity is slow. The effects of xenon are accounted for in analyses of shutdown margin. Xenon reactivity impacts are also considered during core design and monitoring.

Also refer to Chapter 15, Subsection 15.2.4.1 for discussion regarding the effects of core reactivity on AOOs and DBAs.

4.3.4.3 Shutdown Margin

The minimum cold shutdown margin for the reference BWRX-300 equilibrium core is presented in Figure 4.3-9 and documented in the Nuclear Design Report, NEDC-33985P (Reference 4.3-1).

4.3.4.4 Thermal Limit

The margins to thermal limits for the reference BWRX-300 equilibrium core are presented in Figure 4.3-7 through Figure 4.3-9 and documented in the Nuclear Design Report (Reference 4.3-1).

4.3.4.5 Xenon Stability

BWRs are not susceptible to xenon oscillations. The xenon stability evaluation has been demonstrated by:

- No observed xenon instabilities in operating BWRs
- Special tests conducted on operating BWRs forcing the reactor into xenon instability demonstrate that xenon transients are highly damped by the large negative moderator void feedback
- Simulation calculations

All these indicators demonstrate that xenon transients are highly damped in a BWR due to the large negative moderator void feedback. Moreover, the BWRX-300 reactor core, fueled with GNF2, is evaluated to be less susceptible to xenon oscillations as compared to KKM (that never experienced an oscillation mode) as the void fraction is expected to be higher and the corresponding void reactivity coefficient more negative.

4.3.4.6 Thermal-Hydraulic Stability

The most limiting stability condition in the BWRX-300 normal operating region is at rated power/flow condition. The BWRX-300 core remains stable throughout the entire operating domain. Refer to Subsection 4.4.8 for a discussion of thermal-hydraulic stability.

4.3.4.7 Load Following Operation

Core power maneuvering is a normal occurrence for BWRs in response to several motivations (e.g., periodic surveillance testing, managing equipment performance degradation, control rod exchanges to reduce fuel duty, low electricity demand, etc.). The load follow operation involves a rapid power increase or reduction. The thermal limits established for the fuel are developed to assure conformance to regulatory requirements during all modes of operation, including load following. Additionally, BWR Operating Guidelines have been developed to assist in mitigating low frequency Stress Corrosion Cracking Pellet Clad Interaction and are encoded into the core monitoring function such that predictions may be performed in advance of power maneuvers.

The fuel rod thermal-mechanical operating limits described in Section 4.2 have been established to be applicable to all modes of operation, including frequent power cycles associated with load following. The principal performance evaluation affected by frequent fuel rod power cycling pertains to material fatigue. The GNF2 fuel rod thermal-mechanical suite of analyses includes a fatigue evaluation assuming frequent power cycling and is discussed in the GNF2 Fuel Rod Thermal Mechanical Design Report, (Reference 4.2.11). Note that fuel rod performance degradation associated with material fatigue has not been encountered in BWR operation.

The thermal-hydraulic limits established for every operating cycle (e.g., the OLMCPR) are developed to assure the Fuel Cladding Integrity Safety Limit described in Section 4.4 is maintained during all conditions of power operation above the Low Power Thermal Limit. The OLMCPR is developed including provision for AOOs that initiate when the core is operating at off-rated power & flow conditions and are applicable load following.

During periods of frequent core power maneuvering, transient xenon affects the core reactivity and power distribution. The effects of transient xenon are predicted by the core monitoring function prior to performing the core power maneuver and control rod adjustments are applied to maintain the core power at specified levels.

4.3.5 Changes from Previous Reactor Design

The BWRX-300 reactor core operates via natural circulation flow and is considered a composite design comprised of the most desirable features developed and applied to the BWR fleet. The key features are:

1. There are 240 fuel assemblies arranged identically to the KKM reactor core.
2. The lattice type is the N-lattice that originated with the ABWR. The N-lattice provides additional moderator volume in the intra-assembly bypass gap as compared to earlier lattice types.
3. The core flow results from natural circulation and the nominal bundle flow during power operation is lower than forced circulation reactors.
4. The core average power density is low compared to most forced circulation BWRs and approximately 20% lower than KKM (i.e., 870 MWth vs. 1097 MWth).
5. The CRDs are fine motion that were developed for the ABWR.
6. GTs are used in-lieu of the traversing in-core probe system for neutron instrument calibration.
7. The reference control rod type is the most modern commercially available control rod – the Ultra-HD.

While the exact configuration of the BWRX-300 reactor core is new, the configuration is similar to the BWR operating fleet, and the performance of all principal aspects have been proven in fleet application. Any differences from KKM are encompassed within the current approved nuclear methods.

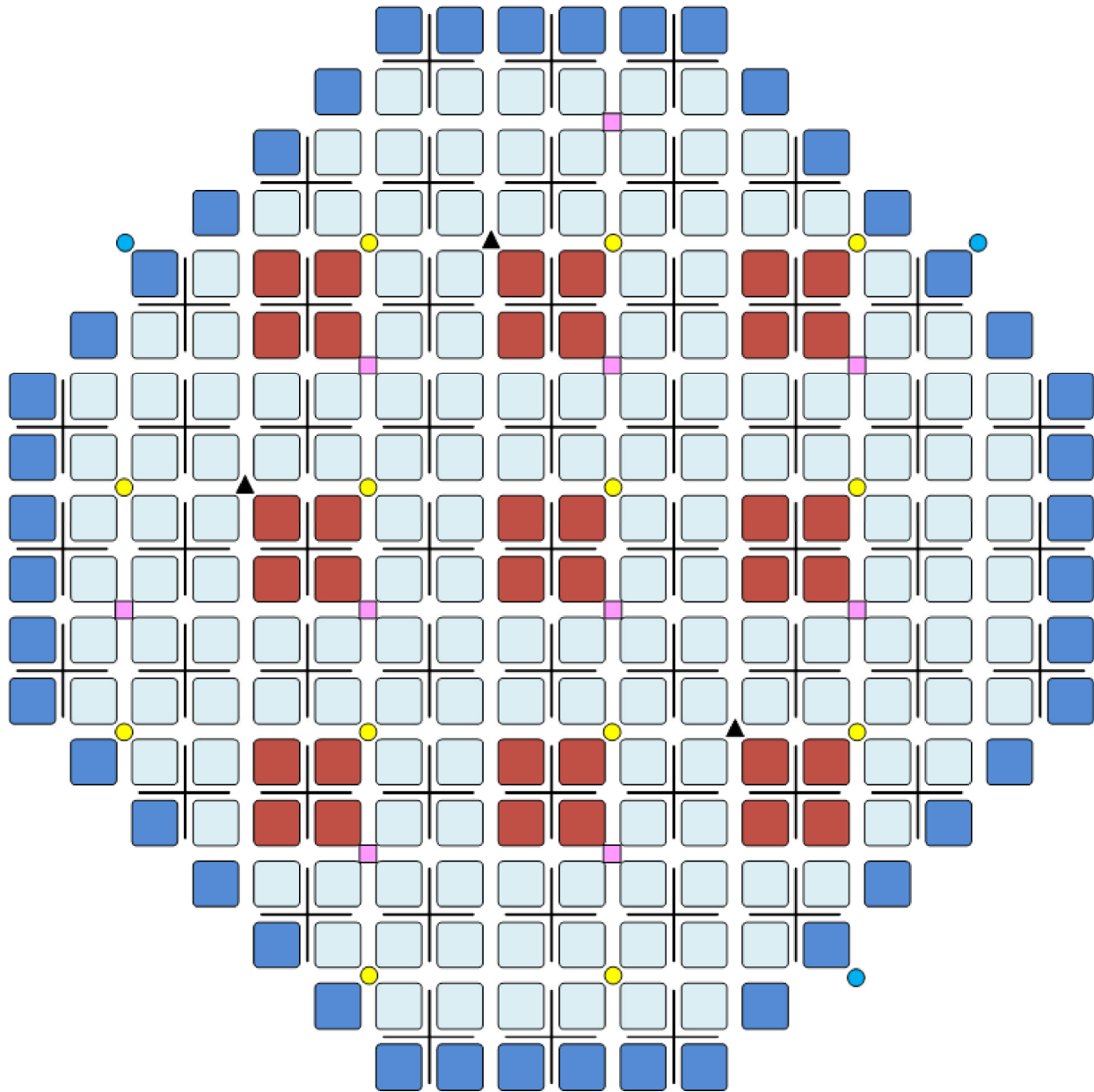
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**Table 4.3-1: Cycle Independent Safety Analyses and Operating Envelope Evaluations
Potentially Affected by Operating cycle Length**

Potentially Impacted Area	Evaluation Scope
Transient & stability analyses	Changes to plant & fuel specific, cycle-independent safety analyses
Core isotopic inventory and dose consequence	Changes to accident analyses' dose consequence and equipment qualification
Accident performance analysis	Changes to margin for fuel & reactor limits under accident conditions
Fluence impacts	Changes on the projected lifetime fluence for the RPV and reactor internals
Decay heat impacts	Changes to decay heat assumptions within fuel pool cooling, fire event analyses, and containment analyses
Technical specification surveillance intervals and instrument setpoints	Align to the outage schedule










- | | | | |
|---|---------------------------------|---|--------------------------------|
|  | - Interior Assembly Locations |  | - LPRM + GT Locations (13) |
|  | - Interior Control Cells (A2) |  | - WRNM Locations (10) |
|  | - Peripheral Assembly Locations |  | - Level Sensor Locations (3) |
| | |  | - Neutron Source Locations (3) |

Figure 4.3-1: Core Map and Preliminary Instrumentation Layout

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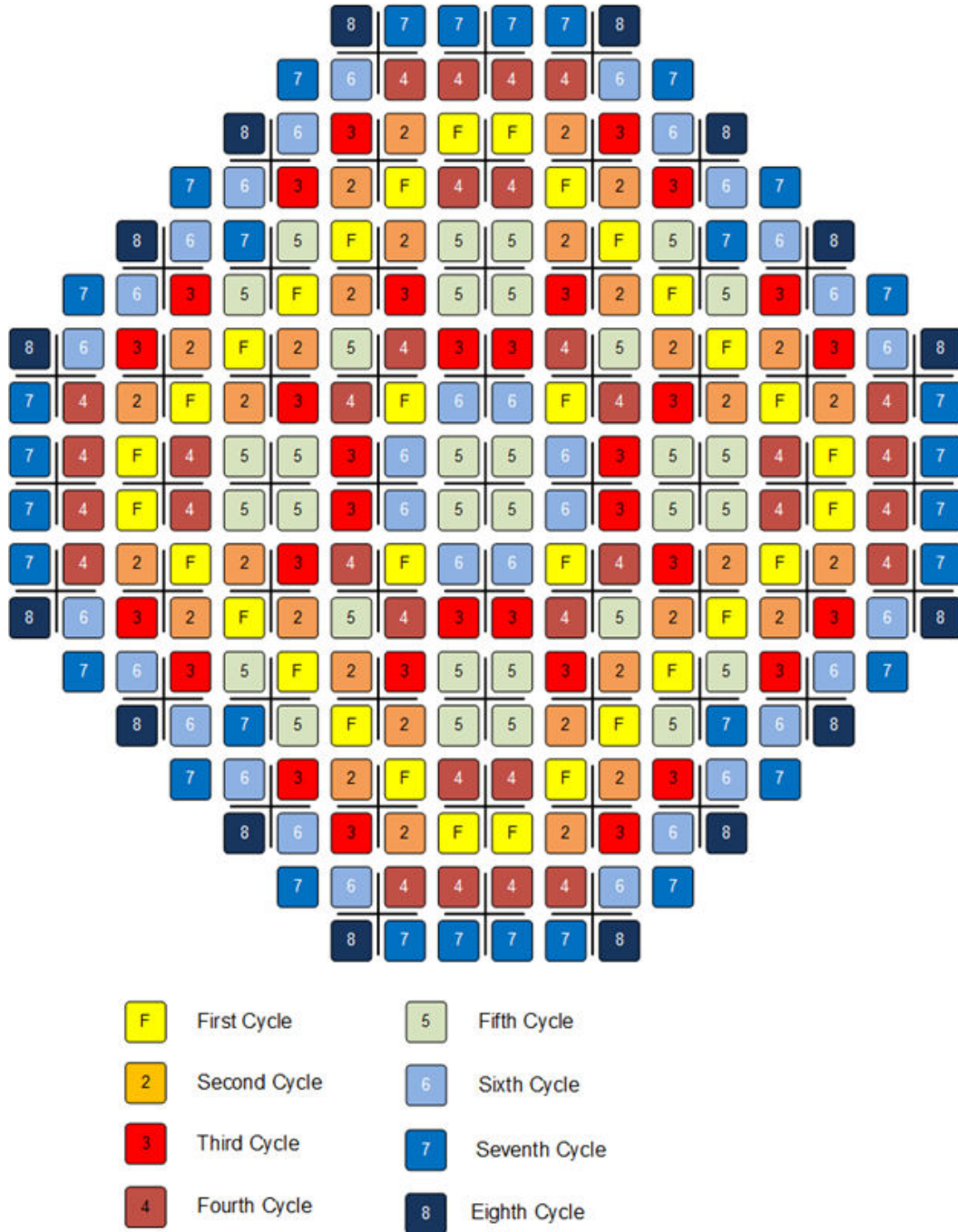


Figure 4.3-2: Core Loading Map – Reference Equilibrium Cycle

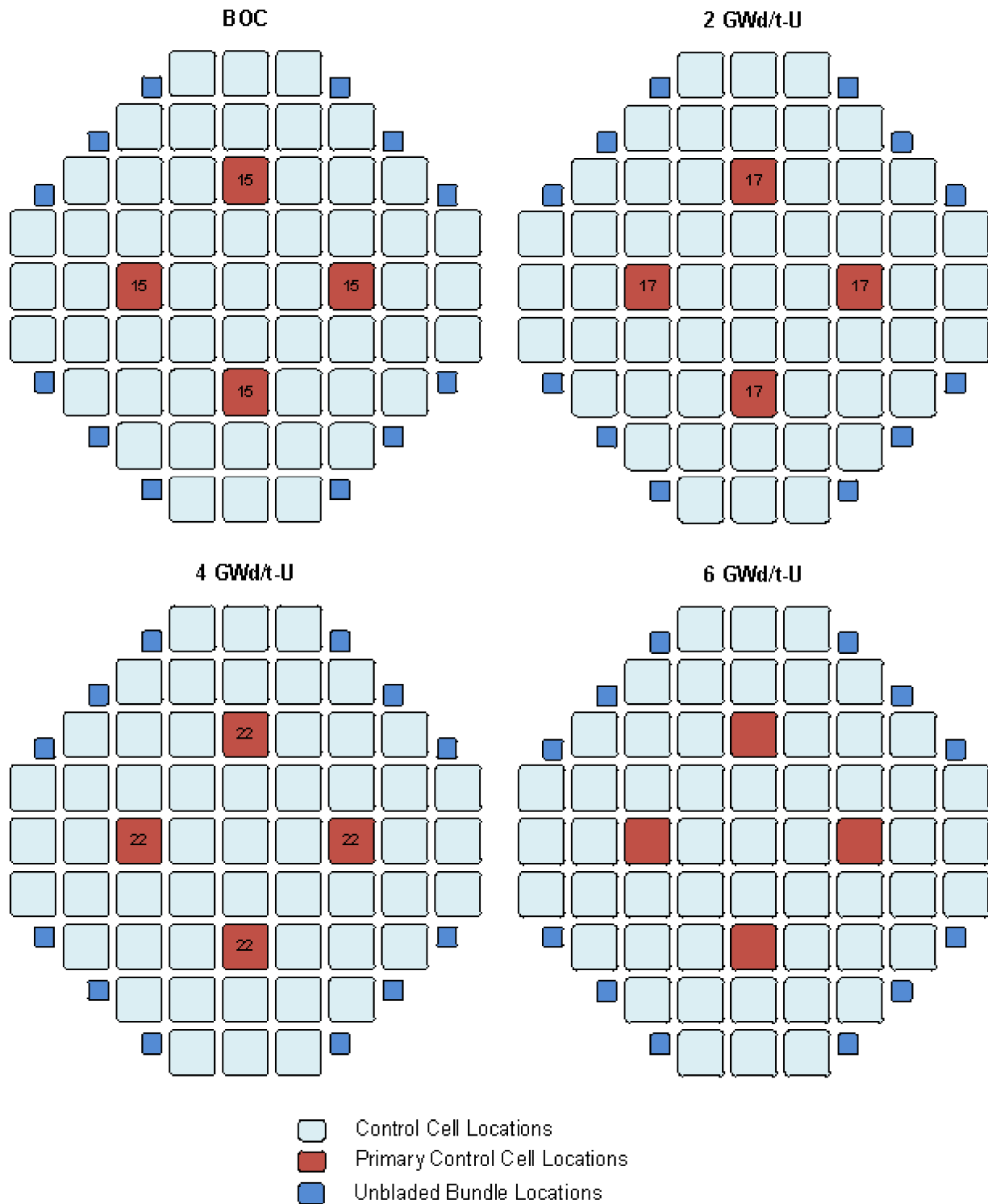
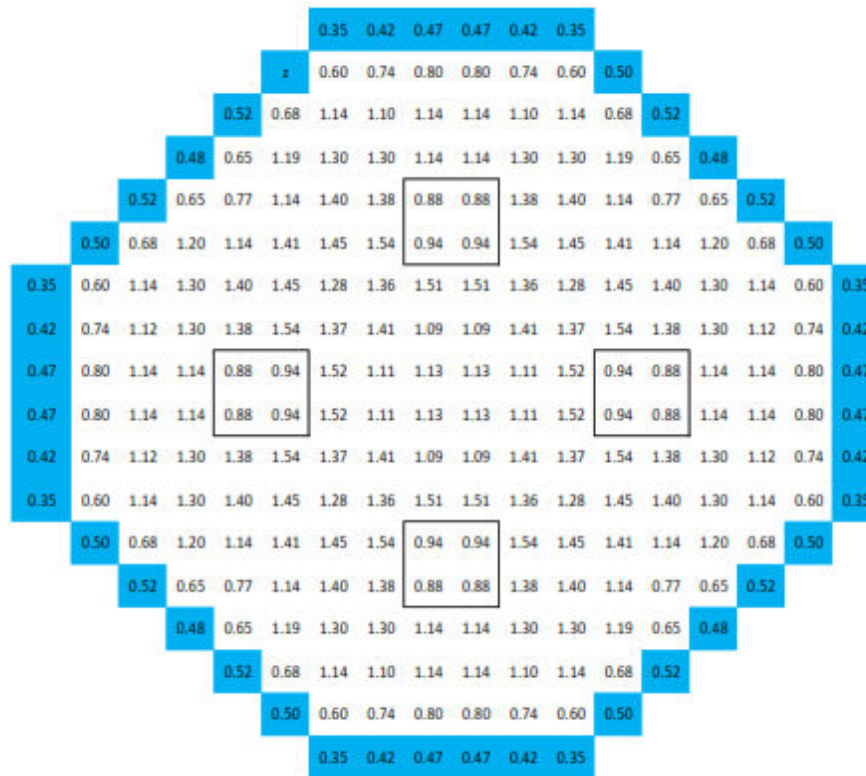


Figure 4.3-3: Representative Control Rod Patterns

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**Figure 4.3-4a: Peak to Average Bundle Power Ratio at Beginning of Cycle
(Average Bundle Power – 3.625 MW)**

						0.35	0.42	0.48	0.48	0.42	0.35			
					0.49	0.59	0.74	0.82	0.82	0.74	0.59	0.49		
				0.51	0.67	1.12	1.11	1.26	1.26	1.11	1.12	0.67	0.51	
			0.46	0.64	1.18	1.29	1.42	1.16	1.16	1.42	1.29	1.18	0.64	0.46
		0.51	0.64	0.77	1.13	1.51	1.36	0.89	0.89	1.36	1.51	1.13	0.77	0.64
	0.49	0.67	1.18	1.13	1.52	1.41	1.47	0.92	0.92	1.47	1.41	1.52	1.13	1.18
0.35	0.60	1.13	1.30	1.52	1.41	1.22	1.28	1.40	1.40	1.28	1.22	1.41	1.52	1.30
0.42	0.74	1.13	1.43	1.36	1.47	1.30	1.51	1.04	1.04	1.51	1.30	1.47	1.36	1.43
0.48	0.83	1.27	1.16	0.90	0.92	1.42	1.05	1.06	1.06	1.05	1.42	0.92	0.90	1.16
0.48	0.83	1.27	1.16	0.90	0.92	1.42	1.05	1.06	1.06	1.05	1.42	0.92	0.90	1.16
0.42	0.74	1.13	1.43	1.36	1.47	1.30	1.51	1.04	1.04	1.51	1.30	1.47	1.36	1.43
0.35	0.60	1.13	1.30	1.52	1.41	1.22	1.28	1.40	1.40	1.28	1.22	1.41	1.52	1.30
		0.49	0.67	1.18	1.13	1.52	1.41	1.47	0.92	0.92	1.47	1.41	1.52	1.13
		0.51	0.64	0.77	1.13	1.51	1.36	0.89	0.89	1.36	1.51	1.13	0.77	0.64
		0.46	0.64	1.18	1.29	1.42	1.16	1.16	1.42	1.29	1.18	0.64	0.47	
		0.51	0.67	1.13	1.11	1.26	1.26	1.11	1.13	0.67	0.51			
		0.49	0.59	0.74	0.82	0.82	0.74	0.60	0.49					
						0.35	0.42	0.48	0.48	0.42	0.35			

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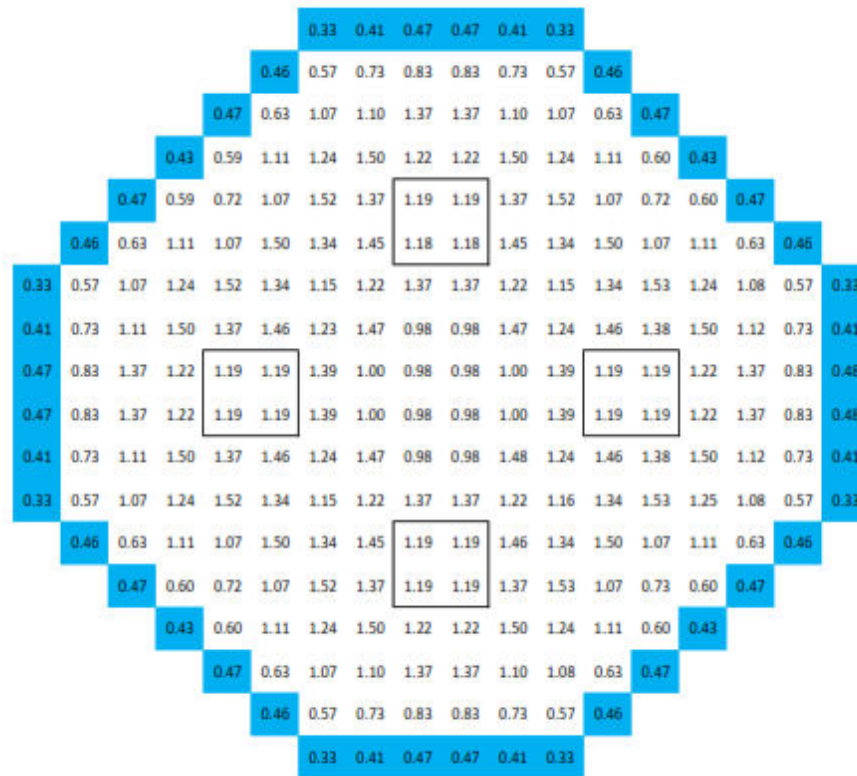


Figure 4.3-4c: Peak to Average Bundle Power Ratio at End of Cycle
(Average Bundle Power – 3.625 MW)

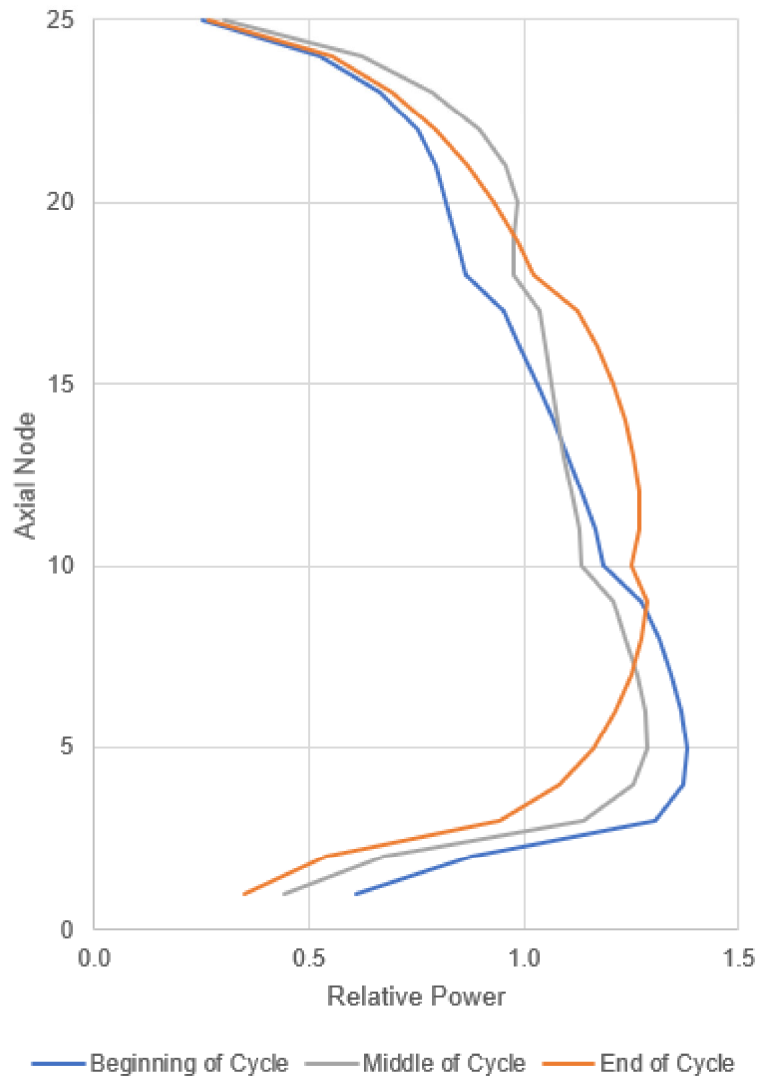


Figure 4.3-5: Axial Power Shapes

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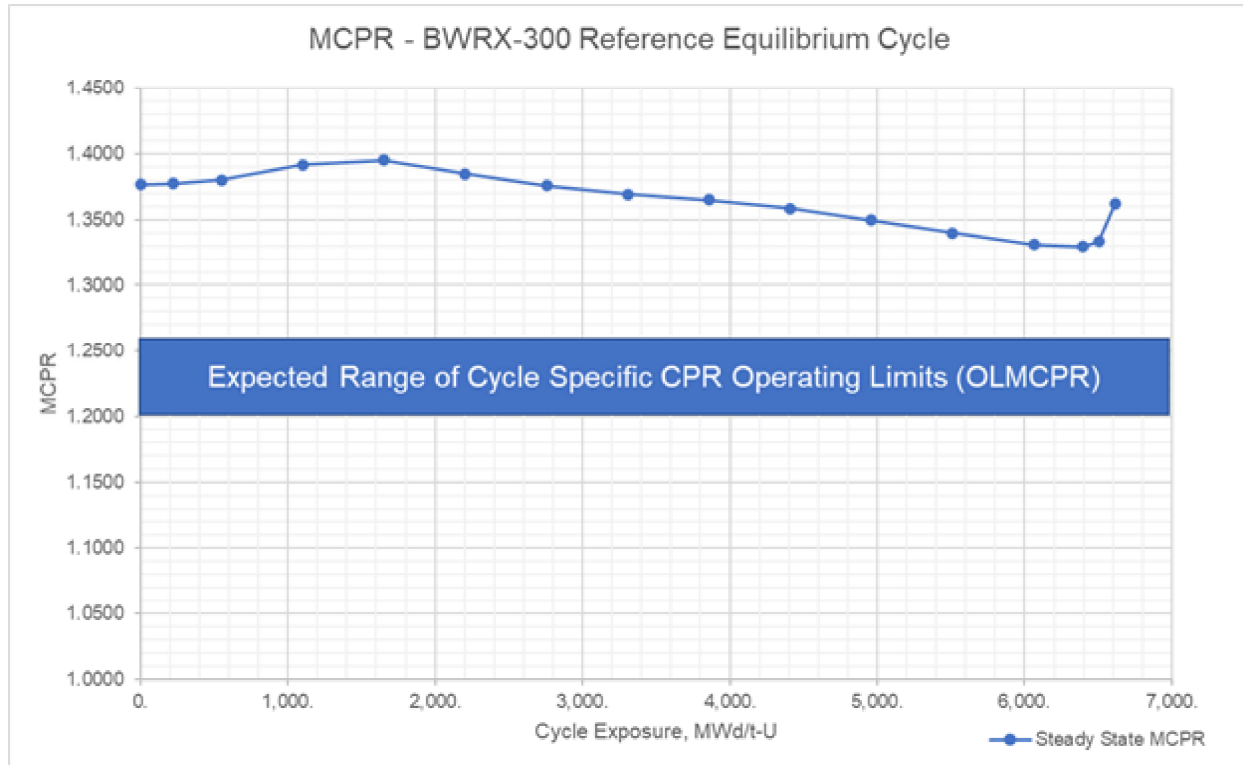


Figure 4.3-6: Reference Equilibrium Cycle Thermal Margin – Minimum Critical Power Ratio

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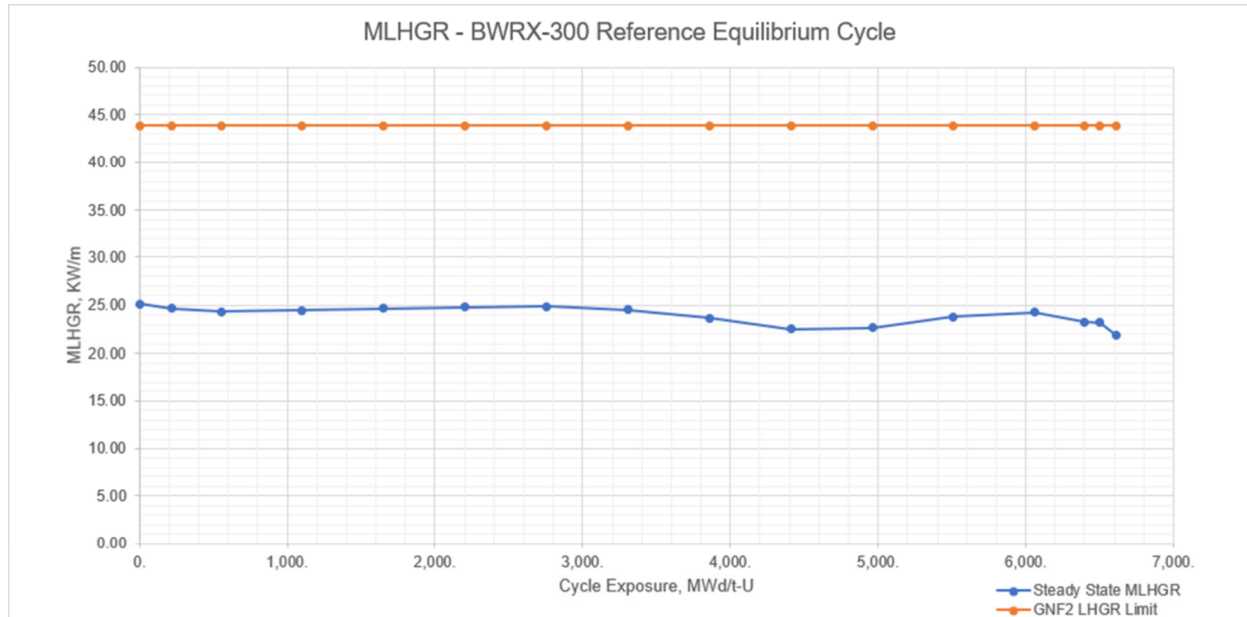


Figure 4.3-7: Reference Equilibrium Cycle Thermal Margin – Maximum Linear Heat Generation Rate

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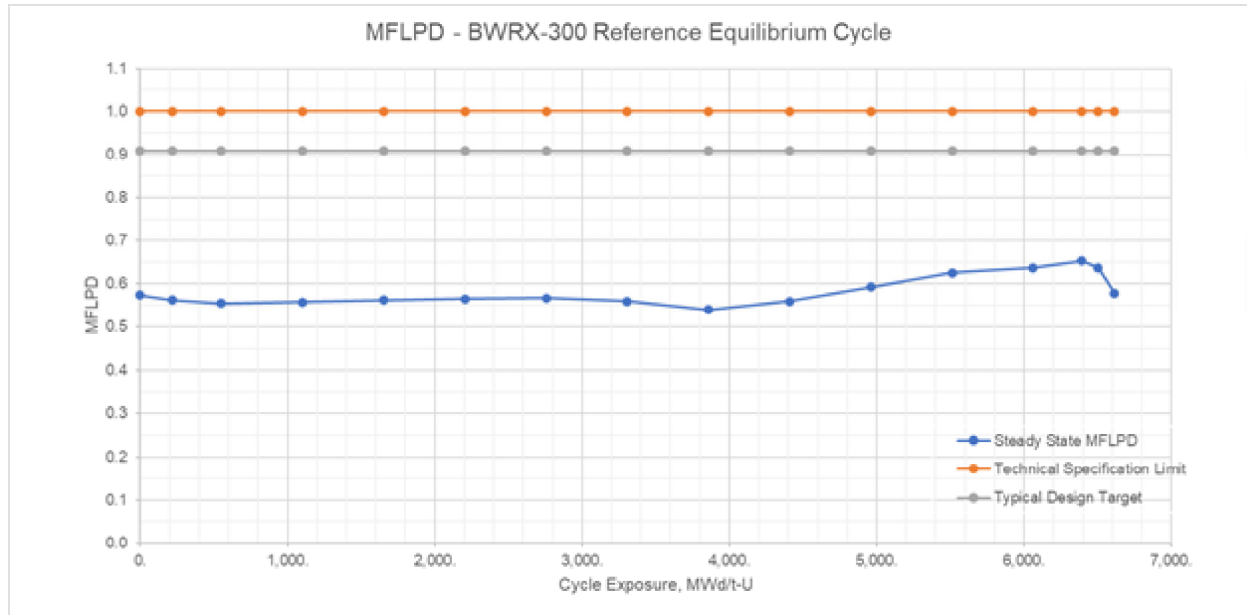


Figure 4.3-8: Reference Equilibrium Cycle Thermal Margin – Maximum Fraction Limiting Power Density

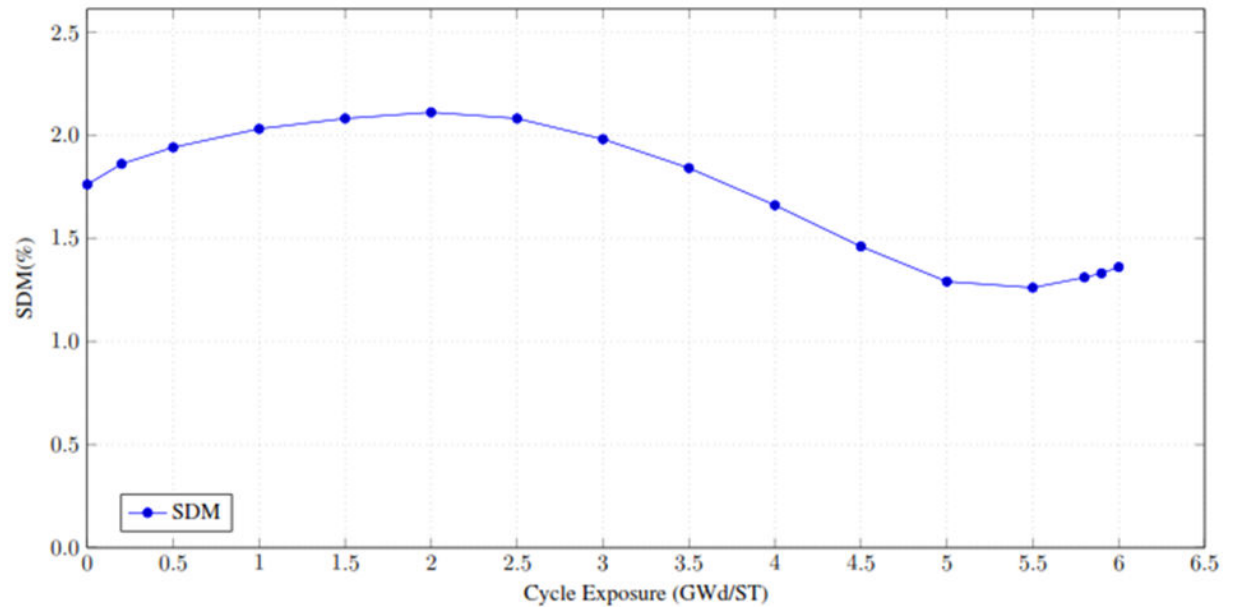


Figure 4.3-9: Shutdown Margin vs Cycle Exposure

4.4 Thermal and Hydraulic Design Basis

The thermal and hydraulic design of the reactor core provides adequate heat transfer from the fuel to the heat removal system, NBS, normal Shutdown Cooling System (SDC), or Isolation Condenser System during normal operation and AOOs.

4.4.1 Design Bases

Loss of fuel rod cladding integrity is not expected during normal reactor operation and AOOs. To satisfy this requirement, the following design bases have been established for the thermal and hydraulic design of the reactor core.

The core thermal-hydraulic design establishes thermal-hydraulic operating limits used in assuring safety margin in accordance with CNSC REGDOC-2.5.2, Version 1 (Reference 4.1-1). The core thermal-hydraulic stability performance addresses CNSC REGDOC-2.5.2 requirements in Subsection 4.4.8 Thermal-Hydraulic Stability.

Margin to the SAFDL is maintained during normal steady state operation when the MCPR is greater than the required OLMCPR and the MLHGR is maintained below the maximum LHGR limit(s). The steady state OLMCPR and Thermal Mechanical Operating Limit are established for the most limiting AOO and are analyzed in Chapter 15, Subsection 15.5.3 including uncertainties that provide reasonable assurance that no fuel damage results during AOOs.

4.4.1.1 Critical Power

The objective for normal operation and AOOs is maintaining nucleate boiling and precluding boiling transition.

The figure of merit confirming compliance with this objective is the CPR. The CPR is the ratio of the bundle power where at least one fuel rod point within the assembly experiences the onset of boiling transition to the operating bundle power. A calculated CPR of 1.0 corresponds to the best estimate value for the onset of boiling transition as determined by the product-specific GEXL correlation (e.g., GEXL17 for GNF2).

CPR limits are specified for maintaining adequate margin to the onset of the boiling transition. Adequate margin is defined to be a 95 % probability at a 95 % confidence level that no fuel rods are susceptible to boiling transition. These limits are calculated based on the three-step process defined in the sections that follow.

4.4.1.1.1 Fuel Cladding Integrity Safety Limit

The Fuel Cladding Integrity Safety Limit is calculated so that no significant fuel damage occurs during normal operation and AOOs on a cycle-independent basis. The Fuel Cladding Integrity Safety Limit is defined as the MCPR that ensures there is a 95% probability at a 95% confidence level that no fuel rods are susceptible to boiling transition. This limit is also referred to as Safety Limit MCPR or $MCPR_{95/95}$. The value is dependent only on the fuel design and establishes a lower limit for the cycle-specific $MCPR_{99.9\%}$ value.

4.4.1.1.2 $MCPR_{99.9\%}$

The $MCPR_{99.9\%}$ is determined on a cycle-specific basis to support the determination of the Operating Limit MCPR (OLMCPR). The $MCPR_{99.9\%}$ ensures that 99.9 % of the fuel rods in the core are not susceptible to boiling transition when considering the nuclear core design, plant system uncertainties, manufacturing uncertainties, and calculational uncertainties.

4.4.1.1.3 Operating Limit Minimum Critical Power Ratio

A cycle-specific OLMCPR provides adequate assurance that the $MCPR_{99.9\%}$ is not exceeded during normal operation and AOOs. By operating with the MCPR at or above the OLMCPR, the Fuel Cladding Integrity Safety Limit for that plant is not exceeded during normal operation and AOOs. This operating limit is obtained by combining the maximum Delta Critical Power Ratio Over Initial Critical Power Ratio ($\Delta CPR/ICPR$) (the change in CPR through the transient divided by the Initial CPR (ICPR)) value for the most limiting AOO and the $MCPR_{99.9\%}$. The significance of the $\Delta CPR/ICPR$ is that it measures the transient response. The maximum $\Delta CPR/ICPR$ from the AOOs is used in combination with the $MCPR_{99.9\%}$ to establish the OLMCPR on a cycle-specific basis.

4.4.1.2 Maximum LHGR

The Maximum LHGR (MLHGR) bases are described in Subsection 4.3.1. Thermal mechanical operating limits ensure margin to design limits for circumferential cladding strain and centerline fuel temperature. The adequacy of LHGR limits is evaluated for the most severe AOOs providing assurance that no fuel damage results during these postulated events.

4.4.1.3 Void Fraction Distribution

The void fraction in a BWR fuel bundle has a strong effect on the neutron flux and power (or fission rate) distribution. Accurate prediction of the void fraction is important for evaluating the performance of the BWR reactor and fuel. The void fraction is evaluated using correlations based on the characteristic dimensions of the fuel bundle and hydraulic properties of the two-phase flow in the fuel bundle.

4.4.1.4 Core Pressure Drop and Hydraulic Loads

An accurate model of core pressure drop is essential for modeling natural circulation flow, fuel and core inlet flow, and hydraulic loads for input to other evaluations.

4.4.1.5 Core Coolant Flow Distribution

Based on the prediction of core pressure drop, the distribution of flow into the fuel channels and the core bypass regions are calculated. The core coolant flow distribution forms the basis for predicting steady-state and transient MCPR and void fraction.

4.4.1.6 Fuel Heat Transfer

Engineering models in both steady state and transient analysis tools predict heat transfer between fuel pellet, cladding gap, cladding, fuel rod surface and the coolant in the evaluation of core and fuel safety criteria.

4.4.1.7 Summary of Design Bases

Steady-state operating limits ensure the design bases are satisfied during normal operation and the most limiting AOO. These limits are determined using analytical methodologies to evaluate core flow distributions, void fraction distributions, and fuel heat transfer rates as a function of both cycle exposure and core state.

The limiting AOO effects are confirmed maintaining compliance with the design acceptance criteria for the fuel, the RPV, and containment set forth in Chapter 15, Subsection 15.3.1. The barriers maintain their integrity and function.

4.4.2 Thermal and Hydraulic Methods

The analytical methods described in this section are qualified according to the provisions of American Society of Mechanical Engineers (ASME) NQA-1 which are equivalent to the provisions of CSA N286.7 (Reference 4.4-1). These methods have been approved by the USNRC (the country of origin) and are generally accepted for use by other countries with operating BWRs (e.g., Mexico, Spain, Switzerland).

4.4.2.1 Fuel Bundle Critical Power Methods

The critical power is the fuel bundle thermal power at the onset of boiling transition. Maintaining the bundle power below the critical power during steady-state operation and AOOs precludes the onset of boiling transition and satisfies the SAFDL pertaining to heat transfer from the fuel to the coolant (i.e., the Fuel Cladding Integrity Safety Limit). The methods applied in determining the bundle critical power and the associated operating limits are described below.

4.4.2.1.1 Fuel Bundle Critical Power Performance

The bundle critical power performance methodology was originally described in NEDO-10958-A (Reference 4.4-2). This original methodology evolved into the current form of the correlation, i.e., the GEXL correlation. The GEXL correlation is a critical quality and boiling length correlation used to predict the occurrence of boiling transition in BWR fuel. Each fuel bundle design has a specific set of correlation coefficients developed from full-scale test data. The specific GEXL correlation applied in analyzing GNF2 for all BWR types, including BWRX-300, is designated GEXL17 NEDC-33292 (Reference 4.4-3). The GEXL17 correlation application range established for fleetwide application envelopes the hydraulic conditions that the BWRX-300 experiences during normal operation and AOOs.

4.4.2.1.2 Fuel Cladding Integrity Safety Limit Method

The fuel cladding integrity safety limit, named the $MCPR_{95/95}$, ensures there is a 95% probability at a 95% confidence level that no fuel rods are susceptible to boiling transition using a limit that is derived from comparing the predicted critical power to the experimental data for a specific fuel bundle design. The Experimental CPR is defined as the ratio of the calculated critical power as determined by the GEXL correlation to the experimental critical power.

Each experimental data point has a predicted value and associated Experimental CPR. The Experimental CPR is evaluated for all the points in the dataset resulting in a probability distribution. The Experimental CPR probability distribution serves as the basis for the correlation uncertainty. Thus, for a given critical power correlation, a limit that bounds 95% of a correlation's Experimental CPR distribution at a 95% confidence level is determined and set as the $MCPR_{95/95}$.

The determination of the $MCPR_{95/95}$ is further described in TSTF-564 (Reference 4.4-9).

4.4.2.1.3 $MCPR_{99.9\%}$

The cycle specific $MCPR_{99.9\%}$ limit is established as described in NEDC-33939, Appendix C (Reference 4.3-2).

4.4.2.1.4 OLMCPR Calculation Method

BWRX-300 AOOs are analyzed using the TRACG model described in NEDE-32176P (Reference 4.3-3). The application of TRACG to AOO analyses is described in TRACG Application for BWRX-300, NEDO-32082 (Reference 4.3-4) along with the process to calculate the $\Delta CPR/ICPR$.

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This OLMCPR is obtained by combining the maximum $\Delta\text{CPR}/\text{ICPR}$ value for the most limiting AOO and the $\text{MCPR}_{99.9\%}$. The OLMCPR is specified in one of two ways. It can be conservatively defined as the most limiting combination of $\Delta\text{CPR}/\text{ICPR}$ and $\text{MCPR}_{99.9\%}$ over an entire cycle, or it is defined on a cycle exposure-dependent basis using combinations of cycle exposure-dependent values of $\Delta\text{CPR}/\text{ICPR}$ and/or $\text{MCPR}_{99.9\%}$. Chapter 15, Section 15.5 provides the analysis of limiting AOOs.

4.4.2.2 Maximum Linear Heat Generation Rate Method

The MLHGR methods are described in Subsection 4.3.1. Margin to design limits for circumferential cladding strain and centerline fuel temperature is evaluated for AOOs in accordance with the TRACG Application for BWRX-300 (Reference 4.3-4).

4.4.2.3 Void Fraction Distribution Methods

The empirical correlations used for the calculating void fraction are the GEH void fraction correlation used in the 3D core simulator, steady-state thermal hydraulic calculations, and the correlations for the interfacial shear used in TRACG. The TRACG void fraction model is described in NEDE-32176P (Reference 4.3-3). The core simulator model is described in NEDC-33939P (Reference 4.3.2).

4.4.2.4 Core Pressure Drop and Hydraulic Loads Methods

The total bundle pressure drop is defined as the sum of four components: friction, elevation, acceleration, and local losses. In these models, the bundle is also divided into control volumes where the four components of total pressure drop are evaluated separately. This allows capturing the effects on pressure drop of axially variable geometry parameters such as flow area, hydraulic diameter, wetted/heated perimeters, heat flux, and spacer elevations. The hydraulic diameter is defined as four times the axial flow area divided by the wetted perimeter, at any axial location and includes the fuel rod, channel inner wall, and water rod perimeters. The geometry of heated surfaces consists of the number of fuel rods and the fuel rod diameter in a fuel assembly. For fuel assembly types with partial length rods, the number of partial length rods and the associated length(s) are also accounted for in defining fuel assembly hydraulic diameter.

The TRACG methods for core pressure drop modeling are described in NEDE-32176P (Reference 4.3-3). The TRACG hydraulic formulation for core pressure drop is identical to the model used in the core design analysis except for the acceleration pressure drop component. The models used in the core design analysis are described in NEDE-333939 (Reference 4.3-2). The fuel design specific loss coefficients and assembly pressure drop models are developed from and confirmed by data from full scale testing of prototypical assemblies spanning the range of hydraulic conditions where hydraulic models are applied. The adequacy of the pressure drop model applied to GNF2 fuel is summarized in GNF2 pressure drop characteristics and is inclusive of the BWRX-300 operating conditions in GNF2 Pressure Drop Calculations, NEDO-32082 (Reference 4.4-4).

Hydraulic loads are determined based on the reactor internal pressure differences. The TRACG computer code is used to analyze the transient conditions within the reactor vessel following AOOs.

4.4.2.5 Core Coolant Flow Distribution Methods

The core coolant flow distribution methods used in TRACG are described in NEDE-32176P, Chapters 6 and 7 (Reference 4.3-3). TRACG treats all fuel channels as one-dimensional (axial) components, but the vessel is modeled as a three-dimensional component. Hence, the pressure drop across two planes in the vessel is the same at all radial and azimuthal locations if the geometry of the components in the vicinity of these planes has radial and azimuthal symmetry. Otherwise, this pressure differential displays some (locally) radial and azimuthal non-uniformity.

The flow distribution to the fuel assemblies and bypass flow paths in the core simulator model is calculated assuming the pressure drop across all fuel assemblies and bypass flow paths is the same. The bundle pressure drop evaluation includes frictional, local, elevation, and acceleration losses described above. The core inlet flow is an input to the core simulator. The value used in core design analysis is determined based on the TRACG prediction of the natural circulation core inlet flow. In operation, the core monitoring function determines core inlet flow based on plant instrumentation discussed in Chapter 7, Subsection 7.3.3.2.

The bypass flow methodology is described in NEDE-32176P (Reference 4.3-3), Subsection 7.5.1. The same methodology is used in the core simulator model.

4.4.2.6 Fuel Heat Transfer Methods

The Jens-Lottes heat transfer correlation is used to determine the cladding-to-coolant heat transfer coefficients for nucleate boiling as described in (Reference 4.4-5). For the single-phase convection or liquid region, the Dittus-Boelter correlation is used. The methodology for fuel cladding, gap and pellet heat transfer is described in Subsection 4.4.1.1.4 and (References 4.2-5 through 4.2-8).

4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Core

Several metrics of interest for the BWRX-300 are compared against a typical BWR/6 plant and the ABWR in Table 4.4-1.

4.4.3.1 Reactor Coolant System

The Reactor Coolant System is described in Chapter 5. The BWRX-300 reactor coolant system is shown in Figure 4.1-1. The BWRX-300 thermal hydraulic design is similar to operating BWRs except that it does not require recirculation pumps or associated coolant piping. Circulation of the reactor coolant through the BWRX-300 core is accomplished via natural circulation. Natural circulation is enabled mainly by the addition of a tall chimney between the top of the core at the top guide plate of the core to the bottom of the chimney head and steam separator assembly. The natural circulation flow rate depends on the difference in water density between the regions. The core flow varies according to the power level because fluid density changes with the power level. Therefore, a core power-flow map reduces to a single line and there is no active control of the core flow at any given power level, as shown in Figure 4.4-2.

4.4.3.2 Core Hydraulics

Accurate prediction of bundle flow and power distributions is important in the calculation of margin to the thermal limits of each fuel bundle. Pressure drop characteristics are included in plant cycle specific analyses for the calculation of the Operating Limit MCPR.

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Because of the channeled configuration of BWR fuel assemblies, there is no assembly-to-assembly cross flow inside the core. The only issue of hydraulic compatibility of various bundle types in a core is the bundle inlet flow rate variation and its impact on margin to thermal limits (i.e., MCPR or MLHGR). The coupled thermal-hydraulic-nuclear analyses are performed each cycle to determine fuel bundle flow and power distribution. The analyses use the various bundle pressure loss coefficients to determine the flow distribution required to maintain total core pressure drop boundary conditions applied to all fuel bundles. The margin to thermal limits of each fuel bundle is determined using this consistent set of calculated bundle flow and power.

The flow distribution to the fuel assemblies and bypass flow paths is calculated using various pressure drop models that include friction loss coefficients, local loss coefficients, two-phase multipliers, and void-quality correlations. These models are developed from pressure drop data with a best-fit basis. Pressure drop measurements made in operating reactors, confirm that the total measured and calculated core pressure drops agree. This information is collected normally as part of core management activities and its purpose is to identify anomalous behavior. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor.

An iteration is performed on flow through each flow path (fuel assemblies and bypass flow paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows to the total core flow. The total core flow less the control rod cooling flow enters the lower plenum. A fraction of this passes through various bypass flow paths. The remainder passes through the orifice in the fuel support casting and/or peripheral fuel support (experiencing a pressure loss) where some of the flow exits through the fit-up between the fuel support and lower tie plate and through the lower tie plate holes into the bypass flow region. All fuel bundles have lower tie plate holes. Most of the flow continues through the lower tie plate (experiencing a pressure loss) where some flow exits through the flow path defined by the fuel channel and lower tie plate into the bypass region.

The unique GNF2 fuel assembly hydraulic characteristics include the inlet orifice, lower tie plate, upper tie plate, spacers, water rod, and various leakage paths. The hydraulic characteristics of these components flow paths have been developed and confirmed by test comparisons. These unique GNF2 hydraulic characteristics are used in all analysis models and methods where the fuel assembly hydraulics are needed.

The analytical methods used in the analysis of the BWRX-300 reactor are Global Nuclear Fuels Americas LLC (GNF A) standard codes in use throughout the industry and licenced in other jurisdictions.

Flow pressure drop characteristics shall be included in plant cycle specific analyses for the calculation of the Operating Limit MCPR

4.4.3.3 Core Response in Transient and Accident Analysis

Core and fuel response to postulated AOOs and accident events is explicitly modeled via the TRACG computer code employing three-dimensional kinetics that is consistent with the BWR Core Simulator PANAC11. A description of the core response during off-normal condition is provided in Chapter 15, Subsection 15.2.4 The results of AOO and DBA analyses are presented in Chapter 15, Subsections 15.5.3 and 15.5.4.

4.4.3.4 Physics and Reactivity Coefficients Including Effects or Power Coefficient of Reactivity

The principal reactivity coefficient for the BWRX-300, and any BWR, is the void coefficient, which is the dominant constituent of the power coefficient of reactivity and required to be negative under all reactor states.

Power control to ensure compliance with LHGR limits, including aspects of loss of reactivity control, is based on the use of cruciform shaped control rods (which are sometimes referred to as blades), like other BWR types, for reactor thermal power, power distribution, and control of the BWRX-300 core. Fast acting shutdown capability is provided using stored hydraulic energy HCUs as a diverse motive force relative to the FMCRDs, as described in Section 4.6. The Core Monitoring function (Chapter 7, Section 7.3) in concert with reactor operators ensure compliance with requirements implemented by the thermal limits described in Subsections 4.3.1 (MLHGR) and 4.4 (MCPR) during power operation.

4.4.3.5 Operation Limits and Conditions for Core, Core Instrumentation and Control, and Nuclear Fuel

The core design strategy for various refueling cycle intervals is based on management of thermal limit margins for fuel design.

Plant OLMCPR is established by considering the limiting AOOs for each operating cycle. This may be calculated as a function of exposure. Reports are normally prepared for reactor operators in implementing the Cycle Management Report (CMR) requirements for each refueling cycle interval.

For each new fuel design, the applicability of generic or country-specific MCPR analyses are confirmed for each operating cycle, or a plant-specific analysis is performed.

4.4.3.6 Load Following Characteristics

The thermal-hydraulic limits established for every operating cycle (e.g., the OLMCPR) are developed to assure the Fuel Cladding Integrity Safety Limit described in Section 4.4 is maintained during all conditions of power operation above the Low Power Thermal Limit. The OLMCPR is developed including provision for AOOs that initiate when the core is operating at off-rated power & flow conditions and are applicable load following.

The BWRX-300 is a natural circulation BWR. The power-flow map depicted in Figure 4.4-2 illustrates that the core flow is fairly constant in the normal range of core thermal powers associated with load following. As the reactor power is the principal parameter that varies, and reactor coolant flow is fairly constant, compliance to thermal limits is readily maintained during periods of off-rated power operation. Further discussion of load following is provided in Subsection 4.3.4.7.

4.4.4 Thermal and Hydraulic Evaluation

4.4.4.1 Critical Power Evaluations

Compliance to representative steady-state MCPR operating limits is demonstrated for a typical simulation of an equilibrium cycle described in Section 4.3, Core Nuclear Design. The typical OLMCPR evaluation process is outlined in the sections that follow.

4.4.4.1.1 Fuel Cladding Integrity Safety Limit Evaluation

The GNF2 Fuel Cladding Integrity Safety Limit value is 1.07 as per TSTF-564-A (Reference 4.4-9).

4.4.4.1.2 *MCPR_{99.9%}*

The MCPR_{99.9} is evaluated for a specific core design with uncertainties documented in (Reference 4.3-2), NEDC-33939. The MCPR_{99.9} limit is computed on a cycle-specific basis and reported in a cycle specific Core Operating Limits Report (COLR).

4.4.4.1.3 *Minimum Critical Power Ratio Operating Limit Evaluation*

The MCPR Operating Limit is computed on a cycle-specific basis and reported in a cycle specific COLR. The expected range of cycle-specific OLMCPRs for the reference BWRX-300 equilibrium cycle is depicted in Figure 4.3-6.

4.4.4.2 Maximum Linear Heat Generation Rate Evaluations

Compliance to steady-state MLHGR limits is demonstrated for the reference equilibrium cycle in Section 4.3 and the Nuclear Design Report (Reference 4.3-1). The AOO analysis for the reference equilibrium cycle is documented in Chapter 15, Subsection 15.5.3. Compliance to design limits for circumferential cladding strain and centerline fuel temperature during AOO events are confirmed on a cycle-specific basis and associated limits are reported in the COLR.

4.4.4.3 Void Fraction Distribution Evaluation

The void fraction distribution is dependent upon the reactor state and varies throughout an operating cycle. The calculation of the void fraction distribution is integral to the TRACG methodology and the 3D core simulator, PANAC11. Representative values for the core average axial void fraction are depicted in Figure 4.4-4.

4.4.4.4 Core Pressure Drop and Hydraulic Loads Evaluations

The expected operating pressure for the BWRX-300 is within the qualification basis of the pressure drop methods. The MCPR_{99.9%} calculation method also assumes pressure drop uncertainty.

4.4.4.5 Core Coolant Flow Distribution Evaluation

The core coolant flow distribution (i.e., the inlet flow to each fuel assembly) is determined by the coupled nuclear and thermal hydraulic steady-state methods in NEDC-33939 (Reference 4.3-2) and in the design and analysis of the reference core nuclear design summarized in Section 4.3.

4.4.4.6 Fuel Heat Transfer Evaluations

Fuel heat transfer evaluations are dependent upon the reactor state. The calculation of fuel heat transfer is integral to the TRACG methodology and the 3D core simulator, PANAC11.

4.4.5 Evaluation of the Validity of Thermal and Hydraulic Design Techniques

The thermal and hydraulic design technique comprises qualified analytical methods employed in developing a self-consistent set of design outcomes that conform to design bases. The TRACG method described in Subsection 4.4.2 demonstrates accurate model system performance in BWRs. The thermal hydraulic design bases evaluated for the BWRX-300 described in Subsection 4.4.1 are applicable and adequate to those established for the operating fleet. The BWRX-300 reactor core was developed from applying qualified analytical methods and the results demonstrate compliance to the design bases that are used in BWRs.

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4.4.6 References

- 4.4-1 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 4.4-2 NEDO-10958-A SH 0001,"General Electric BWT Thermal Analysis Basis (GETAB) Data, Correlation and Design application, Licensing Report," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-3 NEDC-33292P, "GEXL17 Correlation for GNF2 Fuel," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-4 NEDC-33976P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Fuel Assembly Pressure Drop Characteristics," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-5 ANL-4627, "Analysis of Heat Transfer, Burnout, Pressure Drop and Density Data for High-Pressure Water," Argonne National Laboratory.
- 4.4-6 NEDC-33083, "TRACG Application for ESBWR," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-7 NEDE-32177, "TRACG Qualification," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-8 NEDE-24011, "NEDE-24011-P-A-31, General Electric Standard Application for Reactor Fuel (GESTAR II)," GE-Hitachi Nuclear Energy Americas, LLC.
- 4.4-9 TSTF-564-A, "Safety Limit MCPR," Technical Specifications Task Force.

Table 4.4-1: Typical Thermal–Hydraulic Design Characteristics of the Reactor Core

General Operating Conditions	BWR/6	ABWR	BWRX-300
Reference design thermal output (MWt) ⁽¹⁾	3579	3926	870
Power level for engineered safety features (MWt)	3730	4005	887
Steam flow rate (kg/s)	1940	2122	503
Core coolant flow rate (kg/s)	13104	14502	1890
Feedwater flow rate (kg/s)	1936	2118	507
System pressure, nominal in steam dome (kPa)	7171	7171	7171
Coolant saturation temperature at core design pressure (°C)	288	288	288
Average power density (kW/L)	54.1	50.6	39.6
Core inlet enthalpy (kJ/kg)	1227	1227	1218
Core inlet temperature (°C)	278	278	269-272
Core maximum exit voids within assemblies (%)	79.0	75.1	90.0
Core average void fraction, active coolant	0.414	0.408	0.530
Total core pressure drop (kPa)	182.0	168.2	71.98
Core support plate pressure drop (kPa)	151.7	137.9	49.6

(1) 1MW = 1E6 J/sec

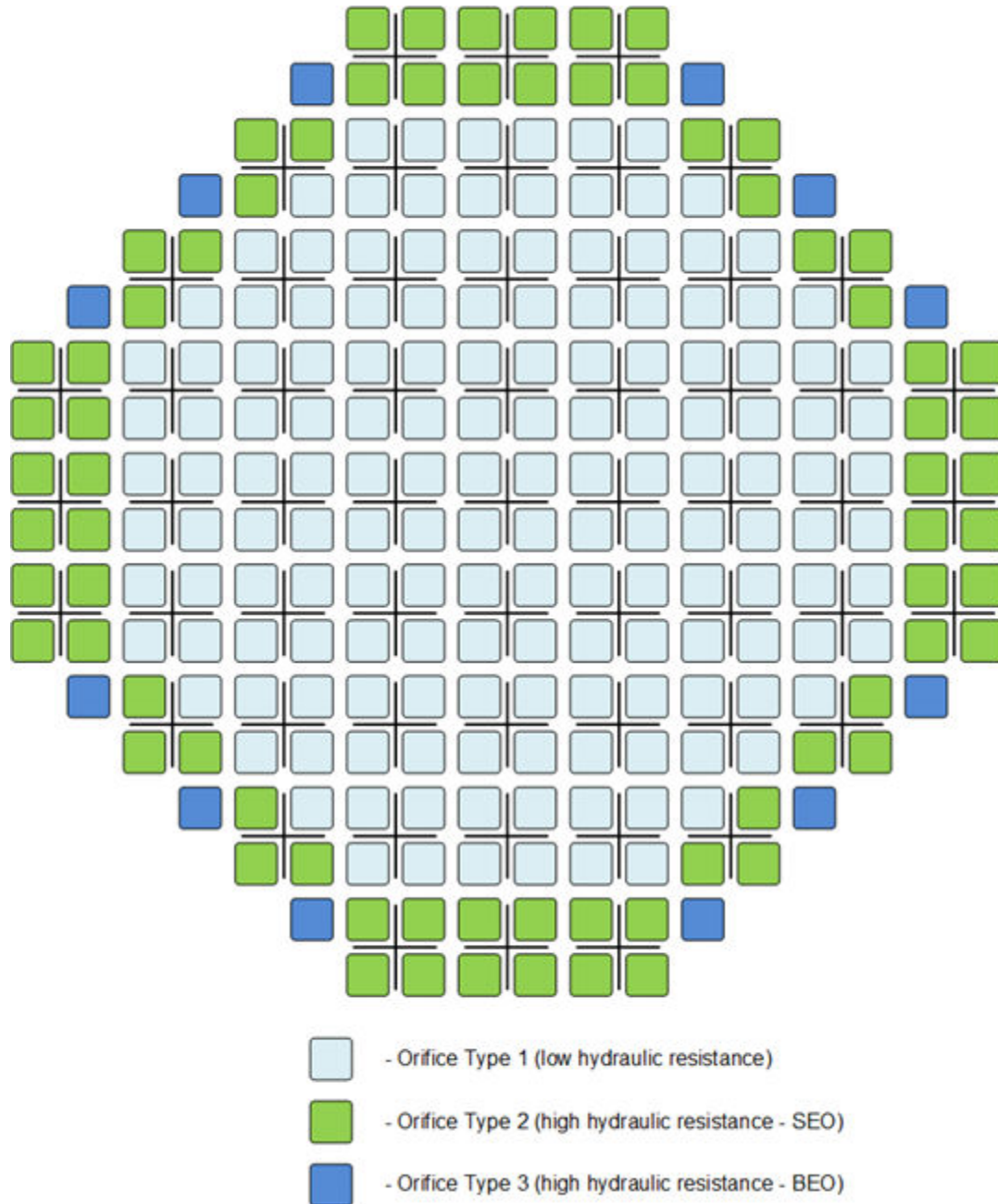


Figure 4.4-1: BWRX-300 Core Inlet Orifice Type Arrangement

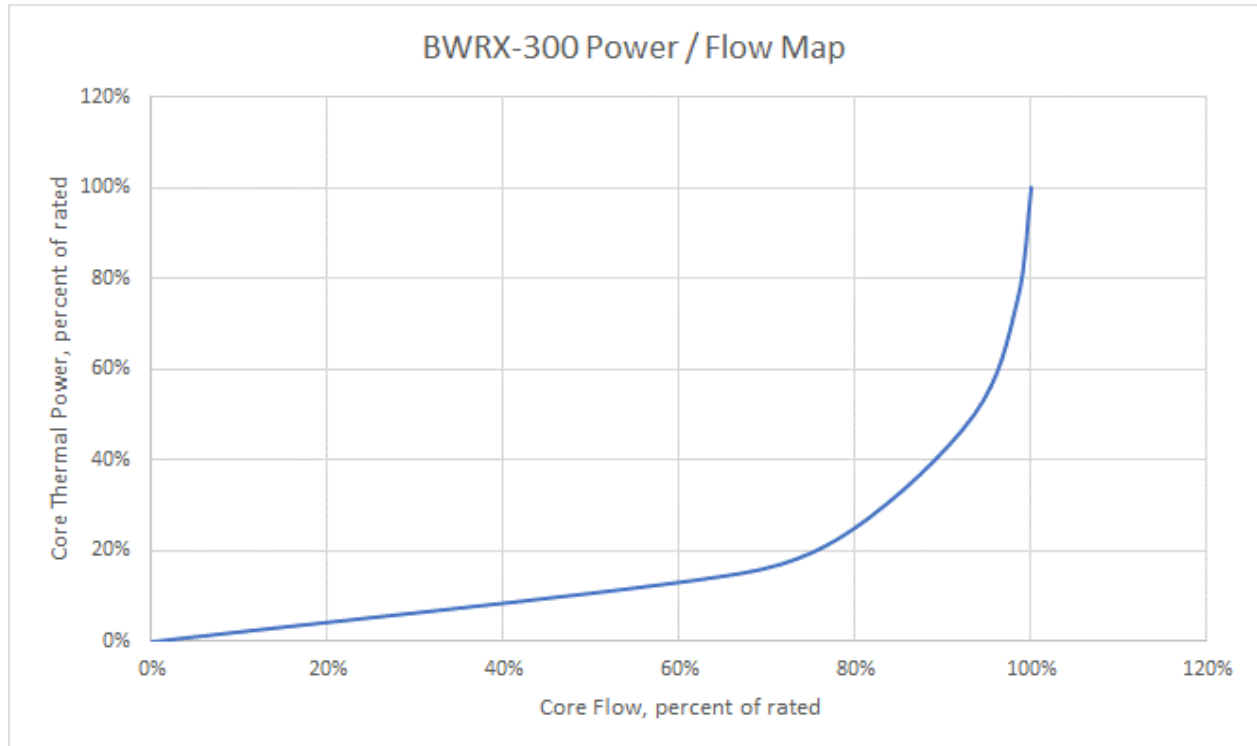


Figure 4.4-2: BWRX-300 Power / Flow Map

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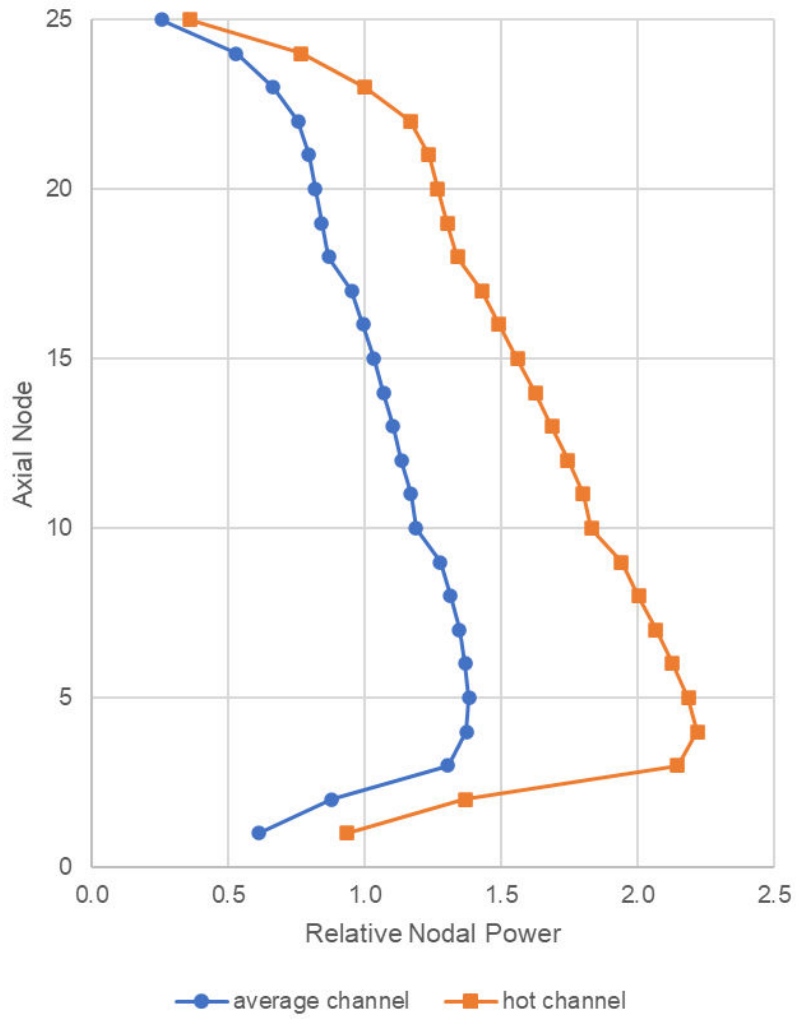


Figure 4.4-3: Relative Power for Analyzed Node (hot channel and average channel)

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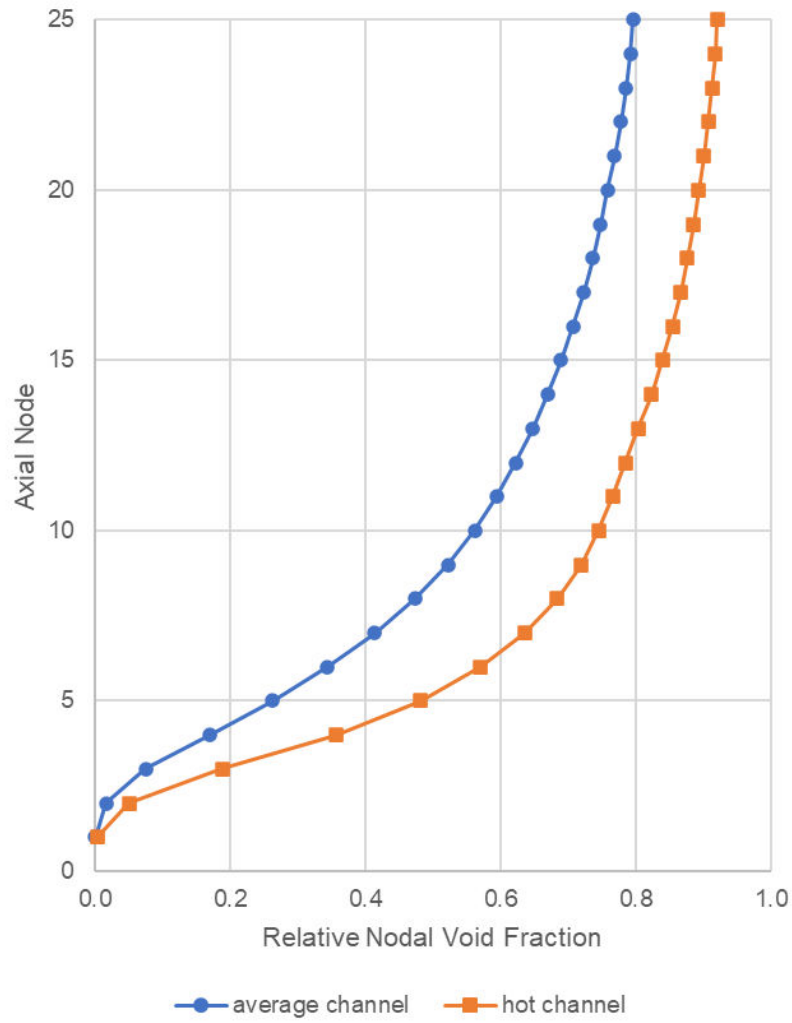


Figure 4.4-4: Relative Void Fraction for Analyzed Node (hot channel and average channel)

4.5 Reactor Internals Material

The materials used in the reactor internals are described in Chapter 5, Subsection 5.2.5, and Table 5.2-3.

The materials comprising the fuel and control rod assemblies are described in 4.2, and associated references.

The reactor coolant pressure boundary components of the FMCRD, consists of the lower component housing and the upper component's middle flange. These components are made with 300 series stainless steel materials compatible with the reactor coolant in accordance with ASME Code, Section III. The installation bolts used to attach the middle flange and lower component to the CRD housing are low alloy steel material.

4.5.1 References

None.

4.6 Design of Reactivity Control Systems

Reactivity Controls consist of:

- Control rods (Section 4.2) and Control Rod Drive Systems
- Supplementary reactivity control in the form of gadolinia-urania fuel rods (Section 4.2.2)

This section describes the CRD system design, including the design bases, functions, system configuration & operation.

4.6.1 Control Rod Drive Design Bases

The CRD system provides the primary means of reactivity control during normal, abnormal and accident conditions. The system design basis includes two diverse motive forces for the CRD insertion (scram) using high pressure water from the HCU's, and control rod insertion using the FMCRD motor. Incorporated into the design are positioning and protective features that prevent inadvertent withdrawal, drop, and ejection of the control rod due to a component break or other malfunction. Complete system functionality is described below in Subsection 4.6.2.

4.6.2 Description

The CRD system includes three major elements:

- Electro-hydraulic FMCRD mechanisms
- HCU
- CRD hydraulic subsystem

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion (scram) of control rods.

The hydraulic power required for scram is provided by high pressure water stored in the individual HCU's. Each HCU contains a scram accumulator (nitrogen-water), charged to high pressure and the necessary valves and components to scram two FMCRDs. Additionally, during normal operation, the HCU's provide a flow path for purge water to the associated FMCRDs.

The CRD hydraulic subsystem provides clean, demineralized water that is used to charge the scram accumulators and purge water flow to the FMCRDs during normal operation. The CRD hydraulic subsystem is also the source of pressurized water for purging the SDC pumps and filling the Nuclear Boiler System (NBS) reactor water level reference leg instrument lines.

The CRD system performs the following functions:

1. Control changes in core reactivity by positioning neutron-absorbing control rods within the core in response to control signals from the Rod Control and Information System (RC&IS).
2. Provides positioning of control rods in increments to enable optimized power control and core power shape in response to control signals from RC&IS.
3. Provides the ability to position large groups of rods simultaneously in response to control signals from RC&IS.
4. Supplies rod status and rod position data for rod pattern control, performance monitoring, operator display, and scram time testing to the RC&IS.
5. Supplies purge water flow to the FMCRDs for cooling and foreign material exclusion.
6. Supplies water for sampling to the Process Radiation Monitoring System (PRM).

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7. Supplies a continuous flow of water to the NBS to keep the reactor water level reference leg instrument lines filled.
8. Supplies charging water to fill and pressurize the HCU accumulators.
9. Supplies a makeup source for the RPV pressure test.
10. The FMCRD maintains Reactor Coolant Pressure Boundary (RCPB) integrity.
11. The FMCRD prevents rod ejection from the core due to a break in the drive mechanism pressure boundary or a failure of the attached scram line.
12. Supplies purge water for the SDC pumps.
13. Performs the scram follow function following any hydraulic scram signal.
14. Provides the capability for hydraulic scram in response to signals from the Anticipatory Trip System.
15. Provides the capability for Selected Control Rod Rapid Insertion (SCRRI) in response to signals from the Anticipatory Trip System.
16. In conjunction with RC&IS and the Emergency Rod Insertion Control Panel (ERICP), the CRD system blocks the withdrawal of control rods in response to signals from the Advanced Thermal Limit Monitor (ATLM) and the Multichannel Rod Block Monitor (MRBM).
17. In conjunction with RC&IS, the CRD system detects separation of the hollow piston from the ball nut or separation between the control rod blade and the hollow piston and imposes a rod withdrawal block.
18. Provides the capability for hydraulic scram upon receiving a scram signal from the SC1 I&C system.
19. Provides the capability for hydraulic scram that is independent from the SC1 hydraulic scram upon receiving the scram signal from the Diverse Protection System.
20. Performs the Alternate Rod Insertion (ARI) function to provide an alternate means of hydraulic scram.
21. Provides the capability for fast run-in of the FMCRDs upon receiving a signal from the FMCRD controllers with input from the ERICPs.
22. Provides maximum available flow to the RPV (as part of DL4b when a signal is received from the I&C systems).

The design bases and further discussion of both the RC&IS and the hydraulic scram function as well as their control interfaces with the CRD system, are presented in Chapter 7.

The CRD system separates the SC1 equipment from the rest of the system. The FMCRDs are mounted to the reactor vessel bottom head inside containment. The HCUs are housed in four rooms located directly outside of containment in the Reactor Building. These rooms are arranged around the periphery of the containment wall. The HCUs are connected to the FMCRDs by the scram insert piping that penetrates containment.

The balance of the hydraulic system equipment (e.g., pumps, valves, filters) is physically separated from the HCUs by barriers and is connected to the HCUs via the purge water header, scram accumulator charging water header and scram air header.

The system interfaces of the CRD system are shown in Figure 4.6-1.

4.6.2.1 Fine Motion Control Rod Drive Mechanism

The FMCRD used for positioning the control rod in the reactor core is an electro-hydraulic actuated mechanism (Figure 4.6-2 and Figure 4.6-3). An electric motor-driven ball nut and ball screw assembly positions the drive at both nominal increments and continuously over its entire range at a nominal speed. The FMCRDs also have the capability for motor-driven fast control rod insertion. Hydraulic pressure from the HCU is used for scrams. A single HCU powers the scram action of two FMCRDs, except for the FMCRD in the center of the core which has its own HCU. The FMCRD penetrates the bottom head of the RPV. The FMCRD does not interfere with refueling and is operative even when the head is removed from the RPV.

4.6.2.2 FMCRD Components

Simplified schematic of the FMCRD is provided in Figure 4.6-3. The basic elements of the FMCRD are as follows:

- Components required for electrical rod positioning or fine motion control (including the motor, brake release, ball screw, ball nut and hollow piston)
- Components required for hydraulic scram (including hollow piston and buffer)
- Components required for pressure integrity (including the middle flange, installation bolts and lower component)
- Rod position indication (position signal detectors)
- Reed position switches for scram surveillance and full-in indication
- Control rod separation detection devices (dual FMCRD separation switches)
- Bayonet coupling between the hollow piston and control rod
- Brake mechanism to prevent rod ejection in the event of a break in the FMCRD primary pressure boundary
- Ball check valve to prevent rod ejection in the event of a failure of the scram insert line
- Integral internal drive blowout support (to prevent drive blowout)
- Magnetic coupling

These features and functions of the FMCRD are described below.

Components for Fine Motion Control

The fine motion capability is achieved with a ball nut and ball screw arrangement driven by an electric motor. The ball nut is keyed to the guide tube (roller key) to prevent its rotation as it traverses axially as the ball screw rotates. A hollow piston rests on the ball nut and upward motion of the ball nut drives the control rod into the core. The weight of the control rod keeps the hollow piston and ball nut in contact during withdrawal.

The drive motor, located outside the pressure boundary, is magnetically coupled to the drive shaft located inside the pressure boundary. A splined coupling connects the drive shaft to the ball screw. The lower half of the splined coupling is keyed to the drive shaft and the upper half keyed to the ball screw. The tapered end of the drive shaft fits into a conical seat on the end of the ball screw to keep the two axially aligned. A drive shaft thrust bearing carries the entire weight of the control rod and drive internals.

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The axially moving parts are centered and guided by radial rollers. The ball nut and bottom of the hollow piston includes radial rollers bearing against the guide tube. Radially adjustable rollers at both ends of the labyrinth seal keep the hollow piston precisely centered in this region.

A stationary guide supports the top of the rotating ball screw against the inside of the hollow piston. A hardened bushing provides the circumferential bearing between the rotating ball screw and stationary guide. Rollers of the guide engage with axial grooves in the hollow piston to prevent the guide from rotating with the ball screw.

Components for Scram, Scram Follow, and Run-In

Opening of the scram valve in the HCU initiates the hydraulic scram action. High pressure water lifts the hollow piston off the ball nut and drives the control rod into the core. A spring washer buffer assembly stops the hollow piston at the end of its stroke. Departure from the ball nut releases spring-loaded latches in the hollow piston that engage slots in the hollow piston guide tube. These latches support the control rod and hollow piston in the inserted position. Simultaneous with the initiation of any hydraulic scram, a scram follow signal is generated such that each FMCRD motor drives the ball nut upward to a position just below the fully inserted and latched hollow piston. With the ball nut below the hollow piston, the piston remains latched to maintain the control rod in the fully inserted position.

The automatic run-in of the ball nut using the electric motor drive following the hydraulic scram provides a diverse means of rod insertion as a backup to the accumulator scram. During motor run-in, the hollow piston remains in contact with the upper surface of the ball nut. This results in the weight of the hollow piston/control rod assembly being supported by the ball nut with the hollow piston latches remaining in the retracted position.

Table 4.6-2 shows the scram performance requirements for the CRD system in terms of the maximum elapsed time for each control rod to attain the listed scram position (percent insertion) after loss of signal to the scram solenoid pilot valves (time zero).

FMCRD Pressure Boundary

The ASME Section III Class 1 RCPB components include the CRD housing (attached to the RPV), the FMCRD middle flange including the ball check valve, and the FMCRD lower component housing (which encloses the lower part of the drive). The middle flange is attached to the CRD housing by four threaded bolts. The lower housing is held to the middle flange and secured to the CRD housing by a separate set of eight main mounting bolts that become a part of the reactor pressure boundary. This arrangement permits removing the lower housing without disturbing the rest of the drive. Removing the lower housing transfers the weight of the drive line from the drive shaft to a seat in the middle flange. Both the ball screw and drive shaft are locked to prevent rotation while the two are separated.

The part of the drive inserted into the CRD housing is contained within the outer tube. The outer tube is the drive hydraulic scram pressure boundary, eliminating the need for designing the CRD housing for scram pressure. The outer tube is welded to the middle flange at the bottom and is attached at the top with the CRD blowout support, which bears against the CRD housing (refer to Figure 4.6-8). The blowout support and outer tube are attached by a slip-type connection that accounts for any slight variation in length between the drive and the CRD housing.

Purge water continually flows through the drive. The water enters through the ball check valve in the middle housing and flows around the hollow piston into the reactor. O-rings seal the lower housing. A labyrinth seal near the top of the drive restricts the flow into the reactor. During a scram, the labyrinth seals discharge the high-pressure scram water into the RPV without adversely affecting the movement of the hollow piston.

Rod Position Indicator

Control rod position indication is provided by the FMCRDs to the control system by a position detection system that consists of position detectors and position signal converters.

Each FMCRD provides two position detectors, one for each control system channel, in the form of signal detectors directly coupled to the motor shaft. This configuration provides continuous detection of rod position during normal operation.

Scram Position Indicator

Scram position indication is provided by a series of magnetic reed switches to allow for measurement of adequate drive performance during scram. The magnetic switches are located at intermediate intervals over 60% of the drive stroke. They are mounted in a probe exterior to the CRD housing. A magnet in the hollow piston trips each reed switch in turn as it passes by.

As the bottom of the hollow piston contacts and enters the buffer, a magnet is lifted that operates a reed switch indicating scram completion. This continuous full-in indicating switch (shown on Figure 4.6-4) provides indication whenever the drive is at the full-in latched position or above. In the event of failure of the buffer mechanism a buffer contact reed switch is positioned above the full-in switch to provide failed buffer indication.

Control Rod Separation Detection

Two redundant switches are provided to detect the separation of the hollow piston from the ball nut or between the control rod blade and the hollow piston (see Figure 4.6-5). The separation switches function to detect a detached control rod and cause a rod withdrawal block, thereby preventing a rod drop accident. Actuation of either switch also initiates an alarm in the Main Control Room (MCR).

During normal operation, the weight of the control rod and hollow piston resting on the ball nut causes the ball screw assembly to compress a spring on which the lower half of the splined coupling between the drive shaft and ball screw assembly rests (the lower half of the splined coupling is also known as the "weighing table"). When the hollow piston separates from the ball nut, or when the control rod separates from the hollow piston, the spring is unloaded and pushes the weighing table and ball screw assembly upward. This action causes a magnet in the weighing table to operate the reed switches located outside the lower housing.

Bayonet Couplings

There are two bayonet couplings associated with the FMCRD. The first is the FMCRD/control rod guide tube/housing interface (see Figure 4.6-8). This bayonet locks the FMCRD and the base of the control rod guide tube to the CRD housing and functions to retain the guide tube during normal operation and dynamic loading events. The bayonet also holds the FMCRD against ejection in the event of a hypothetical failure of the CRD housing weld. The locating pin on the core plate that engages the flange of the control rod guide tube and the bolt pattern on the FMCRD/housing flange, assure proper orientation between the control rod guide tube and the FMCRD to confirm that the bayonet is properly engaged.

The second bayonet (Figure 4.6-6) is located between the control rod and FMCRD. The coupling spud at the top end of the hollow piston engages and locks into a mating socket at the base of the control rod. The coupling requires a 45° rotation for engaging or disengaging. Once locked, the drive and rod form an integral unit that must be manually unlocked by specific procedures before the components can be separated.

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The FMCRD design allows the coupling integrity of this second bayonet to be checked by driving the ball nut down into an overtravel out position. After the weighing spring has raised the spindle assembly to the limit of its travel, further rotation of the spindle in the withdraw direction will drive the ball nut down away from the piston (assuming the coupling is engaged). If the hollow piston is not properly coupled to the control rod, the hollow piston will remain in contact with the ball nut and move with it to the overtravel position. A reed switch at the overtravel position will detect this movement of the hollow piston and provide alarm in the MCR.

FMCRD Brake and Ball Check Valve

The FMCRD design incorporates an electromechanical brake (Figure 4.6-7) keyed to the motor shaft. The brake is normally engaged by passive spring force when the FMCRD is stationary. It is disengaged for normal rod movements by signals from the RC&IS. Disengagement is caused by the energized magnetic force overcoming the spring load force. The braking torque and the magnetic coupling torque between the motor and the drive shaft are sufficient to prevent control rod ejection in the event of failure in the pressure retaining parts of the drive mechanism. The brake is designed so that its failure will not prevent the control rod from rapid insertion (scram) or motor run-in. The brake is a unidirectional mechanism that allows the motor to drive the control rod only in the insert direction if the brake is unintentionally engaged.

The electromechanical brake is located in the motor unit. The stationary spring-loaded disk and coil assembly is contained within the brake mounting bolted to the bottom of the motor unit top flange. The rotating disk is keyed to the motor unit output shaft.

A ball check valve is located in the middle flange of the drive at the scram inlet port. The check valve actuates to close the scram inlet port under conditions of reverse flow caused by a break of the scram line. This prevents the loss of pressure to the underside of the hollow piston and the generation of loads on the drive that could cause a rod ejection.

Integral Internal Blowout Support

The CRD blowout support (Figure 4.6-8) is designed to prevent ejection of the FMCRD and the attached control rod in the unlikely event of:

- A failure of the weld between the CRD housing and the stub tube penetration of the bottom head, or a failure through the housing along the fusion line of the weld
- A total failure of all flange bolts attaching the lower component flange and middle flange to the CRD housing flange

The internal CRD blowout support utilizes the FMCRD outer tube integral with support internal to the RPV to provide the anti-ejection support. The outer tube, which is welded to the drive middle flange, attaches by a bayonet lock to the control rod guide tube base. The core plate in turn supports the control rod guide tube.

With a CRD housing failure, the weight plus pressure load acting on the drive and housing would tend to eject the drive. In this event, the control rod guide tube base remains supported by the intact housing extension inside the RPV and the FMCRD remains supported in turn by its positive lock to the control rod guide tube base. Coolant leakage is restricted to the small annular area between the CRD outer tube and the inside of the CRD housing. In the event of a total failure of the weld itself, with the entire CRD housing intact, the housing would tend to be driven downward by the total weight plus RPV pressure. However, after the interconnected assembly of the housing, FMCRD and control rod guide tube move down a short distance, the flange at the top of the control rod guide tube contacts the core plate and stops further movement of the assembly. Since the FMCRD is positively locked to the control rod guide tube base, it cannot eject. In this case, the housing that bears on top of the blowout support is also prevented from leaving the

penetration. Coolant leakage for this condition is restricted to the small annular area between the outside of the CRD housing and the inside of the penetration stub tube.

The FMCRD design provides an anti-rotation device that engages when the lower component is removed for maintenance (Figure 4.6-11). This device prevents rotation of the ball screw and hence prevents control rod motion when the lower component is removed. The anti-rotation device consists of:

- The coupling piece on the bottom of the ball screw that engages with the lower component drive shaft
- The back seat of the middle flange

The coupling between the lower component drive shaft and ball screw is splined to permit removal of the lower housing. The underside of the coupling piece on the ball screw has a circumferentially splined surface that engages with a mating surface on the middle flange backseat when the ball screw is lowered during lower component removal. When engaged, ball screw rotation is prevented. In the unlikely event of total failure of all the drive flange bolts attaching the lower component flange and middle flange of the drive to the housing flange, the anti-rotation device will engage when the lower component falls. The middle flange/outer tube/CRD blowout support will be restrained by the control rod guide tube base bayonet coupling, thus preventing rod ejection.

Magnetic Coupling

The magnetic coupling is located at the bottom of the lower component (Figure 4.6-3). It is employed to achieve leak-free operation of the FMCRD without seals. The magnetic coupling consists of an inner and an outer rotor. The inner rotor is located inside the lower component pressure boundary. The outer rotor is located outside the pressure boundary. Each rotor has permanent magnets mounted on it. As a result, the inner and outer rotors are locked together by the magnetic forces acting through the pressure boundary and work as a synchronous coupling. The outer rotor is coupled with the motor unit and driven by the motor such that the inner rotor follows the rotation of the outer rotor.

The magnetic coupling is designed so that its maximum coupling torque exceeds the maximum torque of the motor unit to prevent decoupling or slippage due to motor torque.

The magnetic coupling is designed to not have any undesirable effects on other magnetic sensitive subcomponents.

Materials of Construction

The materials of construction for the FMCRD components which form part of the RCPB are discussed in Chapter 4, Section 4.5. For the non-pressure retaining FMCRD components, materials are selected for compatibility with the reactor coolant, wear resistance, corrosion resistance and material strength to ensure reliability and design life requirements are met in the BWRX-300 environment.

4.6.2.2.1 Hydraulic Control Units

Upon receipt of a scram signal, each HCU furnishes pressurized water for hydraulic scram to two FMCRD units (except for the FMCRD in the center of the core which has its own HCU). Additionally, each HCU provides the capability to adjust purge flow to the two drives. A test port is provided on the HCU for connection to a portable test station to allow for controlled venting of the scram insert line to test the FMCRD ball check valve during plant shutdown. The check valves shown inside the HCU boundary function to close under system pressure, fluid flow and temperature conditions during scram. The check valves ensure that the water stored in the HCU

accumulator is delivered to the FMCRDs to accomplish the scram function. A simplified single line diagram of the HCU is provided in Figure 4.6-9. The major components of each HCU are provided below.

Scram Solenoid Valve Assembly

The scram solenoid valve assembly consists of two solenoid valves which control the position of the scram valve. The solenoid valves are normally energized and closed. Upon loss of power to the solenoids, the valves open which vents air to open the scram valve. The assembly is designed so that the power must be removed from both solenoids before air pressure can be discharged from the scram valve operator. This prevents the inadvertent scram of the drives associated with a given HCU in the event of a failure of one of the valve solenoids.

Scram Valve

The scram valve opens to supply pressurized water to the bottom of the drive piston. This quick opening globe valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A position indicator switch on this valve energizes a light in the MCR as soon as the valve starts to open.

Scram Accumulator

The scram accumulator stores sufficient energy to fully insert two control rods at any anticipated reactor pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A check valve in the accumulator charging line prevents loss of water pressure in case charging water header pressure is lost. During normal plant operation, the accumulator piston is seated at the bottom of its cylinder. Pressure sensors provide local and MCR nitrogen pressure indication. Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the MCR. The alarm would prompt operator action to repressurize the nitrogen bottle using an external supply of nitrogen gas. To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A level sensor actuates an alarm in the MCR if water leaks past the piston barrier and collects in the accumulator instrumentation block.

Purging Panel

The purging panel controls the purge water flow to the associated FMCRDs. Each panel has a restricting orifice in the purge water line to control the purge water flow rate. This orifice maintains the flow at a constant value while the drives are stationary. A bypass line containing a solenoid-operated valve is provided around this orifice. The valve is signaled to open and increase the purge water flow whenever either of the two associated FMCRDs is commanded to insert by the RC&IS. During FMCRD insertion cycles, the hollow piston moves upward, leaving an increased volume for water within the drive. Opening of the purge water makeup valve increases the purge flow to offset this volumetric increase and precludes the backflow of reactor water into the drive, thereby preventing long-term drive contamination.

4.6.2.2.2 Control Rod Drive Hydraulic Subsystem

The CRD hydraulic subsystem supplies clean demineralized water to the following:

1. HCU accumulators for charging
2. FMCRDs for purge water
3. RPV for makeup
4. SDC pumps for seal purge water

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5. NBS reactor water level reference leg instrument lines
6. PRM for sampling

The subsystem consists of two trains, each providing the required functions with the pumps, valves, filters, piping, and instrumentation described below. A simplified single line diagram for the CRD is provided in Figure 4.6-10.

The equipment and arrangement are comprised of the following:

1. CRD system pump suction piping from the Condensate and Feedwater Heating System (CFS) or the Condensate Storage Tank (CST).
2. Two 100% capacity trains. One train is normally operating, and the other train is in standby. Each train consists of the following:
 - a. One 100% capacity purge water pump with a suction and discharge filter
 - b. One 100% capacity charging water pump with a suction and discharge filter
 - c. Each purge pump has a minimum flow line at its discharge which is routed to the condenser or the CST
 - d. Each purge water and charging water pump has discharge pressure sensor that provides indication locally and in the MCR
 - e. Each charging water pump has a bypass line at its discharge which is routed to the CST
 - f. The distribution piping, branching off the main discharge line is downstream of the purge water pump discharge filter, and provides:
 - i. Purge water flow to the SDC pumps
 - ii. Purge water flow to the reactor water level reference leg instrument lines
 - iii. Sample water to the PRM system
 - g. Downstream of the purge water filters is a pressure sensor used to start the standby purge pump in the event of failure of the in-service purge pump
 - h. A flow element and sensor are used to control the position of a flow control valve which controls the total purge water flow rate to all the HCU's
3. The common purge water header directs purge water flow to the individual HCUs, where it is then directed to the FMCRDs via the scram insert piping.
4. The common charging water header includes a bladder-type nitrogen accumulator used to maintain charging water header pressure when the charging pump is not running.
5. Piping connections are provided between the individual HCUs and their respective FMCRDs
6. A scram air header, containing redundant pressure control valves, is connected to the scram valves in the individual HCUs
7. Solenoid-operated ARI valves are positioned in the scram air header to depressurize and isolate the scram air header on receipt of a scram signal

4.6.2.2.3 Control Rod Drive System Operation

The operating modes of the CRD system are described in this subsection.

Normal Operation

Normal operation is defined as operating periods where the CRD system provides charging pressure to the HCU and supplies purge water to the FMCRDs, the SDC pumps, the NBS reactor water level reference leg instrument lines, and the PRM.

One of the two purge water pumps is operating to pressurize the purge water header from the CFS or CST. The other purge water pump is shut down and on standby. A constant portion of the purge water pump discharge is continuously bypassed back to the condenser or CST in order to maintain a minimum flow through the pump. This prevents overheating of the purge pump if the discharge line is blocked. The standby purge water pump provides a full capacity backup capability to the operating pump and starts automatically if failure of the operating pump is detected by pressure instrumentation located in operating pump's discharge piping. The purge water flow control valve automatically regulates the purge water flow to the FMCRDs. The purge water flow rate is indicated in the MCR.

To maintain the ability to scram, the charging water header pressure is maintained by one of the two charging water pumps. The charging pumps are designed for intermittent operation. These pumps also draw water for the CFS or CST, with pump bypass line returning to the CST. The charging water header is maintained at the pressure required for the scram accumulators to scram the control rods. The HCU scram valves remain closed except during scram. During normal operation, no flow passes through the charging water header. A separate accumulator, located on the charging water header, assists with maintaining the header at the required pressure. When intermittent pump operation is used the pump automatically starts on low charging water pressure and stops when pressure is restored. Pressure in the charging water header is monitored continuously and displayed in the MCR along with a low-pressure alarm.

Control Rod Insertion and Withdrawal

The FMCRD design provides the capability to move a control rod in fine steps. Normal control rod movement is under the control of the RC&IS. The RC&IS controls the input of actuation power to the FMCRD motor from the electrical power supply to complete a rod motion command. The FMCRD motor rotates a ball screw, which in turn causes the vertical translation of a ball nut on the ball screw. This motion is transferred to the control rod via a hollow piston that rests on the ball nut. Thus, the hollow piston with the control rod is raised (inserted) or lowered (withdrawn) depending on the direction of rotation of the FMCRD motor and ball screw.

During a control rod insertion, the solenoid-operated purge water makeup valve within the associated HCU is opened to increase the purge water flow to the FMCRD. The increased flow offsets the volumetric displacement within the FMCRD as the hollow piston is inserted into the core and prevents reactor water from being drawn back into the drive.

Scram

In response to an automatic or a manual scram the power is interrupted to both scram solenoid valve assembly coils which vent the air from the scram valve operator in each HCU. Venting of the air causes the scram valves to open by spring action. When the scram valve opens in the associated HCU, hydraulic pressure provides an insertion force to the respective FMCRD. When the hydraulic force is applied, the FMCRD's hollow piston lifts off the ball nut and inserts the control rod rapidly. The water displaced from the FMCRD is discharged into the RPV. Indication that the scram has been successfully completed (i.e., all rods full-in position) is displayed in the MCR.

Simultaneously with the hydraulic scram, each FMCRD motor is signaled to start from the ERICP. This signal causes the FMCRD motor to drive the ball nut upward to a position just below the fully inserted and latched hollow piston. This action is known as the scram follow function.

After reset of the scram logic, each scram valve recloses and the CRD charging water pump recharges the HCU accumulators.

Reactor scram logic is described in Chapter 7.

Select Control Rod Rapid Insertion

During SCRRRI, a predefined set of control rods insert rapidly. Upon receipt of the SCRRRI initiation signal, the CRD purge water flow control valve opens to provide increased purge water flow to the associated HCU. At those HCUs, the solenoid-operated purge water makeup valve opens to increase the purge water flow to the FMCRD to prevent reactor water from being drawn back into the drives.

Alternate Rod Insertion

The ARI function of the CRD System is initiated by the ERICP which provides an alternate, diverse and independent means for actuating hydraulic rod insertion. The signals to initiate the ARI function are the same as those for automatic and manual scrams. Following receipt of any scram signal, the ARI solenoid-operated valves on the scram air header actuate to depressurize the scram air header which opens all HCU scram valves. The FMCRDs then insert the control rods hydraulically. The same signals that initiate ARI simultaneously initiate an FMCRD scram follow function. Each vent path has two valves in series such that both valves are required to actuate to cause depressurization to prevent spurious ARI actuation.

4.6.2.3 Instrumentation and Control

4.6.2.3.1 Safety Class 1 Instrumentation and Control System

The SC1 I&C system is described in Chapter 7.

4.6.2.3.2 Safety Class 2 and 3 Instrumentation and Control System

The SC2 and SC3 instrumentation and control for the CRD system is described in Chapter 7, Section 7.3.

4.6.2.4 Power Supplies

The SC2 and SC3 electrical distribution system provides power to the CRD purge pump motors, CRD charging water pump motors, and CRD FMCRD motors.

4.6.2.5 Qualification

Equipment qualification requirements, including environmental, dynamic, functional, and Electromagnetic Compatibility (EMC) is described in Chapter 3, Section 3.9. The qualification requirements are based on equipment classifications described in Chapter 3, Section 3.2, Appendix 3A.

4.6.3 Safety Evaluation of the CRD System

4.6.3.1 Scram Time

The control rod scram function of the CRD system provides the negative reactivity insertion required by the Safety Design Bases in Subsection 4.6.1. The required scram time is used in the Chapter 15 safety analyses.

4.6.3.2 Scram Reliability

Key system features resulting in high scram reliability include:

- Scram valves open by spring action and are normally held closed by pressurized control air
- To cause hydraulic scram, a de-energizing reactor trip signal is provided to solenoid-operated pilot valves that vent the control air from the scram valves for opening
- The SC1 I&C hydraulic scram is designed so that the HCU scram signal independently initiates a hydraulic scram demand, from whatever source, regardless of any other rod positioning signal
- The FMCRD hollow piston and guide tube are designed so they do not restrain or prevent control rod insertion during scram
- Each FMCRD mechanism initiates electric motor-driven insertion of its control rod simultaneous with the initiation of hydraulic scram upon receipt of I&C signal. This provides a diverse means to assure control rod insertion
- The system is “fail-safe” in that loss of either electrical power to the scram solenoids, or loss of control air pressure to the scram valve operator, causes a scram
- Departure from the ball-nut releases spring-loaded latches in the hollow piston to engage slots in the guide tube
 - These latches support the hollow piston in the fully inserted position
 - Following a hydraulic scram insertion, the control rod cannot be withdrawn until the ball-nut is driven up, re-engaged, and the hollow piston de-latched from the guide tube
- The design also includes ARI pilot valves on the control air header, which serves all 29 scram valves
 - The ARI pilot valves are energized-to-actuate, and provide an alternate path to vent control air and open all scram valves resulting in hydraulic insertion of all control rods

4.6.4 Testing, Inspection, and Maintenance

The BWRX-300 inspection requirements provide mandatory Preservice Inspection, In-Service Inspection (ISI), and In-Service Testing (IST) requirements for components and systems. Preservice Inspection, ISI, and IST requirements are consistent with ASME BPVC, Section XI and ASME Operation and Maintenance of Nuclear Power Plants (OM Code).

The Preservice Inspection, ISI, and IST requirements include examinations, inspections, and testing of the SC1 portions of the CRD system that are designed and installed in accordance with ASME BPVC Section III, applicable Canadian codes and standards, and International Atomic Energy Agency (IAEA) Safety Standards. In addition, this specification includes basic design control considerations within ASME NQA-1, Part 2.

Any required surveillance and post-maintenance testing are determined as part of the work planning / work release approval process.

FMCRD Maintenance

The FMCRD is designed to permit coupling and uncoupling of the control rod from either below the reactor vessel without necessitating the removal of the reactor vessel head or above the reactor vessel with the reactor vessel head removed.

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Maintenance procedures prohibit coincident removal of the control rod and FMCRD of the same assembly. In addition, contingency procedures address core and spent fuel cooling capability and mitigative actions during FMCRD replacement with fuel in the vessel.

The FMCRD design also allows for separate removal of the motor unit, position indicator probe, Separation Indicator Probe, and lower component for maintenance during plant outages without disturbing the upper assembly of the drive. While these FMCRD components are removed for servicing, the associated control rod is maintained in the fully inserted position by one of two mechanical locking devices that prevent rotation of the ball screw and drive shaft.

4.6.5 References

None.

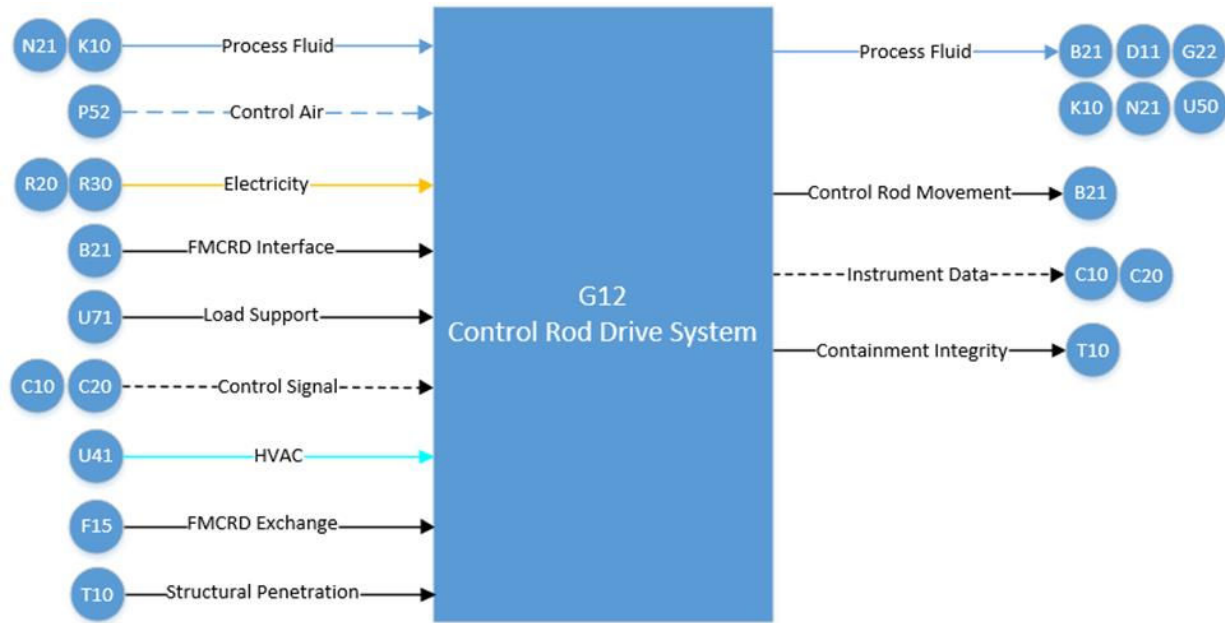
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Table 4.6-2: Control Rod Insertion Versus Time

Reactor Vessel Bottom Gauge Pressure MPaG (psig)	Rod Insertion Position	Required Maximum Time (s) ⁽²⁾
≤ 7.48 MPaG (1085 psig)	10 %	≤ 0.42
	40 %	≤ 1.00
	60 %	≤ 1.44
	100 %	≤ 2.80
≤ 8.75 MPaG (1269 psig)	10 %	≤ 0.46
	40 %	≤ 1.20
	60 %	≤ 1.71
	100 %	≤ 3.70
≤ 9.48 MPaG (1375 psig)	10 %	≤ 0.56
	40 %	≤ 1.40
	60 %	≤ 2.03
	100 %	≤ 4.20

(2) All times are after deenergizing the scram solenoids

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Legend:

B21 Nuclear Boiler System
C10/C20 Safety Class 1/2 Instrumentation and Control System
D11 Process and Radiation Monitoring
F15 Refueling Equipment and Servicing
N21 Condensate and Feedwater System
K10 Liquid Waste Management System
P52 Plant Pneumatics System
R20 Safety Class 2 and 3 Electrical Distribution System
R30 Non-Safety Electrical Distribution System
T10 Primary Containment
U71 Reactor Building Structure
U41 Heating Ventilation and Cooling System
U50 Equipment and Floor Drain System

Figure 4.6-1: CRD Interfaces Diagram

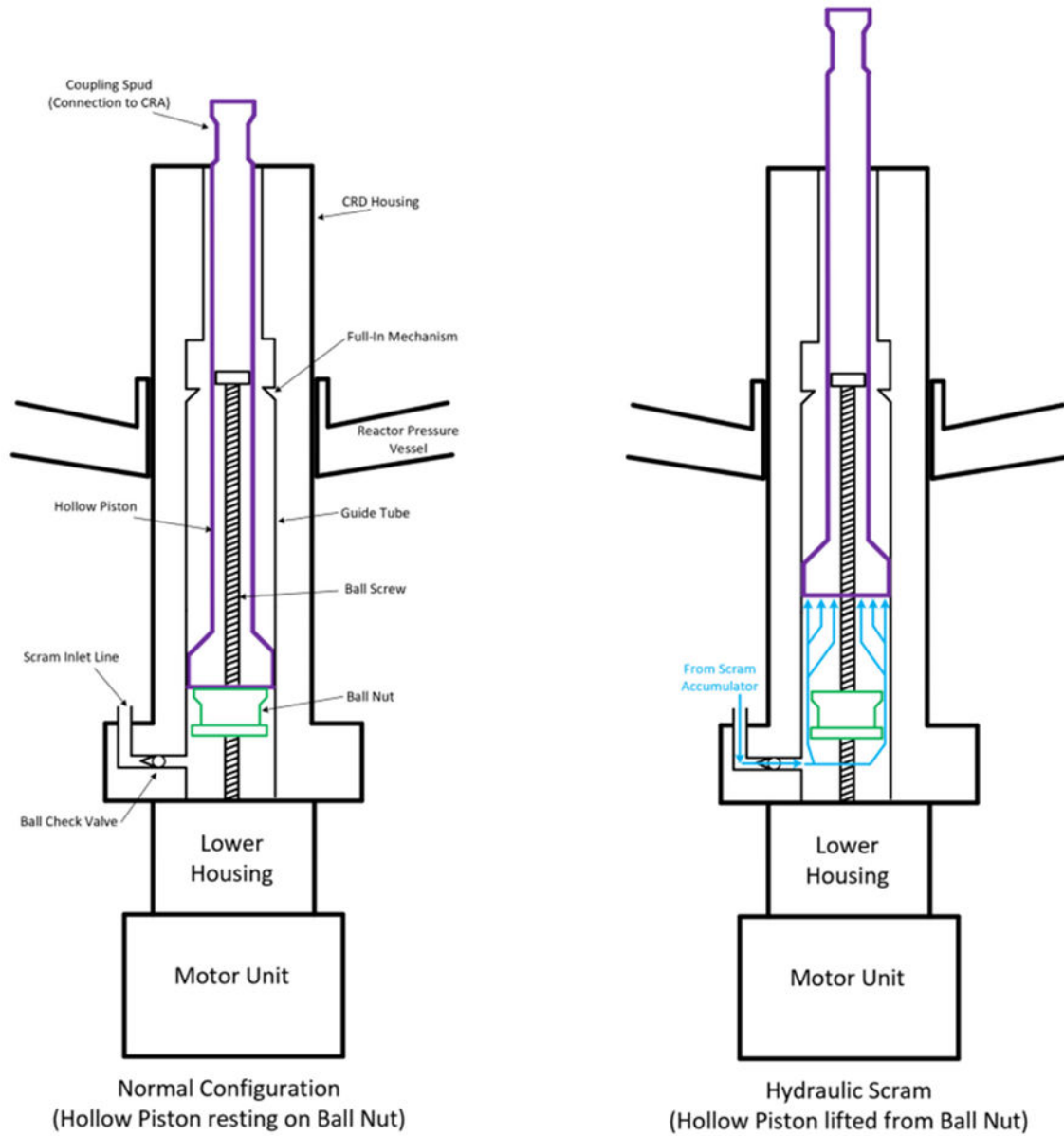


Figure 4.6-2: Magnet Coupling Fine Motion Control Rod Drive Cross-Section

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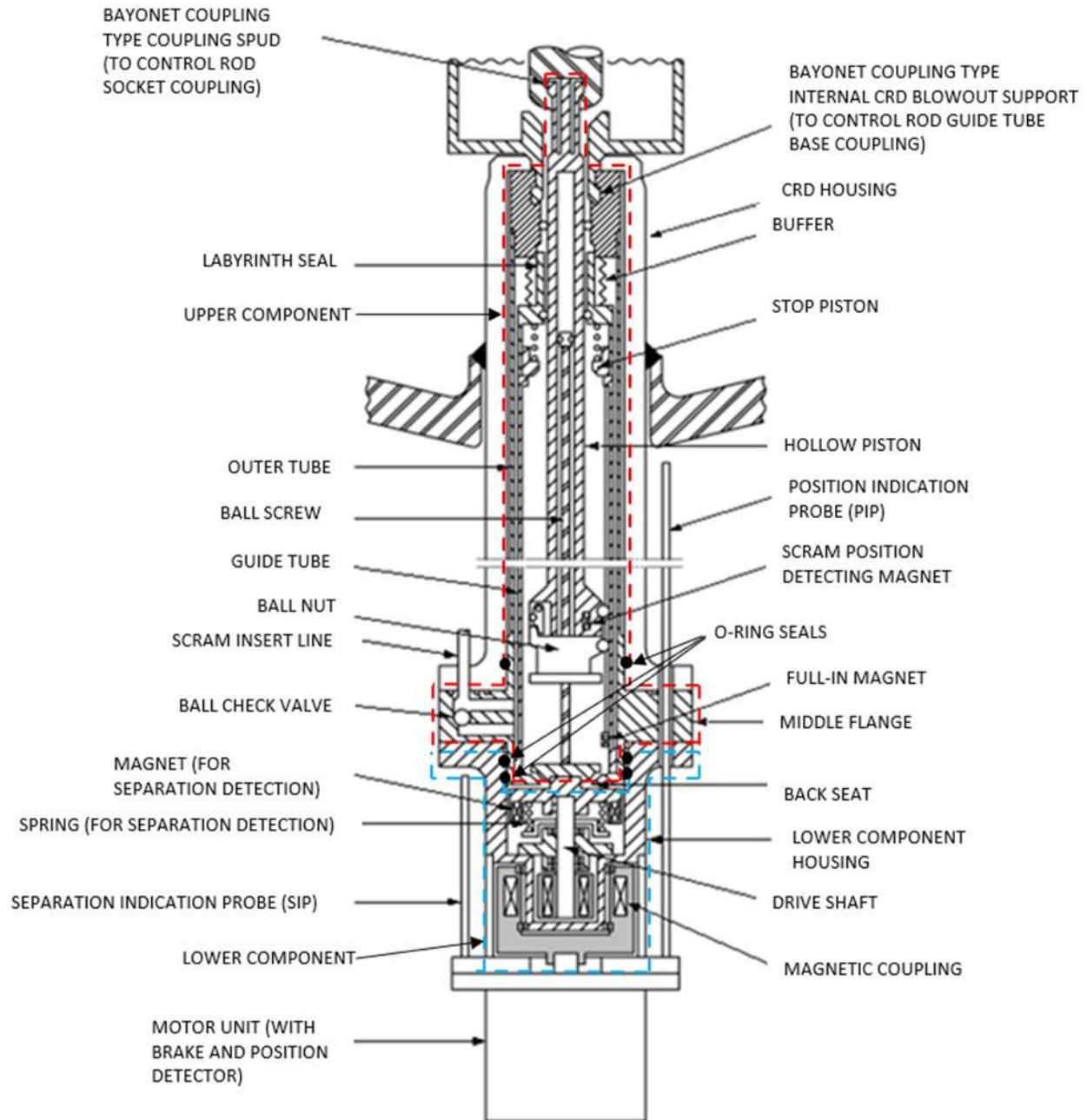


Figure 4.6-3: Schematic Representation of Magnet Coupling Fine Motion Control Rod Drive

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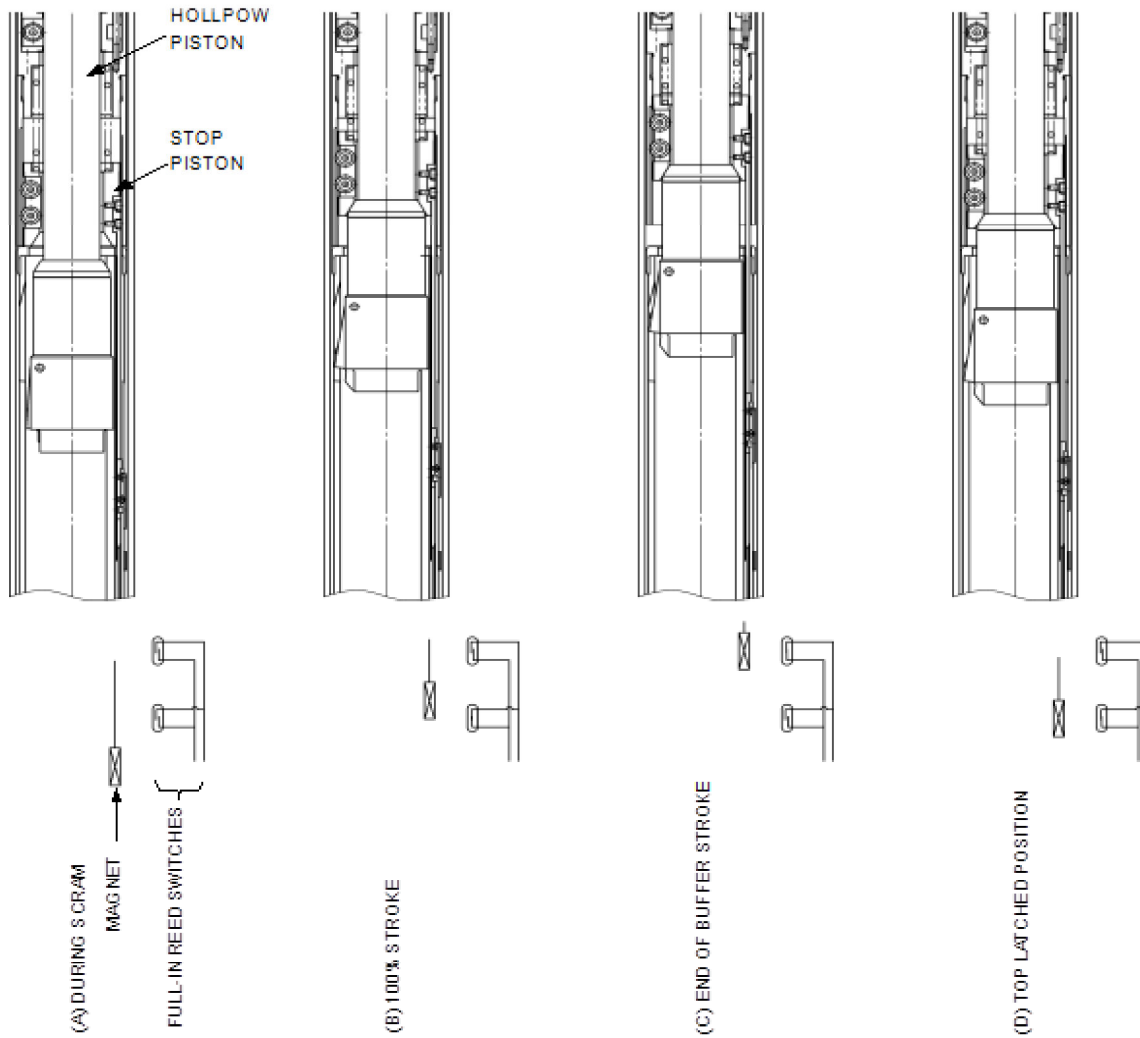


Figure 4.6-4: Representative Continuous Full-in Indicating Device

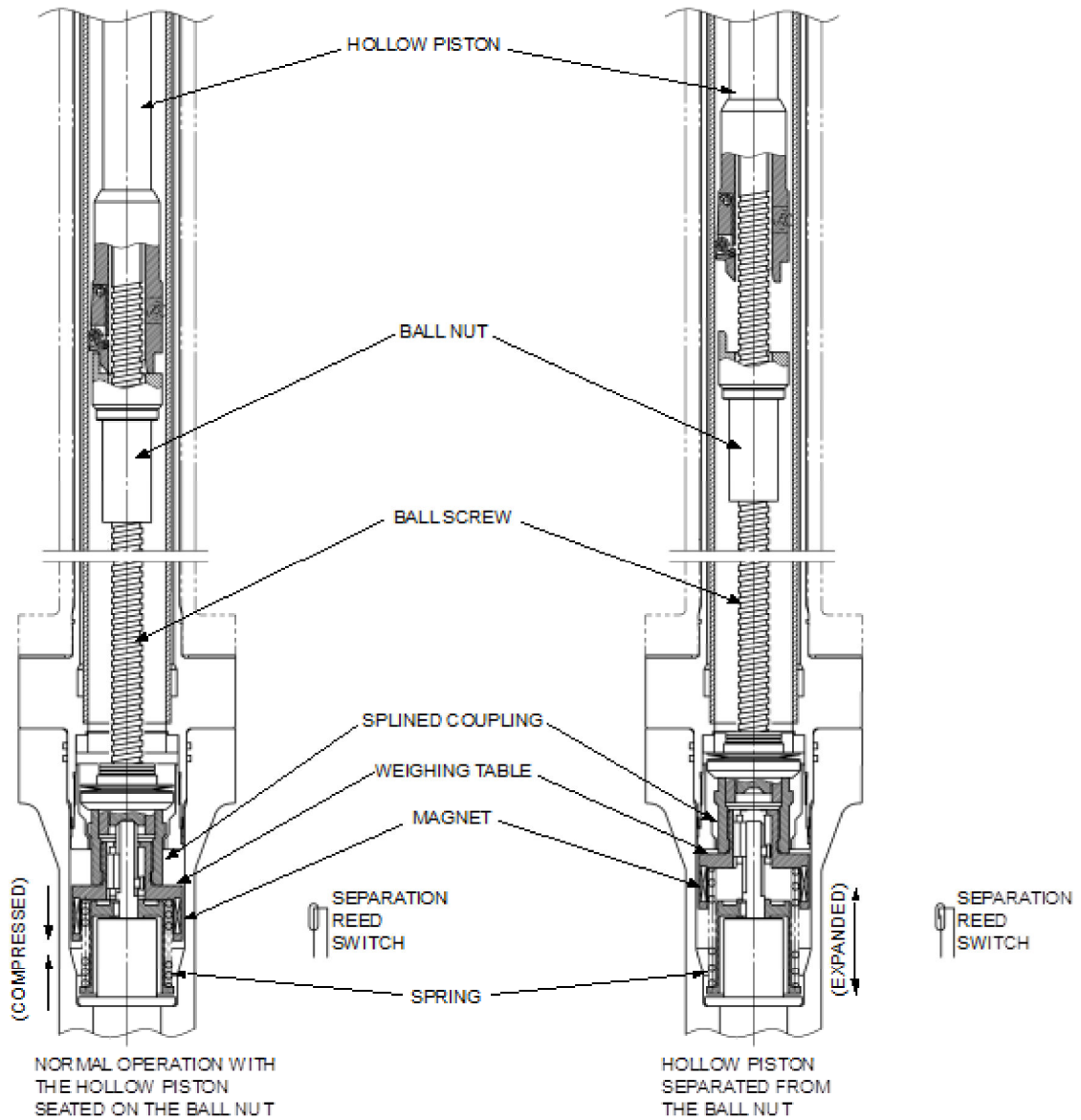


Figure 4.6-5: Representative Control Rod Separation Detection

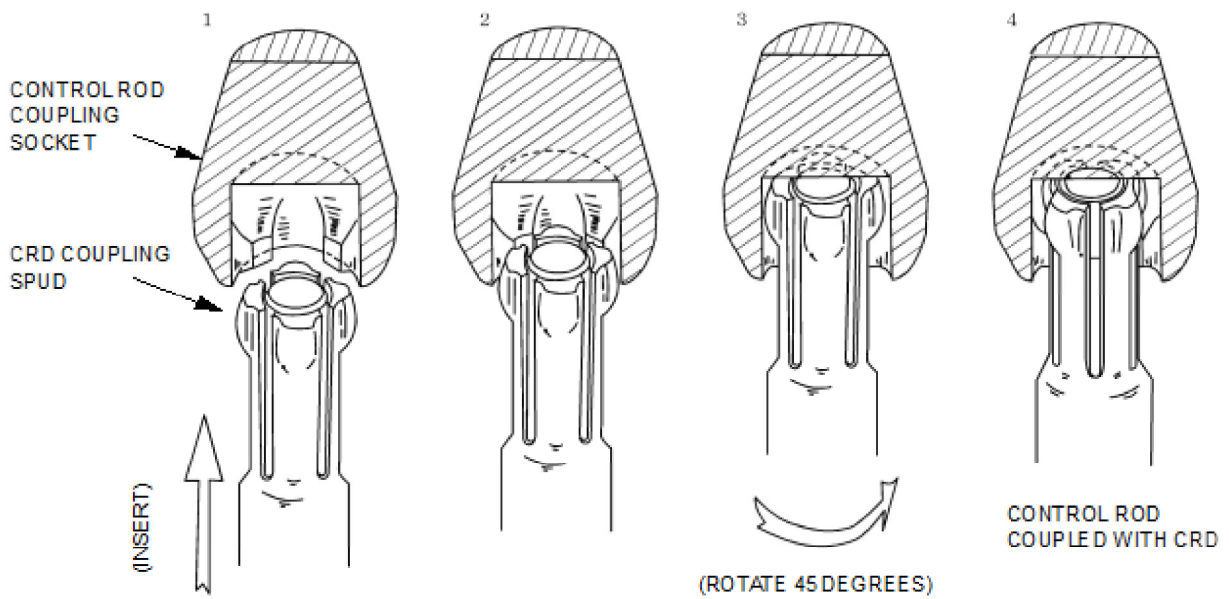


Figure 4.6-6: Representative Control Rod to Control Rod Drive Coupling

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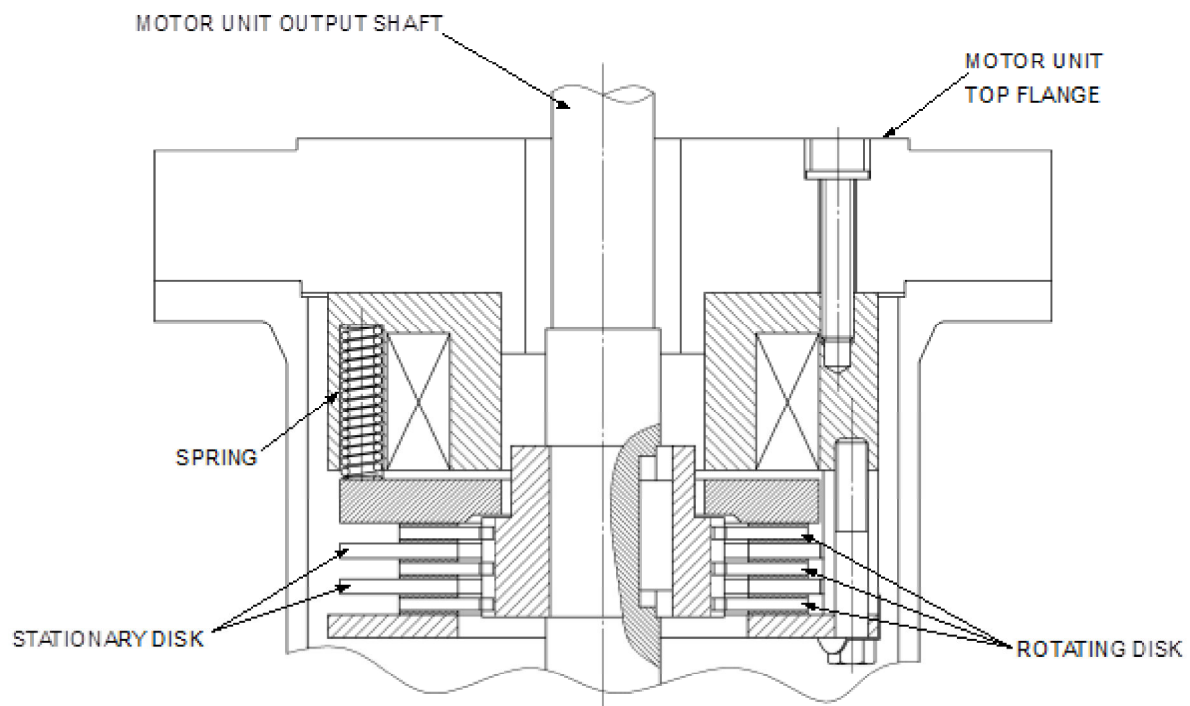


Figure 4.6-7: Representative FMCRD Electro-Mechanical Brake

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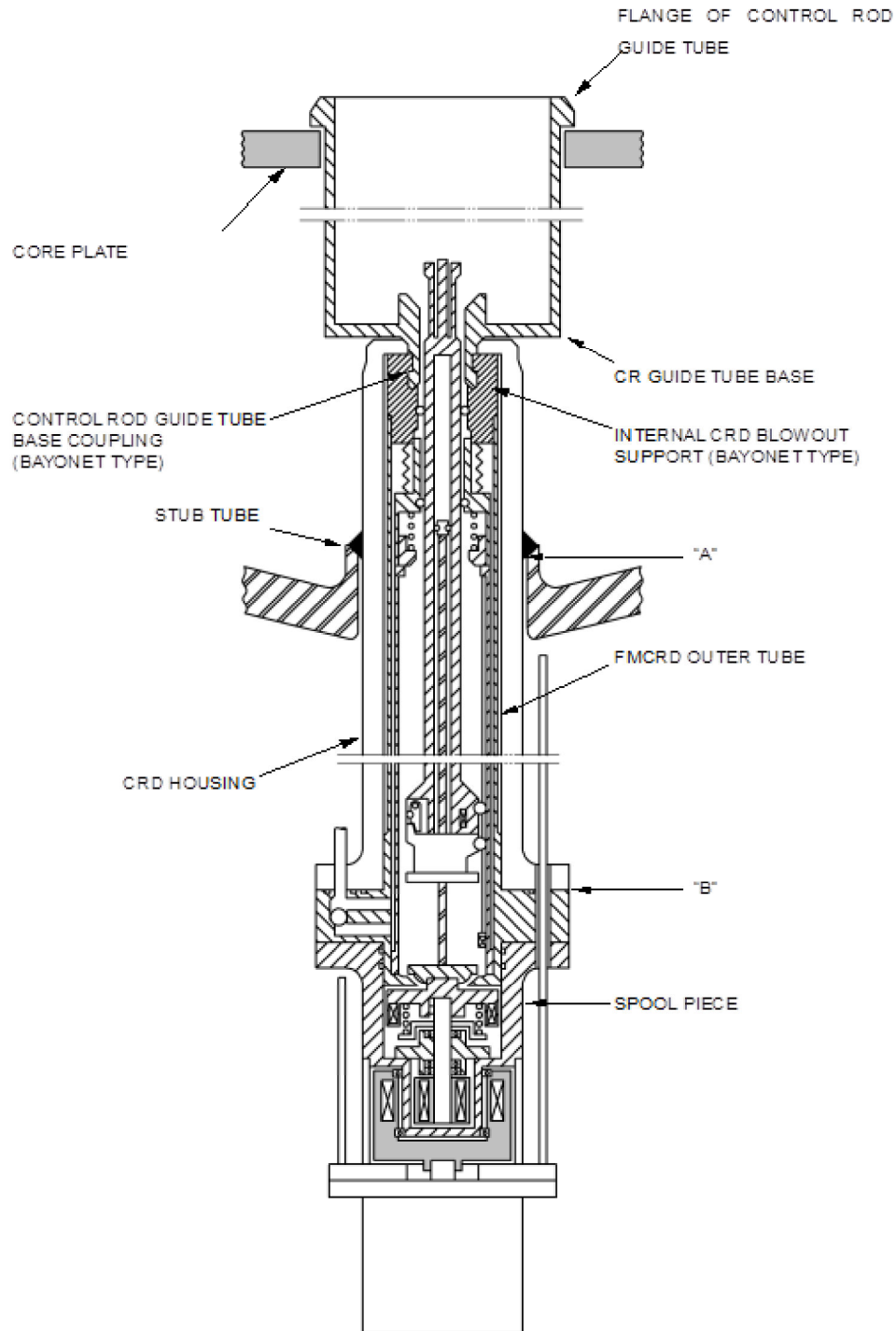


Figure 4.6-8: Representative Internal CRD Blowout Support Schematic

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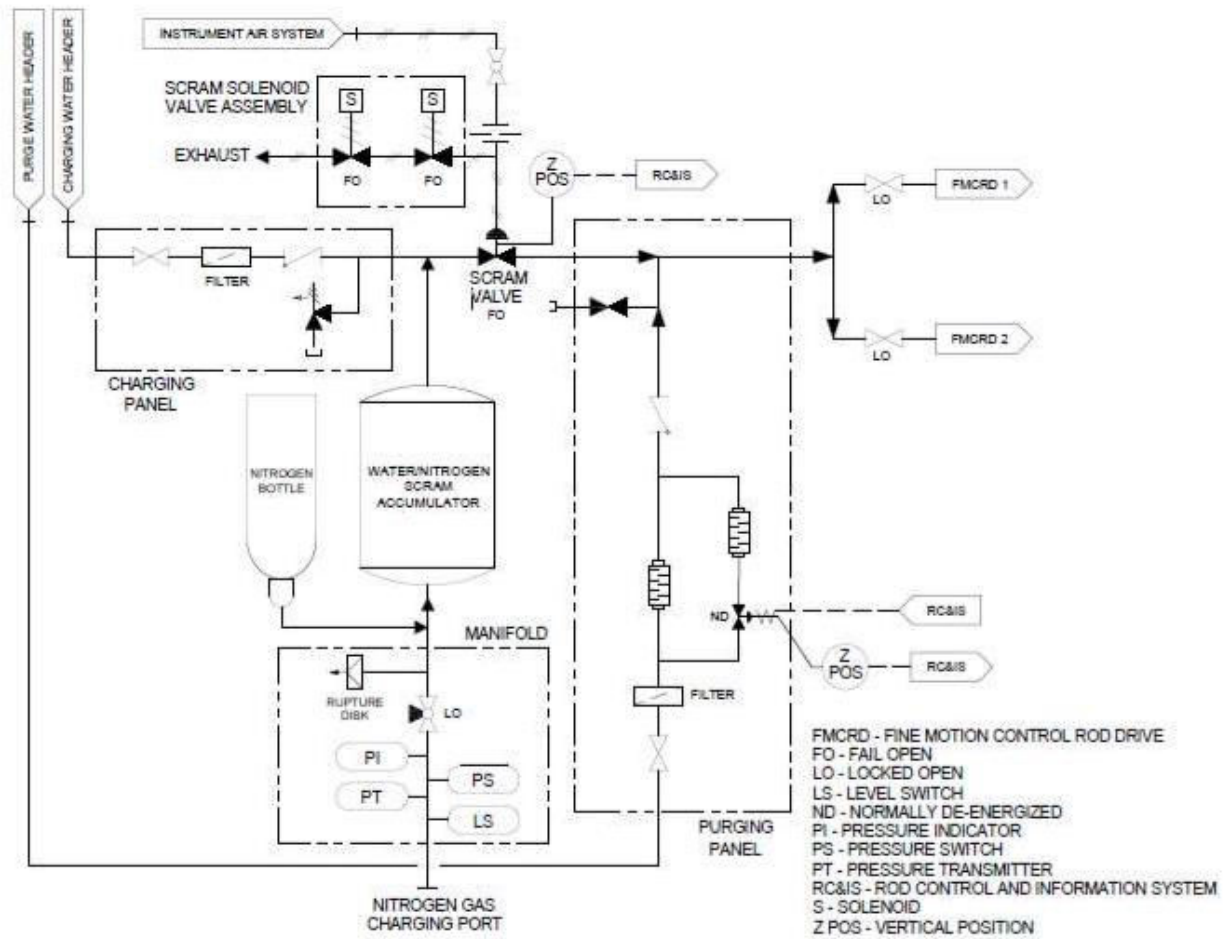


Figure 4.6-9: HCU Simplified P&ID

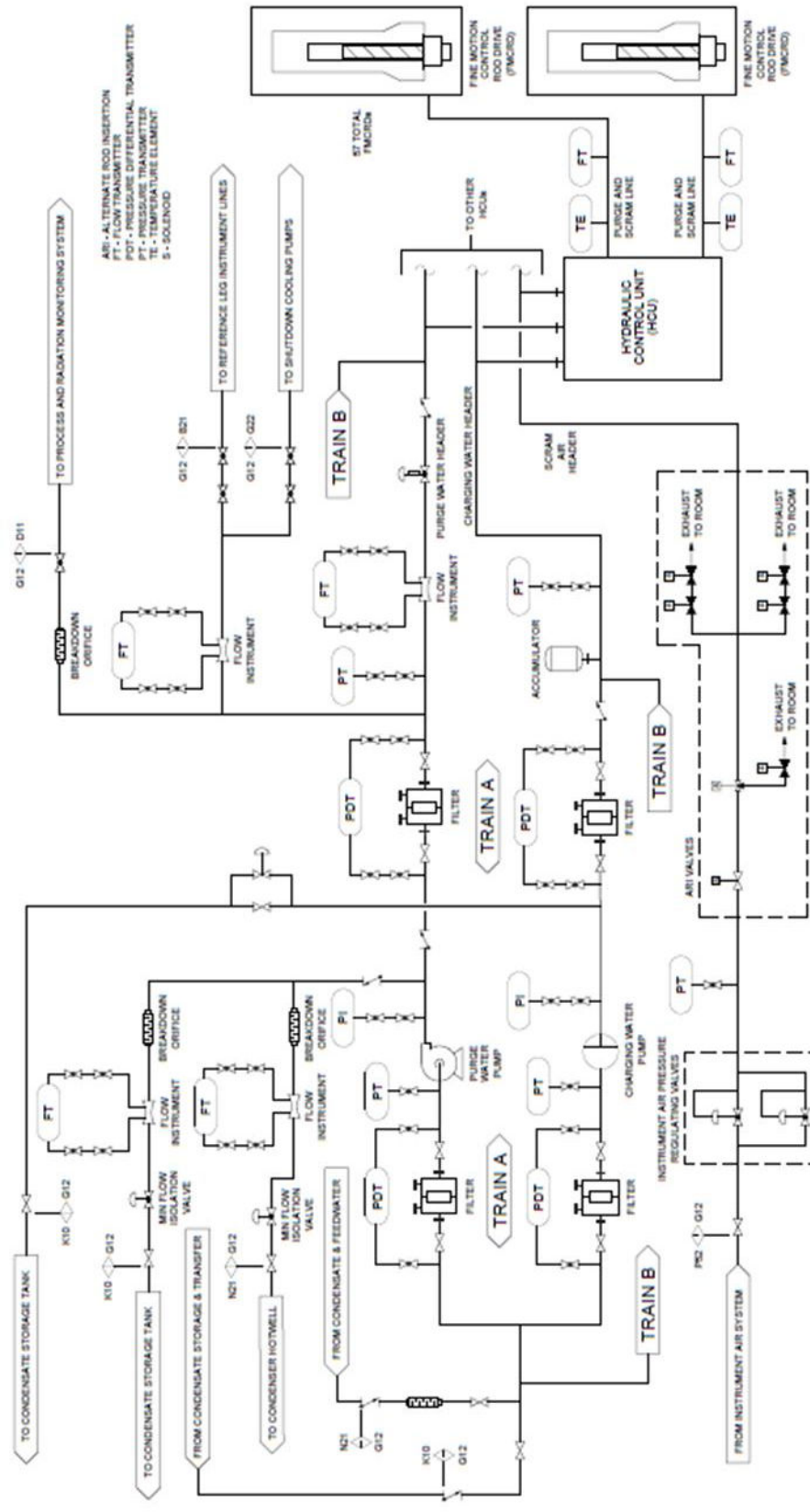
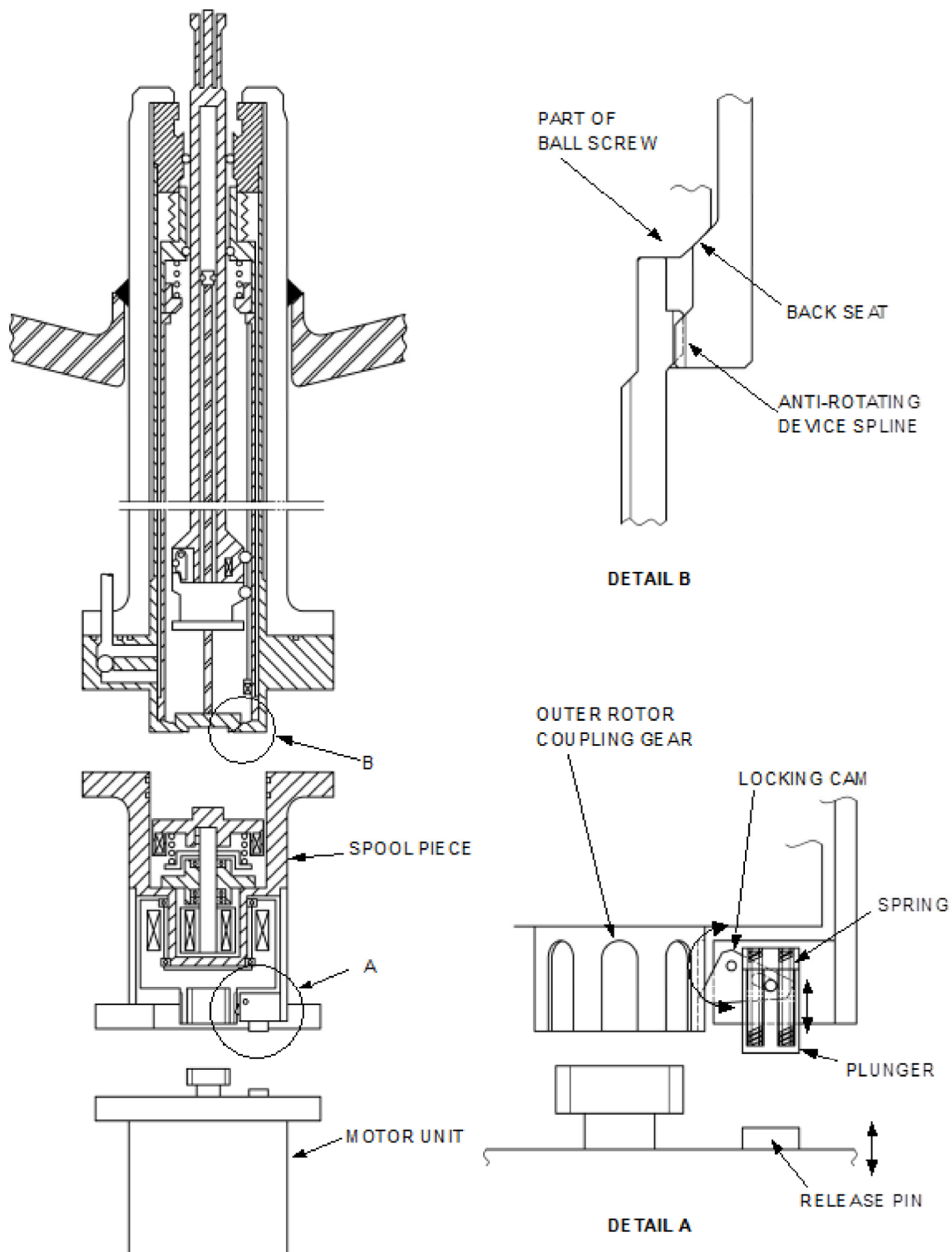


Figure 4.6-10: CRD Simplified P&ID



4.7 Core Monitoring

Core monitoring is a function of the plant computer system that provides three-dimensional core power monitoring. Core monitoring provides confidence that the plant is operating in conformance to SAFDL. Core monitoring obtains instrumentation information from the Distributed Control and Information System (refer to Chapter 7, Subsection 7.3.3.2), calculates power distributions and resulting thermal limits. These power distributions are adapted to signals from plant GTs and LPRMs as applicable.

4.7.1 System and Equipment Functions

The core monitoring function acquires live reactor data from site plant data acquisition systems as required, to define the reactor state for use by the core simulator. The acquired data are validated within acceptable ranges. Core monitoring calculates the accumulated thermal and electrical energy produced by the plant from the beginning of an operating cycle. On a periodic basis, upon user request, or when triggered by a system event, a core simulator determines current core characteristics based on the current reactor state and the previous history of the following core components:

- Thermal limit monitoring
- Fuel conditioning
- GT processing
- Prediction at achievable operating regimes
- Tracking of local and global xenon behavior
- Identification of LPRM drift
- Power adaption based on in-core power measurements
- Graphical display of data
- Power / flow tracking
- BWR operating guidelines implementation
- Isotopic tracking

Core parameters are calculated either by the core simulator or during post processing (including parameters associated with fuel bundles, fuel channels, GTs, LPRMs, and control rods). The core monitoring function then compares these core parameters against technical specifications, licensing limitations, manufacturer's guidelines, and site-specific limits. If user actions are required, the system generates alerts, warnings, and notifications. The primary core monitoring functions are:

- Calculation of current core parameters
- Prediction of future core parameters
- Calculation of LPRM calibration factors
- Calculation and tracking of the isotopic inventory of the fuel
- Generation of visualizations and reports on core performance
- Generates core technical information for use by other plant systems as required

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The core monitoring function uses live reactor and in-core information, and coupled with a three-dimensional BWR simulator model, produces current and predicted core performance information. When the live plant input data, including LPRM, control rod position, and core power are not available, the core monitoring function still performs a core performance calculation using the Manual Monitor option. The performance information includes standard visualizations such as thermal-hydraulic parameters, thermal margins, fuel conditioning margins, fuel burnup, LPRM information, control rod position information and power/flow maps. Information from the core monitoring function is used by the Licenced Plant Operator and the Reactor Engineer in the reactor operational decision-making process, for both steady-state operation and during plant maneuvers, to ensure compliance with licenced limits.

Core monitoring also supports the calibration process for LPRMs by processing GT data and live LPRM data to yield individual LPRM calibration factors. The core monitoring function does not directly calibrate the LPRMs, instead the calibration adjustments require operator action to be implemented in the LPRM systems.

The core monitoring function uses a coupled nuclear thermal-hydraulic diffusion theory model, to provide a three-dimensional simulation of BWR core characteristics performance. The accuracy of the modeling is enhanced by adaptive algorithms that conform results to measured plant data from the LPRMs and GTs. Instrument signals are compared to simulator predictions and outlier data are rejected as anomalous. Unavailability or failure of a limited number of LPRM and GT signals does not significantly affect plant operation.

4.7.1.1 Safety Design Bases

The core monitoring function provides a calculation of core performance parameters, such as the Maximum Fraction Limiting Critical Power Ratio (MFLCPR), the Maximum Fraction of Linear Power Density (MFLPD), and margin to fuel conditioning limits. These values are monitored and used by Reactor Engineers and Licenced Plant Operators to ensure that, during steady-state operation and any reactor maneuvers, these parameters are within the licenced and administrative plant limits.

4.7.1.2 Licencing Basis and Operational Experience

The current core monitoring function embodiment for the operating fleet and the BWRX-300 is named ACUMEN [GESTAR II Rev 23 with ACUMEN SLMCPR AM 42 approval by the U.S. NRC]. ACUMEN has been performing as the official core monitoring function of record at three BWRs since March and April of 2017 and October of 2019, respectively. In all these applications, ACUMEN replaced 3D Monicore and has been built on the experience gained from GNF's legacy core monitoring system which has a rich history and experience of operating at 20+ BWRs for more than 30 years.

The ACUMEN core monitoring and prediction system monitors and predicts fuel performance based on neutron instrumentation inputs and provides reactor engineers and operators with data to optimize the performance of the core. ACUMEN is designed to utilize GNF's 3-D core simulator PANAC11 to compute thermal limits, predict reactor operations using ACUMEN's Predictor feature, utilize failed fuel prevention techniques in Predictor, and monitor for streamlined failed fuel management.

4.7.1.3 Equipment Interfaces

The standard core monitoring function interface provides a means by which to integrate with other systems, including, but not limited to, the plant Distributed Control and Information System. Core monitoring includes a high performance and redundant hardware platform designed to meet the requirements of high availability and redundancy, as well as implement the cyber security requirements of CNSC REGDOC- 2.5.2 (Reference 4.1-1).

4.7.1.4 Core Monitoring Operation

Plant operating conditions that utilize the core monitoring function includes reactor shutdown, ascension to power, steady-state operation, and reactor maneuvering.

In the reactor shutdown mode, all control rods are inserted, the reactor fission power is essentially zero, and the core is subcritical. In this condition, the core monitoring function can perform predictive calculations to confirm required shutdown margin.

At initial start-up of the reactor, the core monitoring function performs predictive calculations of core reactivity for each incremental control rod withdrawal during the approach to initial criticality. High incremental changes in reactivity associated with steps in the control rod withdrawals that would cause a short reactor period can be identified and mitigated. The core monitoring function calculations address the impact of the actual coolant temperature associated with the criticality calculations.

Once the reactor power reaches approximately 5% of the rated thermal power level, the live plant thermal-hydraulic data (temperatures, flow rates, pressure) for the reactor system and the LPRM data become available. The details of the live data and the plant heat balance is monitored by the core monitoring function.

Once the heat balance is stable, the core monitoring function can monitor the core power distribution and margin to operating guidelines and thermal limits (MFLCPR, MFLPD). In addition to providing predictions of the core power distribution, the core monitoring function also tracks the nodal xenon concentration in the core. It is generally not required to monitor thermal margins (e.g., MFLCPR) at very low power levels.

Once the neutronic instrumentation data are judged validated, core monitoring performs a power distribution calculation and adapts the core power distribution calculation to the data. LPRM readings are compared to the GT data, and LPRMs are calibrated to match the full-core normalized GT data.

During power ascension, control rod adjustments are made. The core monitoring function is used during the power ascension to compute core performance so that thermal margins and fuel conditioning limits can be monitored. The Predictor function of core monitoring can be used to analyze and optimize the planned power ascension. The core monitoring and Predictor functions used during plant start-up are also used during plant maneuvers, including load following, periodic maintenance and/or periodic control rod sequence maneuvers during the operating cycle.

4.7.2 References

None.

4.8 Thermal-Hydraulic Stability

4.8.1 Background

Under certain conditions, BWRs can be susceptible to coupled neutronic/thermal-hydraulic instabilities. These instabilities are characterized by periodic power and flow oscillations and are the result of density waves (i.e., regions of highly voided coolant periodically sweeping through the core). If the power and flow oscillations become large enough, and the density waves contain a sufficiently high void fraction, the fuel cladding integrity safety limit could be challenged.

4.8.2 Design Bases

The reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure that SAFDL are not exceeded during any condition of normal operation, including the effects of AOOs.

The reactor core and associated coolant, control, and protection systems are designed to assure that power oscillations that could result in conditions exceeding SAFDL are either not possible or can be reliably and readily detected and suppressed.

4.8.3 Types of BWR Oscillations

There are two types of oscillations associated with BWR stability.

Type 1 instabilities experienced during startup do not result in a reactivity/power response. Type 1 oscillations are characterized by initiation of vapor production in the chimney region leading to a reduction in hydrostatic head in the chimney and a resultant core flow increase, which, in turn, could cause voids to collapse in the chimney. The BWRX-300 reactor goes through an unstable phase during startup. This type of oscillation is unavoidable in a natural circulation reactor because the unstable power/flow region must be crossed prior to establishing a steady two-phase voided region in the chimney; however, the magnitude of the flow oscillations is typically very small. As Type 1 oscillations do not result in a change in core moderator density, there is no power response and, as a result, the Fuel Cladding Integrity Safety Limit is maintained.

Type 2 oscillations are characterized by periodic power and flow oscillations and are the result of density waves (i.e., regions of highly voided coolant periodically sweeping through the core). In Type 2 instability, if the power and flow oscillations become large enough, and the density waves contain a sufficiently high void fraction, the fuel cladding integrity safety limit could be challenged.

Within the Type 2 domain, there are three (3) modes of oscillation:

1. Core-wide (or in-phase) mode oscillations are dominated by the fundamental mode of the neutron dynamics and are detected by the Average Power Range Monitor (APRM) system.
2. Regional (or out-of-phase) mode oscillations, with parallel channel momentum dynamics as the dominant feedback loop and reinforced by a subcritical mode of the neutronics, are not easily detected by the APRM system since the reactor powers in each half of the reactor core along the harmonic axis of oscillations tend to cancel out each other.
3. Single channel thermal-hydraulic mode oscillations are related to the momentum dynamics of heated channel in a two-phase flow regime. The oscillations are purely hydraulic in nature, and with no power response, they are not easily detected by the APRM system.

4.8.4 Design Criteria

The most limiting stability condition in the BWRX-300 normal operating region is at the rated power/flow condition. The BWRX-300 is designed so that the core remains stable throughout the entire operating domain. In the time domain analysis, decay ratio is defined as the ratio of the amplitude of the first two successive peaks. For the BWRX-300, the decay ratio is averaged over the first few peaks.

Conservative design criteria are imposed on the core-wide, regional, and single-channel decay ratios under all conditions of normal operation and anticipated transients. The limiting mode (i.e., highest decay ratio) for thermal-hydraulic instabilities (Type 2 oscillations) is the core-wide mode. The channel and regional modes are highly damped.

4.8.5 Stability Solution Design

The stability licensing criterion for all nuclear power plants is set forth in CNSC REGDOC-2.5.2, Section 8.1 (Reference 4.1-1) requiring assurance that the safety functions prevent unacceptable instabilities during normal operation, AOO conditions, and DBA conditions. An unacceptable oscillation is characterized by an amplitude that grows and may challenge SAFDL. CNSC REGDOC-2.5.2 (Reference 4.1-1) also stipulates that power oscillations that can result in conditions exceeding SAFDL are reliably and readily detected and suppressed. The limiting stability condition in the BWRX-300 during normal power operation region is at the rated power. Therefore, the BWRX-300 is designed so that unacceptable coupled neutronic and thermal-hydraulic power oscillations are not possible throughout the entire operating region. The BWRX-300 is designed to protect acceptable fuel design limits during AOOs. As a Defence-in-Depth feature, the BWRX-300 implements a stability monitoring function to detect instabilities should they occur (Refer to Chapter 7, Subsection 7.3.3) to inform operator action. This solution is similar to the Option I-D long-term stability solution in the U.S. (Reference 4.4-8).

The BWRX-300 design has features that result in stable behavior in normal operation and minimize the effects of potential oscillations in off-normal conditions include:

- Small Core

The small core size and higher inlet orifice pressure drop of the BWRX-300 reduces the likelihood of regional mode instabilities. Conservative analyses are performed to confirm that regional mode oscillations are not possible and that core oscillations would be core-wide dominant. Any unacceptable core-wide oscillation is mitigated by the high flux scram.

 - Tighter neutronic coupling precludes regional mode oscillations
 - Core-wide oscillations are the dominant mode
- Natural circulation
 - No recirculation pump trips that result in significant change from stable to unstable conditions
 - Loss of Feedwater Heating (LFWH) AOO impact on stability is mitigated by SCRR operation for a feedwater temperature reduction of 16.7°C or higher. The SCRR function is described in Chapter 7, Subsection 7.3.3.2. The loss of feedwater heating AOO analysis is described in Chapter 15, Subsection 15.5.3.1.1

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- Tall chimney
 - Increases volume of water
 - Increases driving head and natural circulation flow
 - Dampens oscillations
- Large downcomer area
 - Reduces flow restrictions
- High inlet orifice pressure drop
 - Improves two-phase to single phase pressure drop ratio
- Balanced feedwater temperature
 - Neither thermal margins nor decay ratios are compromised
 - Minimizes normal operation inlet subcooling
- Less subcooling

Compliance with CNSC REGDOC-2.5.2 (Reference 4.1-1) is assured by implementing design criteria for the decay ratio. The stability acceptance criterion for the current GEH fleet of BWRs is a calculated core decay ratio of ≤ 0.80 for normal operation and AOOs considering uncertainties. The BWRX-300 has been established to provide margin to the decay ratio criterion.

4.8.6 Stability Analysis Methods

A detailed discussion of the methods used to analyze BWRX-300 thermal-hydraulic stability is presented in TRACG Application for BWRX-300 (Reference 4.8-1).

TRACG is a GEH proprietary version of the Transient Reactor Analysis Code (TRAC). TRACG uses a multi-dimensional, two-fluid model for the reactor thermal-hydraulics and a three-dimensional reactor kinetics model. The models can be used to accurately simulate a large variety of test and reactor configurations. These features allow for realistic simulation of a wide range of BWR phenomena and are described in detail in the TRACG Model Description Licensing Topical Report (Reference 4.3-3).

TRACG has been extensively qualified against separate effects tests, component performance data, integral system effects tests and operating BWR plant data. The details are presented in the TRACG Qualification Licensing Topical Report (Reference 4.4-7). The USNRC has approved TRACG for application to Economic Simplified Boiling Water Reactor (ESBWR) stability analysis, NEDC-33083P-A (Reference 4.4-6).

TRACG is qualified to accurately model natural circulation for a wide application range that encompasses the BWRX-300. The TRACG qualification bases, NEDE-32177P (Reference 4.4-7) include benchmarking natural circulation flow and instability onset (FRIGG-4 FT-36C Onset of Instability Tests) in which TRACG conservatively predicts the power at the onset of limit cycle oscillations (over a range of system pressures and inlet subcooling) to be reasonably close to measured values.

4.8.7 Stability Evaluation

To determine whether core-wide oscillation is the dominant mode, the flow velocity for symmetric channels on opposite sides of the core are perturbed out of phase. If the core is not susceptible to regional mode oscillations after such a flow velocity perturbation, then symmetric out of phase channels come into phase after a short duration. This analysis confirms that regional mode oscillations are not possible. As such, only the core-wide mode is applicable to the BWRX-300.

4.8.7.1 Stability Performance During NO

Stability analyses are performed at rated conditions and at multiple exposure points during the cycle. Stability analyses are performed by perturbing the system and assessing the core response. Such perturbations might include pressure pulse perturbation or variation in feedwater temperature. The response to a pressure perturbation in the steam line is analyzed to obtain the decay ratio for the BWRX-300 stability analysis.

The resulting decay ratios affirm stable operation, with margin to the stability decay ratio criteria, as described in Chapter 15, Subsection 15.5.2.4.

4.8.7.2 Stability Performance Evaluation During AOOs

In general, the stability margin reduces when the reactor power to flow ratio increases and/or core flow reduces or the core inlet subcooling increases (that also results in power increase). As the BWRX-300 is a natural circulation reactor, recirculation pump trip transients are not applicable and the key state variable that affects decay ratio is the inlet subcooling. As such, the loss of feedwater heating AOO is the limiting transient for stability.

The loss of feedwater heating event increases core inlet subcooling (i.e., the inlet temperature decreases) and increases core power. The BWRX-300 has been established to initiate a SCRR1 in response to a loss of feedwater heating AOO that mitigates the increase in core thermal power. Stability analyses for the LFWH AOO with SCRR1 affirm the stability decay ratio criterion is met and the results are summarized in Chapter 15, Subsection 15.5.2.4.

4.8.8 References

- 4.8-1 NEDC-33987P, "BWRX-300 Darlington New Nuclear Project (DNNP) TRACG Application," GE-Hitachi Nuclear Energy Americas, LLC.



HITACHI

GE Hitachi Nuclear Energy

NEDO-33958

Revision 1

October 7, 2022

Non-Proprietary Information

**Ontario Power Generation Inc.
Darlington New Nuclear Project
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 9A
Auxiliary Systems**

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
0	All	Initial Release
1	9A.6	Adjustments incorporated per customer acceptance review

ACRONYM LIST

Acronym	Explanation
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
AMCA	Air Movement and Control Association
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
ARI	Air Conditioning and Refrigeration Institute
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
BIS	Boron Injection System
BOP	Balance of Plant
BPVC	Boiler and Pressure Vessel Code
CB	Control Building
CCS	Containment Cooling System
CNSC	Canadian Nuclear Safety Commission
CFD	Condensate Filters and Demineralizers System
CFS	Condensate and Feedwater Heating System
CHE	Cranes, Hoists, and Elevators
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CON	Primary Containment System
CRD	Control Rod Drive
CRE	Control Room Envelope
CUW	Reactor Water Cleanup System
CWE	Chilled Water Equipment
CWS	Circulating Water System
D-in-D	Defence-in-Depth
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DCIS	Distributed Control and Information System

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Acronym	Explanation
DEC	Design Extension Condition
DL	Defense Line
DL2	Defense Line 2
DL3	Defense Line 3
DL4a	Defense Line 4a
DL4b	Defense Line 4b
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
EFS	Equipment and Floor Drain System
EFU	Emergency Filter Unit
EHC	Electro-Hydraulic Control
EME	Emergency Mitigating Equipment
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
FAC	Flow Accelerated Corrosion
FCU	Fan Coil Unit
FE	Flow Element
FHA	Fuel Handling Accident
FHA	Fire Hazards Assessment
FPC	Fuel Pool Cooling and Cleanup System
FPP	Fire Protection Program
FPS	Fire Protection System
FT	Flow Transmitter
FW	Feedwater
HCW	High Conductivity Waste
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilation, and Air Conditioning
HVS	Heating Ventilation and Cooling System
HX	Heat Exchanger
I&C	Instrumentation and Control
IC	Isolation Condenser
ICC	ICS Pool Cooling and Cleanup System

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Acronym	Explanation
ICS	Isolation Condenser System
IGSCC	Intergranular Stress Corrosion Cracking
INPO	Institute of Nuclear Power Operations
ISFSI	Independent Spent Fuel Storage Installation
ISI	In-Service Inspection
IST	In-Service Testing
LED	Light Emitting Diode
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LWM	Liquid Waste Management System
MCA	Main Condenser and Auxiliaries
MCR	Main Control Room
MSR	Moisture Separator Reheater System
MSRIV	Main Steam Reactor Isolation Valves
MTE	Main Turbine Equipment
NBCC	National Building Code of Canada
NBS	Nuclear Boiler System
NFPA	National Fire Protection Association
NHS	Normal Heat Sink
NPSH	Net Positive Suction Head
OGS	Offgas System
OLNC	On-Line NobleChem™
PAM	Post-Accident Monitoring
PAS	Plant Automation System
PCW	Plant Cooling Water System
PLSA	Plant Services Area
PPS	Plant Pneumatics System
PREMS	Process Radiation and Environmental Monitoring System
RB	Reactor Building
RBS	Reactor Building Structure
RCPB	Reactor Coolant Pressure Boundary
RES	Refueling and Servicing Equipment System

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Acronym	Explanation
RFP	Reactor Feed Pump
RLC	Reactor Level Control
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SBO	Station Blackout
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCCV	Steel-plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SJAE	Steam Jet Air Ejector
SSC	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SWM	Solid Waste Management System
TB	Turbine Building
TBS	Turbine Building Structure
TMR	Triple Modular Redundant
TS	Technical Specifications
UPS	Uninterruptible Power Supply
WGC	Water, Gas and Chemical Pads

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9A.0 AUXILIARY SYSTEMS

9A.1 Fuel Storage and Handling Systems

9A.1.1 New Fuel Storage and Handling System

New fuel receiving is a service and a material handling process that utilizes the Fuel Handling System and equipment described in Subsections 9A.1.2, 9A.1.4 and 9A.8. New fuel receipt is described in Section 5.1 of the Safeguards Annex. The BWRX-300 design does not employ a new fuel storage vault. The new fuel assemblies are stored in fuel storage racks which are located in the Fuel Pool in the Reactor Building. New fuel and spent fuel are stored in the same Fuel Pool. The following information is provided in support of the requirements of CNSC REGDOC-2.5.2 Section 8.12.1 (Reference 9A.1.1-1) relative to new fuel storage and handling:

1. Nuclear criticality safety, refer to Subsection 9A.1.2.3.1
2. Maintenance, periodic inspection, and testing of components important to safety, refer to Subsection 9A.1.2.8 and Section 5 of the Safeguards Annex
3. Inspection of non-irradiated fuel, refer to the Safeguards Annex, Section 5.1.2 and Chapter 4, Subsection 4.2.5.
4. Prevent loss of or damage to the fuel, refer to Subsection 9A.1.2.3.13 and Safeguards Annex Section 7 for information pertaining to prevention of loss of fuel. Subsection 9A.1.4 addresses fuel handling equipment and Subsection 9A.8 addresses movement of heavy loads in the vicinity of the fuel pool
5. Meet Canada's safeguards requirements for recording and reporting accountancy data, and for monitoring flows and inventories related to non-irradiated fuel containing fissile material, refer to Safeguards Annex, Section 4 "Safeguards and Operation" and Section 7 "Nuclear Material Accountancy"

9A.1.1.1 References

- 9A.1.1-1 Canadian Nuclear Safety Commission REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants"

9A.1.2 Fuel Storage and Handling System

The fuel pool is used to store spent fuel after discharging and new fuel after delivery to the site and before core loading. The above grade portion of the Reactor Building (RB) structure houses the refueling floor, refueling and fuel handling systems, fuel pool, and Reactor Building Polar Crane.

9A.1.2.1 System and Equipment Functions

System and equipment functions associated with the Fuel Storage and Handling System include the followings:

Normal Functions (Non-Safety-Category)

- Permit disassembly and re-assembly of containment and reactor.
- Permit reactor refueling.
- Permit new and spent fuel storage in fuel pool.
- Permit replacement of reactor mechanical components.

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- Permit replacement of nuclear instrumentation.
- Support various outage maintenance.
- Maintain subcriticality during fuel and core component handling operations.

Normal Functions (Safety-Category)

- Fuel Storage and Handling System does not perform any Safety Category functions during off-normal conditions.

Off-Normal Functions (Non-Safety-Category)

- Fuel Storage and Handling System does not perform any Non-Safety-Category functions during off-normal conditions.

Off-Normal Functions (Safety-Category)

- Maintain Fuel Pool structural integrity following a seismic event
- Fuel Storage and Handling System does not perform any Safety-Category functions during off-normal conditions.

9A.1.2.2 Safety Design Bases

The fuel pool is adequately protected against dynamic effects. The fuel pool is designed to ensure that required water levels are maintained in normal and off-normal conditions. Fuel in the fuel pool is protected from damage caused by the postulated drop of fuel assemblies, bundles, or other objects onto stored fuel (Subsections 9A.1.4 and 9A.8) by assessing the impact of the load drop and designing systems to mitigate the effects.

The orientation and location of the turbine shafts are designed such that the fuel storage components are located outside the turbine missile low-trajectory strike zone (Chapter 10, Subsection 10.2.3). Furthermore, the turbine is located in the Turbine Building with Turbine Building and Reactor Building walls between the Fuel Pool and Turbine.

The fuel pool has the following design features: control for airborne release of radioactive material; drains, gates, and weirs that prevents drainage of coolant inventory below an adequate shielding depth; adequate coolant flow provided to the fuel racks; and a system to detect fuel pool leakage.

The addition of water to the fuel pool does not result in a criticality event.

A full array in the loaded fuel rack is designed to be subcritical by at least 5% Δk under all normal and abnormal conditions as per CNSC REGDOC-2.4.3 Section 2.3 (Reference 9A.1.2-1). Safety analyses are performed using validated codes (Chapter 3, Appendix 3E).

Fuel pool storage design meets Canadian Nuclear Safety Commission (CNSC) requirements specified in CNSC REGDOC-2.5.2 Section 8.12 (Reference 9A.1.2-2) as related to ensuring the design and handling of irradiated fuel.

9A.1.2.3 Description of System

The BWRX-300 Steel-plate Composite Containment Vessel (SCCV) and the main RB structure including the reactor cavity, equipment pool and fuel pool are classified as Safety Class 1 (SC1).

The fuel pool base mat is located at approximately the power block grade elevation. The fuel pool extends up to the refuel floor elevation of approximately 13 metres above grade elevation.

The fuel pool size is determined by the volume of cooling water, size of the fuel storage racks, with control rod blade guides, other equipment required to be stored in the pool, and lay-down areas. Refueling outage equipment such as lights, test weights, dummy fuel bundles, and control blades are also stored in the fuel pool. The fuel preparation machine is located on the fuel pool periphery. The new fuel inspection stand is mounted on the refuel floor away from the fuel pool periphery and only used for new fuel before it is installed into the water and to the fuel prep machine. Used reactor core instruments may also be stored in the fuel pool, until such time as they are cut up and packaged.

Additional information pertaining to the design of the new and spent fuel storage system is presented below.

9A.1.2.3.1 Nuclear Criticality Safety

Chapter 15, Subsection 15.5.9.3 analyzes the dose consequence of a representative Out-of-Core-Criticality accident scenario compliant with the requirements of CNSC REGDOC-2.4.3 (Reference 9A.1.2-1).

New and spent fuel are stored in storage racks capable of maintaining fuel subcritical. The fuel pool is comprised of a deep pit filled with water which provides shielding in addition to the shielding provided by the pool confinement structure. Within the pool are storage racks in a grid pattern which contain a fixed neutron absorbing material. The storage rack geometry and material are designed to preclude the possibility of criticality under normal and credible abnormal conditions.

A full array in the loaded fuel rack is designed to be subcritical by at least 5% Δk . Monte Carlo techniques are employed in the calculations performed to assure that k_{eff} does not exceed 0.95 under all normal and abnormal conditions.

9A.1.2.3.2 Heat Removal in Operational States, Design Basis Accidents and Design Extension Conditions

Removal of heat from the fuel pool during operational and design basis accident conditions is described in Subsection 9A.1.3. The design of the Fuel Pool contains provisions for Fuel Pool Design Extension Conditions (DECs) by:

1. Ensuring that boiling in the pool does not result in structural damage (Chapter 3, Subsection 3.5.6.1)
2. Providing temporary connections to enable the refill of the pool using temporary supplies (Subsection 9A.1.3.6.4)
3. Providing temporary connections to heat removal systems for power and cooling water (Subsection 9A.1.3.3 discusses redundant cooling trains)
4. Ensuring that the design of the fuel pool is such that a Fuel Handling Accident does not exceed site Design Basis Accident dose acceptance criteria Chapter 15, Subsection 15.5.8
5. Ensuring that severe accident management actions related to the fuel pool can be carried out (Chapter 15, Subsection 15.1.5)

9A.1.2.3.3 Inspection of Irradiated Fuel

The handling and storage systems are used primarily to support refueling and core shuffling, and the same equipment supports periodic inspections as needed of irradiated fuel. The typical inspection scenario entails the Refueling Platform (Subsection 9A.1.2.3.5) moving fuel to the Fuel Preparation Machine. The Fuel Preparation Machine is typically outfitted with a fuel inspection fixture which is a small rotator installed onto the Fuel Preparation Machine to facilitate fuel inspections. In addition to supporting the loading of new fuel into the fuel pool during outage preparation, the Fuel Preparation Machine supports activities associated with the inspection of irradiated fuel, including removal and reinstallation of the fuel channel to enable access to the bundle/rods. Typically located adjacent to the Fuel Preparation Machine there is a channel handling boom, and below the surface of the fuel pool on the wall, a channel storage rack. When irradiated fuel is moved into the Fuel Preparation Machine for inspection, procedural requirements specify the water coverage that must be maintained over the top of the irradiated fuel. Typically, a mechanical interlock on the drive of the Fuel Preparation Machine is utilized to limit or block raising the bundle above an approved radiologically safe elevation. Plant equipment utilized for servicing irradiated fuel includes the following.

Fuel Preparation Machine

Two Fuel Preparation Machines are provided for handling fuel assemblies while removing or installing channels on the fuel bundles. Each machine consists of a fuel bundle carriage, which rides on a frame that is mounted on the edge of the fuel pool and extends down to depth along the pool wall and a work platform. The work platform includes handrails and extends out over the pool to facilitate viewing of the channeling operations.

Fuel Inspection Fixture

The Fuel Inspection Fixture is used in conjunction with the Fuel Preparation Machine to permit remote inspection of fuel elements. The Fuel Inspection Fixture, when installed, permits rotation of the fuel assembly in the carriage, and in conjunction with the vertical movement of the carriage provides complete access to all quadrants of fuel assembly inspection.

New Fuel Inspection Stand

The New Fuel Inspection Stand is stored on the refueling floor of the reactor building. It consists of a single "U" shaped platform with guardrails, a column which supports two fuel assemblies in a vertical position, and a personnel lift which raises or lowers the work platform along the fuel assemblies for inspection purposes. The lower base of the fuel assembly sits in the fuel seat at the bottom of the stand. The fuel assemblies are mounted on rotatable bearing surfaces which allow for rotation to desired quadrants for inspection.

Channel Handling Tool

The Channel Handling Tool is used in conjunction with the Fuel Preparation Machine to remove, install, and handle fuel channels in the fuel storage pool. The bail hangs from a load balancer on the channel handling boom mounted adjacent to the Fuel Prep Machines.

Channel Transfer Grapple

The Channel Transfer Grapple is an air actuated device consisting of a frame, air cylinder, and two jaws. It is used with the Refueling Platform auxiliary hoist to transport individual irradiated fuel channels between working and storage facilities in the fuel pool.

Channel Bolt Wrench

The Channel Bolt Wrench is a manually operated device used for removing and installing the Channel Fastener.

Channel Handling Boom

The Channel Handling Boom is located in a socket adjacent to the fuel preparation machines and adjacent to the fuel pool. The boom supports the channel handling tool and spring balancer over the Fuel Preparation Machine during channel removal and installation operations.

Inspection Camera

Cameras and tooling for inspections and measurements are curb-mounted next to the Fuel Preparation Machine in support of underwater inspection activities. Assorted hand tools (handling poles) and grapples are used in inspections. Sometimes, inspections are conducted during outages, other times, inspections on discharged fuel occurs post-outage (during the operating cycle).

9A.1.2.3.4 Periodic Inspection and Testing of Components Important to Safety

For information pertaining to periodic inspection and testing of components refer to Subsection 9A.1.2.8.

9A.1.2.3.5 Design for Precluding the Dropping of Irradiated Fuel in Transit

The Reactor Building is supplied with a Refueling Platform for fuel movement and reactor servicing support tasks. Fuel and other components are removed from the reactor core, transported to the fuel pool, and then returned to the reactor as required. Additionally, the Refueling Platform serves as a service area which is used to assist in normal vessel disassembly/reassembly, normal Reactor Pressure Vessel maintenance activities, and in performance of long-term in-vessel inspection and maintenance activities.

The Refueling Platform consists of a bridge which spans the width of the Fuel Pool and Reactor Cavity. A trolley rides on the bridge, traversing the width of the bridge to the maximum width. A full length sized working platform where refueling, servicing, and inspection personnel perform their work support tasks spans the length of the bridge. The Refueling Platform rides on four wheels running on two rails set into the refuel floor.

The Refueling Platform is equipped with a traversing trolley, an operator control console, a main hoist with a telescoping mast and fuel grapple, an auxiliary, and a monorail hoist. An air compressor on the Refueling Platform provides compressed air to the pneumatic system for the fuel grapple and reactor service tooling. A frame mounted hoist and a monorail hoist support handling the smaller core components and tooling.

The Refueling Platform is a rigid structure built to ensure accurate and repeatable positioning during the refueling process. The telescoping mast and grapple are suspended from a trolley system and is used to lift and orient fuel bundles for placement in the core or storage racks. The Refueling Platform includes a control console providing the operator precise control of the motions of bridge, trolley, and all hoists. Bridge, trolley, and mast grapple elevation readouts are located in a clearly understandable and convenient position for the operators to monitor as they position the Refueling platform. The control console includes a light indication to monitor hoist functions and refueling interlocks.

The main hoist and fuel grapple has a redundant load path so that no single component failure results in a fuel bundle drop. Specialized grapple tooling which carries fuel assemblies or control rod blades is designed such that accidental opening is mechanically prevented and, upon loss of power, the grapples fail in the closed position.

Interlocks on the Refueling Platform prohibit bridge movement towards the core when the reactor mode switch is in refueling position and a single control rod is out.

Power shuts off to the main fuel hoist motor when the Refueling Platform is over the core, the hoist load cell is sensing load of a fuel assembly or greater, and the reactor mode switch is in refueling and a single rod is withdrawn.

Hoist motion is prevented when the grapple hooks are not closed, and the fuel hoist senses load of a fuel assembly or greater.

Upon completion of fuel movement operations, a core verification task is accomplished in which individual fuel assemblies are verified to be installed in the correct position using serial numbers assigned to the fuel assemblies. Refer to Subsection 9A.1.4 for information pertaining to the design of equipment and operations associated with precluding dropping of irradiated fuel while in transit.

Spent fuel is transferred from the fuel pool in a Transfer Cask using the single failure proof RB Polar Crane. Refer to Subsection 9A.8.1 for information pertaining to the design and operation of the RB Polar Crane as it relates to precluding the dropping of irradiated fuel while the in transit.

9A.1.2.3.6 Handling Stresses on Fuel Elements or Fuel Assemblies

The structural adequacy of the fuel assembly components is demonstrated by evaluations (analysis or testing) that specifically address the operational duty that results from the BWRX-300 environment. This duty results from steady-state operation (including handling loads), mechanical loads associated with anticipated transients, and accident loads due to external conditions.

The fuel assembly structural components are evaluated to ensure that the components do not fail due to stresses exceeding the fuel assembly component mechanical capability.

Upper Tie Plate

The design loading for the Upper Tie Plate is from bundle handling. Specifically, a load equal to three times the bundle weight is applied at the tie plate handle to grapple attachment. The load is reacted at the eight tie-rod locations.

Lower Tie Plate

The design loading for the Lower Tie Plate is from bundle handling.

Fuel Rod End Plugs

The design loading for the Fuel Rod End Plugs is from bundle handling.

Plenum Spring

The Plenum Spring is designed to resist an acceleration of the fuel pellet column while being transported without deflecting the spring.

Expansion Spring

The Expansion Springs are designed to resist downward forces from grappling and the weight of the suspended components such as the tie plate while allowing expansion from irradiation growth of the individual fuel rods.

Water Rods

The Water Rod tubing was evaluated for a steady-state differential wall pressure. The maximum load that a Water Rod tab could experience due to operating effects of spacer lift forces from flow or differential thermal expansion between the fuel rods and Water Rods is the load required to simultaneously slide all fuel rods through a spacer.

Spacer

Tests are performed to demonstrate that the Spacer design can withstand design basis loading without any significant deformation.

Channel

The design loadings for the Channel include steady-state and transient operating differential pressure. The Channel is tested to demonstrate capability to withstand bending loads from lateral seismic loading.

9A.1.2.3.7 Inadvertent Dropping of Heavy Objects and Equipment on Fuel Assemblies

Refer to Subsection 9A.8.1 and Chapter 3, Subsection 3.4.4.3, for information pertaining to the inadvertent dropping of heavy objects and equipment on fuel assemblies.

9A.1.2.3.8 Inspection and Safe Storage of Suspect or Damaged Fuel Elements or Fuel Assemblies

Although rare, mechanical damage to a fuel assembly can occur due to fuel rod cladding wear from fretting or from a fuel handling accident. In the unlikely event that a Fuel Handling Accident (FHA) does occur, visual inspections are performed to assess the extent of damage followed by an action plan. The damaged assembly is recovered under a special written procedure lifting from the normal bail handle and stored in a fuel storage rack.

Failed fuel requires special consideration because of its significance during operations. Fuel failures are first detected via the plant's Offgas System (Chapter 11, Subsection 11.3). In addition, grab samples for measuring individual isotopic activities with spectrometry equipment is typically performed.

Typically, six major nuclides are measured and the sum of these is reported as the total offgas release rate. Xenon-133 is by far the most important isotope among these for determining the presence of fuel failures. Virtually all "indicators" or calculated parameters for fuel integrity monitoring rely on the relative increase in Xe-133 inventory to help detect leaking spent fuel. Whenever the measured isotopic mixture becomes relatively rich in Xe-133 it indicates a cladding perforation and release of stored gas. Fission product activity in the reactor coolant is monitored separately (e.g., iodine, strontium, and cesium isotopes) and typically only shows an increase for more significant failures.

9A.1.2.3.9 Radiation Protection

Radiation protection complies with the CNSC guidance (Reference 9A.1.2-3) for normal operation and anticipated operational occurrences. Radiation exposure is kept within regulatory limits and As Low As Reasonably Achievable (ALARA) (Chapter 12, Subsection 12.3.1) goals.

Radiation monitoring is provided for the fuel pool storage area and the associated ventilation paths (Chapter 11, Subsection 11.5.3). The Refueling Platform, hoist, and grapple include interlocks to prevent potential refueling errors and reduce the possibility of exposure of plant workers. The Refueling Platform telescoping mast design provides a rigid mechanical stop that makes it physically impossible to raise fuel above the safe level below the water.

The fuel pool is designed to ensure the area dose rate is less than 25 $\mu\text{Sv/hr}$. Fuel handling equipment design ensures that the operator is not exposed to 25 $\mu\text{Sv/hr}$ when fuel and irradiated components are being stored or handled in accordance with ANSI 57.1, Design Requirements for Light Water Reactor Fuel Handling System (Reference 9A.1.2-5). The BWRX-300 design does not have integrated containment shielding in the fuel transfer zone from the core to fuel pool and over the top of containment. This creates the need for a temporary and removable radiation shielding transfer canal or cattle chute that shields personnel in containment from fuel or irradiated components passing through the fuel transfer zone to the fuel pool. The water coverage over spent fuel assemblies in transit to the fuel pool or core is influenced by the grapple normal up limit and the fuel pool normal water level.

The air above the fuel pool and refuel floor equipment areas is monitored. In the event of a Fuel Handling Accident in the Fuel Handling Machine area, a radiation monitor alarm initiates closure of the RB isolation dampers and securing of the RB upper level supply Air Handling Units.

9A.1.2.3.10 Identification of Individual Fuel Bundles

Every fuel bundle contains a serial number on top of the bail handle which is part of the upper tie plate. Not only are these serial numbers used to specify each bundle's location in the core for each cycle but ultimately to track every bundle from manufacturing to ultimate disposal. All special

nuclear material accountability rules are strictly followed, and records are retained for every movement of a bundle over the bundle's life. Refer to Safeguards Annex, Section 7 for information pertaining to nuclear material accountancy.

9A.1.2.3.11 Maintenance and Decommissioning of Fuel Storage and Handling Facilities

Proper maintenance and operation of Fuel Pool systems is necessary to maintain water quality and radionuclides at acceptable levels. Maintenance of water quality is necessary to prevent degradation of the spent fuel and other materials stored in the fuel pool (i.e., control rod blades or incore instrument strings). Fuel pool water treatment and system maintenance programs prevent the buildup of excessive concentrations of contaminants and radionuclides and mitigate the consequences of any potential release from the fuel pool (Subsection 9A.1.3).

Scheduled maintenance ensures that structures and systems required for containing, cooling, cleaning, level monitoring and makeup of water in the fuel pool are operable consistent with the licensing basis. The application of scheduled maintenance reduces high levels of contaminants and radionuclides in the pool water that can have adverse effects on stored fuel, the fuel pool, fuel transfer components, and related equipment.

Operating procedures coupled with surveillances and observations, indicate changes in fuel pool level. Procedures address appropriate maintenance, calibration, and surveillance of available monitoring equipment.

Plant decommissioning and dismantling activities considered at the design phase include considerations of experience gained from the decommissioning of existing plants, as well as those plants that are in long-term safe storage. Refer to Chapter 21 for information related to decommissioning of the BWRX-300.

9A.1.2.3.12 Decontamination of Fuel Handling and Storage Areas and Equipment

Decontamination of fuel handling and storage areas and equipment is performed as required to comply with ALARA requirements as discussed in Chapter 12 and prevent the spread of contamination.

New and spent fuel is stored in the fuel pool. As part of typical operating practice, area's outside of the fuel pool where fuel is handled are surveyed for contamination. The plant is supplied with equipment and features to accomplish effective decontamination without spreading contamination. Wash-down areas and sink drains are routed to the liquid radioactive waste system (Chapter 11, Subsection 11.2). The Reactor Building Ventilation System (Subsection 9A.5.1) maintains negative pressurization of potentially contaminated areas to control leakage of potentially radioactive effluent to the atmosphere. Vendor-supplied services may also be utilized for handling and storage area decontamination.

In the event radiation surveys of tools indicate additional decontamination is required the tools may be sent to a decontamination room for processing.

Design considerations for environment and safety are imposed upon tooling to facilitate operator safety and to ensure ALARA guidance and best practices are incorporated into the design process. Typical design considerations for the decontamination of fuel handling and storage areas and equipment includes:

1. Ease of decontamination
2. Elimination of crevices which facilitates crud removal

3. Minimization of crud buildup which facilitates decontamination and cleaning of equipment, components, and contaminated areas, and helps to prevent airborne contamination dispersion
4. Use of the best construction materials (smooth surface)
5. Use of HEPA filters as required to support cleaning of equipment post use

Criteria for selecting tools, materials, and equipment for use in contaminated areas includes minimizing the use of porous or other materials that are difficult to decontaminate.

9A.1.2.3.13 Implementation of Adequate Operating and Accounting Procedures to Prevent Loss of Fuel

Refer to Safeguards Annex, Section 7, for information related to operating and accounting procedures to prevent loss of fuel.

CNSC regulatory document REGDOC-2.13.1 (Reference 9A.1.2-4), on safeguards and nuclear material accountancy, sets out CNSC's requirements and guidance for the:

- Establishment and maintenance of a safeguards program
- Requirements for event and compliance monitoring reporting by licencees

OPG compliance with CNSC REGDOC-2.13.1 is discussed in the "Safeguards Annex."

9A.1.2.3.14 Measures to Prevent a Direct Threat or Sabotage to Irradiated Fuel

Information pertaining to the measures to prevent a direct threat or sabotage to irradiated fuel is considered Protected Information and withheld from the public.

9A.1.2.3.15 Safeguards Requirements for Recording and Reporting Accountancy Data, and for Monitoring Flows and Inventories Related to Irradiated Fuel Containing Fissile Material

Refer to the Safeguards Annex, Section 4 "Safeguards and Operation" and Section 7 "Nuclear Material Accountancy" for information pertaining to safeguards requirements for recording and reporting accountancy data, and for monitoring flows and inventories related to irradiated fuel containing fissile material.

The fuel pool which is used for fuel storage includes provisions as discussed in the noted sections:

1. Controlling the chemistry and activity of any water in which irradiated fuel is handled or stored (Subsection 9A.1.3, Table 9A.1.3-1)
2. Monitoring and controlling the water level in the fuel storage pool (Subsection 9A.1.3)
3. Detecting Leakage (Subsection 9A.1.2.3.16)
4. Preventing the pool from emptying in the event of a pipe break (Subsection 9A.1.2.2)
5. Sufficient space to accommodate the entire reactor core inventory at all times (Subsection 9A.1.2.3.16)

9A.1.2.3.16 Component Description

Storage Racks

The fuel racks consist of a stainless steel structure composed of neutron absorbing material in a series of square vertical tubes (cells).

The fuel storage racks are top entry racks with bail extended above the rack and designed to preclude the possibility of criticality.

The fuel assemblies are stored in the fuel storage racks that are arranged in the fuel pool as shown in Figure 9A.1.2-1. The fuel storage racks provide space for storage of fresh fuel as well as space for longer term storage of the spent fuel assemblies for cooling prior to transfer to onsite storage or off-site shipment.

The BWRX-300 design includes storage for approximately 600 bundles. The fuel racks can hold an entire reload of fresh fuel and up to approximately eight years of spent fuel and have space remaining to accept a full core of off-loaded fuel.

Liner

The pool liner is designed to meet Seismic Category B requirements. The fuel pool liner is capable of withstanding all design loads as discussed in Chapter 9B. Under certain conditions the fuel pool boils to provide cooling of the spent fuel. The design of the fuel pool liner is capable of withstanding the high temperatures associated with a boiling pool. A leak detection system is provided to detect fuel pool leaks.

For normal conditions of operation, the spent fuel assemblies are cooled by the Fuel Pool Cooling and Cleanup System (FPC). In the event of a loss of the fuel pool cooling and cleanup functions (such as station blackout) the primary defense line maintaining fuel pool temperature is boiling of the fuel pool. Refer to Subsection 9A.1.3 for a description of the FPC.

Refueling Platform

The refueling platform is a gantry-type crane that spans the reactor vessel cavity and fuel pool and is employed to handle fuel and perform other ancillary tasks in the reactor building. It is equipped with a traversing trolley on which is mounted a telescoping mast and a fuel grapple. The refueling platform is a rigid structure built to precise engineering standards to ensure accurate and repeatable positioning during the refueling process. Additional information pertaining to the refueling platform is provided in Subsection 9A.1.2.3.5.

9A.1.2.4 Materials

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials through material chemistry, heat treatment, contamination, and material processes controls.

9A.1.2.5 Interfaces with Other Equipment or Systems

Interfaces associated with the Fuel Pool include those identified in Subsection 9A.1.3.5 for the Fuel Pool Cooling and Cleanup System as well as the Fuel Handling Systems for Cask Loading, (Subsection 9.1.4), Reactor Building Polar Crane (Section 9A.8), and RB HVAC (Subsection 9A.5.1).

9A.1.2.6 Systems and Equipment Operation

Refer to Subsection 9A.1.3 for information related to the operation of the Fuel Pool Cooling and Cleanup System. Refer to Subsection 9A.1.4 for information pertaining to fuel handling operations, Subsection 9A.5.1 for information pertaining to Heating, Ventilation and Cooling System, and Section 9A.8 for information pertaining to the operation of cranes, hoists, and elevators.

9A.1.2.7 Instrumentation and Control

Subsection 9A.1.3 "Fuel Pool Cooling and Cleanup" describes the fuel pool water temperature and water level instrumentation. Chapter 12, Section 12.3 describes design features for radiation protection. Subsection 9A.5.1.8 describes the Heating, Ventilation and Cooling System instrumentation.

9A.1.2.8 Monitoring, Inspection, Testing, and Maintenance

The design of the fuel pool liner and fuel storage racks facilitate inspections of the exposed surface of the liner and fuel storage rack.

Neutron absorbing material coupons are provided with the spent fuel racks for the purpose of monitoring for potential degradation of the material over the design life of the equipment.

Refer to Chapter 3, Subsection 3.5.5.6 for information pertaining to testing and in-service inspection requirements related to the RB. The design of the fuel pool provides monitoring for the loss of decay heat removal capability using the temperature measuring instruments in the Fuel Pool Cooling and Cleanup System as described in Subsection 9A.1.3. Radiation monitors are provided in the Fuel Pool area to detect both general area radiation levels and airborne contamination levels as described in Chapter 12, Subsection 12.3.5.4.

9A.1.2.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

The design of the fuel pool minimizes buildup of contamination and provides shielding. The surface finishes of the components for the fuel storage racks are smooth to minimize accumulation of radioactive materials and to facilitate surface decontamination. The depth of the water above the fuel assemblies and the thick concrete walls of the fuel pool provides shielding for the assemblies.

9A.1.2.10 Performance and Safety Evaluation

The fuel storage racks, and fuel pool (including liner) are located inside of the Reactor Building. The design of the Reactor Building withstands combinations of mechanical, hydraulic, and thermal loads and natural phenomena effects, including severe winds such as hurricanes and tornadoes (Chapter 3, Subsection 3.3.2), floods (Chapter 3, Subsection 3.3.3), external and turbine-generated missiles (Chapter 3, Subsection 3.3.5 and Chapter 10 Subsection 10.2.3), and earthquakes (Chapter 3, Subsections 3.3.1). The Reactor Building protects the fuel pool, liner, and fuel storage racks from these hazards.

The design of the fuel storage racks is such that K_{eff} remains less than or equal to 0.95 under design basis conditions, including fuel handling accidents. Drop of a fuel assembly onto fuel assemblies stored in the fuel pool is discussed in Chapter 15, Subsection 15.5.8. Handling equipment capable of carrying loads heavier than fuel components are prevented by design and administrative controls from carrying loads over the fuel pool (Subsection 9A.8).

9A.1.2.11 References

- 9A.1.2-1 CNSC Regulatory Document REGDOC-2.4.3, "Nuclear Criticality Safety."
- 9A.1.2-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.1.2-3 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."

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- 9A.1.2-4 CNSC Regulatory Document REGDOC-2.13.1, "Safeguards and Nuclear Material Accountancy."
- 9A.1.2-5 ANSI 57.1, "Design Requirements for Light Water Reactor Fuel Handling System," American National Standards Institute.

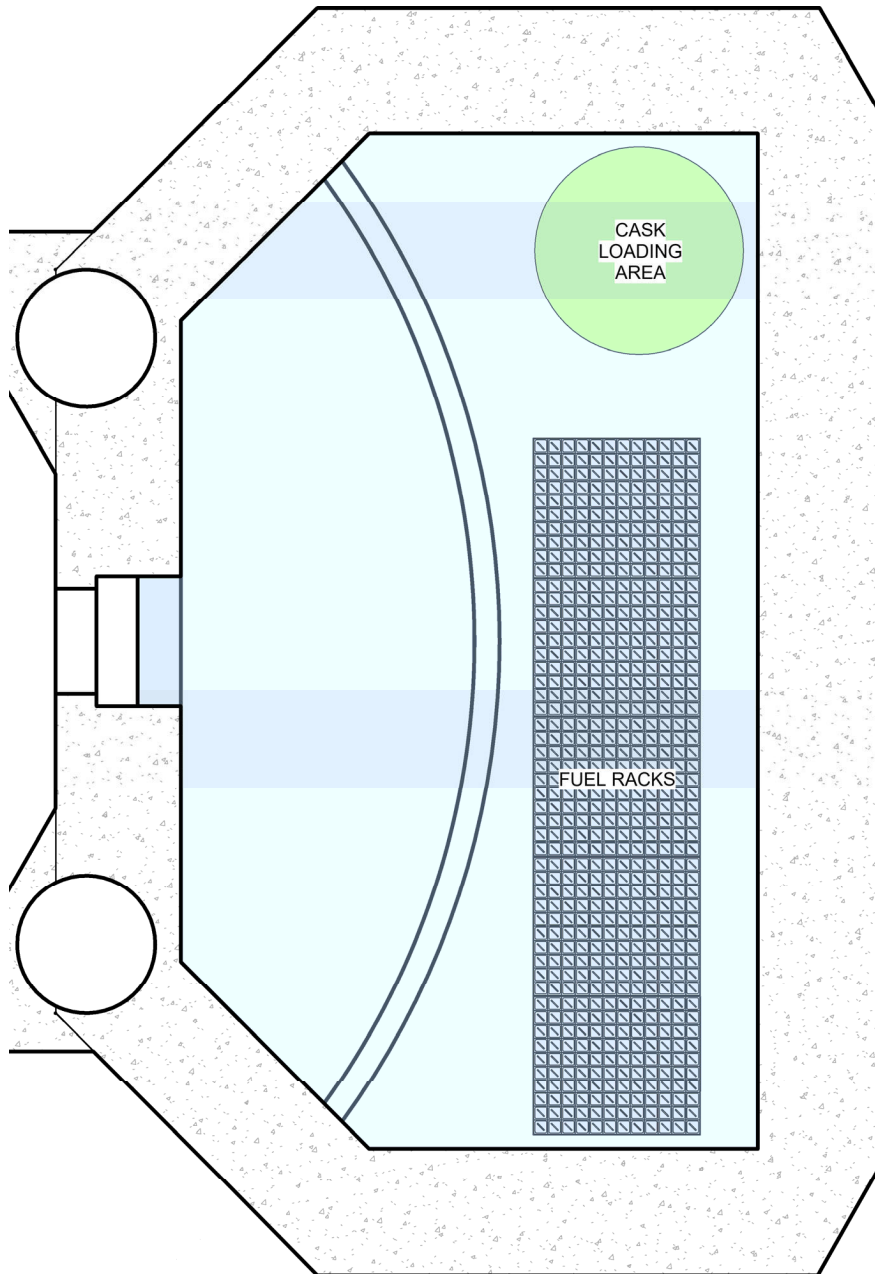


Figure 9A.1.2-1: Fuel Pool Arrangement

9A.1.3 Fuel Pool Cooling and Cleanup System

The primary function of the FPC is to provide continuous cooling of the water volume in the fuel pool to remove decay energy from spent fuel, and to provide replacement coolant inventory from a variety of sources, both to ensure spent fuel is kept cool and submerged throughout the life of the plant. In addition, FPC includes demineralization and particulate filtration to maintain coolant quality and to reduce general area dose. FPC can be realigned to provide cooling and cleanup to the reactor cavity and equipment pools as necessary.

The FPC system is generally classified as Defense Line 2, Safety Class 3 (DL2/SC3) system.

The portion of the system that provides makeup capacity to the pool is classified as Defense Line 4b, Safety Class 3 system.

9A.1.3.1 System and Equipment Functions

System and equipment functions associated with the FPC system include the following.

9A.1.3.1.1 Normal Functions (Non-Safety-Category)

1. The FPC system maintains the water quality of the Fuel Pool, Reactor Cavity Pool, and Equipment Pool through filtration and demineralization during Modes 1-6. Refer to Chapter 16, Subsection 16.7 for definitions of operating Modes.

9A.1.3.1.2 Normal Functions (Safety-Category)

1. The FPC system provides cooling (SC3) of the Fuel Pool, Reactor Cavity Pool, and Equipment Pool during Modes 1-6.
2. The FPC system provides makeup capacity for the Fuel Pool (SC3), Reactor Cavity Pool, and Equipment pool (SC3) during Modes 1-6 delivered directly to the surge tanks and circulated to the pools.
3. The FPC system maintains the water level of the Fuel Pool (SC3) for shielding and cooling.
4. The FPC system provides high heat load cooling of the Fuel Pool and Reactor Cavity Pool during Mode 6 (refueling outage).

9A.1.3.1.3 Off-Normal Functions (Non-Safety-Category)

1. The FPC system is able to restore the Fuel Pool temperature to normal operating limits from elevated temperatures due to an off-normal event upon restoration of the forced cooling components of the system.

9A.1.3.1.4 Off-Normal Functions (Safety-Category)

1. The FPC system provides makeup capacity for the Fuel Pool (SC3) during off-normal events, independent of the forced cooling portion of the system.
2. The FPC system maintains the water level of the Fuel Pool (SC3) for shielding and cooling during off-normal events.

9A.1.3.2 Safety Design Bases

The FPC System provides continuous cooling normal plant operations, and makeup capacity for the fuel pool during normal and off-normal plant operations. Shielding is afforded by maintaining fuel pool water levels. The FPC System provides high heat load cooling of the Fuel Pool and Reactor Cavity Pool during Mode 6 (refueling outage).

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The fuel pool and its supporting cooling system functions are capable of maintaining the fuel pool temperature within a set upper limit given the:

1. Normal decay heat load that occurs when an accumulation of spent fuel equal to 8 years is in the fuel pool, with the newest batch having just been placed in the pool during refueling within the first 72 hours after shutdown
2. Maximum decay heat load, which is the normal heat load plus the addition of the decay heat from a full core that is off-loaded to the fuel pool 72 hours after shutdown

FPC System provides make up capacity to the fuel pool during off normal events to maintain adequate fuel coverage.

The design of the FPC system meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.12.2 (Reference 9A.1.3-1) as related to the handling and storage systems associated with spent fuel specifically heat removal in operational states. Refer to Subsection 9A.1.4 for information related to the handling of spent fuel and Subsection 9A.1.2 for information related to spent fuel storage systems.

9A.1.3.3 Description

Refer to Figure 9A.1.3-1 which depicts the FPC.

The BWRX-300 includes capabilities for the handling and storage of new and spent fuel to support operation of the plant with fuel cycle durations from 12 to 24 months. The FPC system provides cooling and cleaning of the water in the Fuel Pool, reactor cavity pool, and equipment pool during power operations.

The FPC system and its supporting cooling system functions are capable of maintaining the Fuel Pool temperature within a set upper limit as described in Subsection 9A.1.3.2.

The FPC System provides make up capacity to the Fuel Pool during Off-Normal events to maintain adequate fuel coverage. The FPC system is constructed such that any breach of the system cannot cause draining of the Fuel Pool to a level below 3.05 m above the top of active fuel.

The FPC System consists of two trains of equipment, each with a pump, demineralizer, and heat exchanger. The capability to bypass the demineralizer while providing active cooling to support restoration of the pool temperature from conditions exceeding the demineralizer operational temperature limits is provided. Each set of components are placed in parallel to provide single train operation and cross connecting of trains should a component fail. A single train is sufficient to prevent bulk boiling in the Fuel Pool. If both trains are rendered inoperable, the Fuel Pool is sized such that it can retain sufficient coverage of the fuel for seven days, and FPC can provide makeup capacity independently of the forced cooling trains.

9A.1.3.3.1 Component Description

The following information is provided relative to the major equipment and components in the FPC.

Surge Tank Description

The surge tank(s) receive inventory from the reactor cavity and fuel pool as it overflows weirs located near the top of the reactor cavity and fuel pool. Surge tank(s) are utilized to provide protection from a breach in the system draining either pool volume (versus direct suction) and provide a dampening effect for volume changes in the pools due to outage or dry cask evolutions. The primary advantage of a separate surge volume is that the pool level can be constant, while natural evaporation or addition or removal of submerged equipment only varies the level of the

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surge tank. Coolant inventory addition can be added directly to the surge tank during normal operating conditions. The FPC system surge tanks meet the requirements of American Society of Mechanical Engineers (ASME) Section III NCA and NCD.

Pumps

Each pump represents the beginning of two separate trains of the FPC system. They can be run individually during normal heat removal and cleanup modes (A1, A2) or in combination for high heat load (B, C) operating modes. Pumps are sized for maximum efficiency at the normal operating point. Pump internal recirculation cavitation is avoided within the entire range of operation. The pump selection, installation and system design assure that the minimum available Net Positive Suction Head (NPSH) and minimum submergence meet the Hydraulic Standards Institute guidelines for all steady-state and transient conditions of operation. If parallel operation is specified, the head rise from rated point to shutoff is at least 10%.

The FPC system pumps meet the requirements of ASME Section III NCA and NCD.

Heat Exchangers

Each train includes two heat exchangers (four in total) used to reject waste heat to the Plant Cooling Water System (PCW) (Subsection 9A.2.1). Each heat exchanger is a shell and tube design, with reactor coolant flowing on the tube side to take advantage of inherent shielding and reduce dose. Each heat exchanger is sized to fit within the Reactor Building hatch to allow removal or replacement during the operating life and decommissioning of the plant. Heat exchangers are designed, manufactured, installed, and tested in accordance with applicable codes and standards, including Thermal Exchanger Manufacturers Association, API-662, ASME Boiler and Pressure Vessel Code (BPVC), Section III and ASME BPVC Section VIII. Large shell and tube heat exchangers are designed with provisions for either tube bundle replacement or individual tube replacement and individual tube plugging in place. Heat exchanger fouling factors are established considering conservative predictions of material buildup based on actual system and equipment designs and expected plant operating conditions. Heat exchanger tubes are seamless.

Piping/Valving

Piping in radioactive systems such as the FPC System have butt-welded connections, rather than socket welds, to reduce crud traps. Features to prevent flow discontinuities that can lead to retention of corrosion products (crud traps) in the walls of the equipment and components are incorporated into the design. Bends, branches, corners, dead legs, and low points are avoided in piping and piping layout. Mitigating engineering features are added where avoidance is not possible. These piping and valve design features reflect implementation of ALARA guidance as presented in Chapter 12, Section 12.1.

FPC system piping potentially containing resin is designed to be continuously sloped downward to the receiving system or tank.

The FPC system piping, and valves meet the requirements of ASME Section III NCA and NCD.

Demineralizers / Filters

Each demineralizer train of FPC system contains a particulate filter and deep bed mixed resin demineralizer which can each support 100% of the train's flow. The filter may be backwashed to the Liquid Waste Management System (Chapter 11, Section 11.2) to remove accumulation and reduce dose. The demineralizers each operate in both normal (A1, A2) and high heat (B) modes, but can also be bypassed if the temperature of the system coolant exceeds the operational limits of the demineralizer (C).

Process equipment that accumulates a radiation source from filtering process streams such as the Condensate Filters and Demineralizers System (CFD) (Chapter 10, Subsection 10.3.1), and the FPC systems are remotely operated, including the backwashing operations. Provisions are made for remotely backflushing the filters and demineralizers. FPC system filters are backwashed into a backwash receiving tank, which then is routed to the Radwaste Systems. All FPC System valves (e.g., inlet, outlet, recycle, vent, and drain) on the filters and demineralizers are located outside the shielded cubicles in a separate shielded cubicle or area together with associated piping, headers, and instrumentation.

The FPC system filter element shell and demineralizer vessel meet the requirements of ASME Section III NCA and NCD.

Off-Normal Makeup

The Off Normal makeup portion of the system represents the seismic DL4b portion of the system intended to provide replacement inventory to the fuel pool. It utilizes multiple sources to maintain pool level in the event of a failure of the active cooling portion of FPC.

9A.1.3.4 Materials

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from Intergranular Stress Corrosion Cracking (IGSCC) (as applicable) through material chemistry, heat treatment, contamination, and material processes controls.

Features to prevent flow discontinuities that can lead to retention of corrosion products (crud traps) in the walls of the equipment and components are incorporated into the design. Bends, branches, corners, dead legs, and low points are avoided in piping and piping layout.

9A.1.3.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.1.3-1 for FPC interfaces with other equipment or systems.

9A.1.3.6 System and Equipment Operation

The FPC system is primarily intended to maintain pool temperatures during plant operation and refuel outages. During Modes 1-5, a single train of FPC is running to provide cooling and cleanup of the Fuel Pool, Reactor Cavity, and Equipment Pool inventory (Mode A1 and A2.) Supply to each pool may be balanced based on temperature variations and heat loadings in each volume. During refuel outages (Mode 6) the FPC system may be run in normal (FPC mode A1 or A2) or High Heat Mode (FPC mode B) as the temperature of the Reactor Cavity and Fuel Pool dictates. In the event of elevated pool temperatures, which exceed the operating specifications of the demineralizers such as restoration from an Off-Normal event, the FPC system may be run in High Heat – Bypass (FPC mode C) which utilizes both equipment trains of pumps and heat exchangers to provide maximum cooling without cleanup. Once the pool temperature returns to acceptable levels, the FPC system may then return to Mode B or Mode A as required.

9A.1.3.6.1 Initial Configuration (Pre-Startup)

As a system intended to always be in operation, there are no specific pre-startup conditions. Operation of the FPC system is independent of reactor operation.

9A.1.3.6.2 System Startup

Generally, a single train of FPC is running continuously.

During startup of a train, confirmation of the following is obtained:

- Valve alignment for the intended mode
- Sufficient surge capacity
- Pump inlet pressure to avoid cavitation
- Pump start
- Anticipated component performance indications

This same process may be used to add a second train during high heat modes. In the changeover from one train to the other, it is preferable to start the second train and confirm normal operation prior to shutting down the first train.

9A.1.3.6.3 Normal Operations

During normal operations the FPC system operates in the following modes:

Mode A – Normal Heat Load

The most frequent mode of operation of the FPC system is “normal heat load” operation, intended to be used during plant operating Modes 1-5. This mode includes a single train of the FPC system operating to remove heat generated by spent fuel stored in the fuel pool, as well as any thermal contribution from the Passive Containment Cooling System (Chapter 6, Subsection 6.3.3) to the equipment pool, or thermal leakage from containment to the reactor cavity pool. Cleanup consisting of particulate filtration and demineralization of 100% of the water passing through the system is performed. During Mode A, coolant inventory can be added from the condensate storage tank as required to makeup evaporation losses or over-boarded to the condensate storage tank after cleanup and cooling to create additional pool volume for submersion of spent fuel dry cask equipment. Mode “A1” and “A2” may be used to signify which train is in operation, both trains function identically.

Mode B – High Heat Load

Mode B utilizes both trains of FPC equipment running simultaneously to double the volume of water cooled and cleaned. This mode still retains 100% filtration and demineralization and is intended to be used predominately during plant operating Mode 6 “Refueling,” when high activity spent fuel is transferred to the fuel pool, and water clarity and dose are of particular concern. As with Mode A, coolant inventory can be added or overboarded to support outage operations and maintain water level.

9A.1.3.6.4 Off-Normal Operations

Mode C – High Head Load – Bypass

Mode C functions identically to Mode B, except the demineralizers are isolated from the system and a bypass is utilized. Mixed bed resin demineralizers tend to have modest thermal operating limits, typically less than 60 °C which when exceeded can cause resin excursions contaminating plant equipment and pools. To retain the high heat removal capacity but prevent damage to the resin beds during high pool temperature events, Mode C may be utilized until the water temperature is restored to a threshold allowing Mode B service.

Mode D – Active Cooling Inoperative

Mode D is an off-normal condition where the active cooling and cleanup portion of the FPC system is unable to be operated such as during a site blackout. Mode D operations allows for multiple sources of coolant addition directly to the Fuel Pool, bypassing the remaining system or any potential breeches that may have occurred. The plant sources include the condensate storage

tank, and the Fire Protection system. Mode D also includes an Emergency Mitigating Equipment (EME) connection allowing addition of coolant through temporary means. The Fuel Pool volume is intended to contain sufficient water inventory to allow seven days of spent fuel decay heat to be absorbed while maintaining sufficient coverage of 3.05 m over fuel. Operations in Mode D is required for additional inventory to be added to meet the 30-day event criteria.

9A.1.3.6.5 System Shutdown

The FPC system is intended to operate, with at least a single train, during all modes of plant operation. Individual trains may be idled or placed in standby through shutdown of pumps and isolation of equipment as heat load or maintenance and testing require.

9A.1.3.7 Instrumentation and Control

As an SC3 system, control of the FPC system is performed through the Safety Class 2 and 3 Instrumentation and Control System indication of pool temperature and water level. During normal operation, pumped capacity is varied through use of one or both system trains, with output balanced through motor operated control valves to the equipment, reactor cavity, and fuel pool to distribute cooled water in a manner best matching the sources of thermal energy. Addition or reduction of inventory, both as a natural function of environmental conditions, as well as supporting maintenance and outage evolutions, can be performed manually.

During off-normal events, with the forced cooling portion of the system assumed to be inoperable, make up capacity can be initiated from various sources, again through the Safety Class 2 and 3 Instrumentation and Control System, as redundant Fuel Pool level indication requires. Major components (pumps, demineralizers, heat exchangers) have performance indication through the Safety Class 2 and 3 Instrumentation and Control System, and component protection such as bypassing of demineralizers if system temperature exceeds limits and stopping of pumps on low flow and low suction pressure signals are automated.

9A.1.3.8 Monitoring, Inspection, Testing, and Maintenance

Maintenance and testing support equipment reliability. SSC are designed to facilitate operation and maintenance. Trending of operational characteristics of equipment such as demineralizers, filters, and heat exchangers identifies reductions in performance signaling maintenance is required. Trains are switched periodically to balance equipment usage and provide surveillance opportunity to both trains. In-Service Inspection (ISI) and In-Service Testing (IST) requirements are established and include inspection/test frequency for SSC.

Maintenance activities related to the FPC system fall into two categories:

1. Ongoing/Frequent Maintenance: Anticipated maintenance activities such as backflushing of particulate filters, flushing and replacement of spent resin in demineralizers, and decontamination of lines to reduce area dose.
2. Infrequent Maintenance: Anticipated activities that occur during the life of the plant which includes equipment replacement of components such as pump impellers, motors, valves, heat exchangers, and filter cartridges. These maintenance evolutions take place as a response to long-term trending of system performance or as preventive maintenance scheduled based on anticipated component life span.

In both cases, maintenance on the FPC system is performed predominately during Plant Mode 1 operation, when thermal loads in the Fuel Pool are low allowing a single operable train to satisfy plant needs while the other may be isolated for work.

Maintenance practices consider industry best practices and operating experience and conform with plant safety requirements to minimize potential for personnel injury. Maintenance activities implement ALARA practices to minimize work activity dose. Maintenance activities involving plant equipment may require involvement of vendors or industry specialists.

9A.1.3.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions. Chapter 12, Subsection 12.3.8.3 presents Fuel Pool Cooling major ALARA design considerations.

9A.1.3.10 Performance and Safety Evaluation

The FPC System is designed to perform its function in a reliable and failure tolerant manner. This reliability is achieved with the use of rugged and redundant equipment. Each set of components (pumps, demineralizers, heat exchangers) are placed in parallel to provide single train operation and cross connecting of trains should a component fail. A single train is sufficient to prevent bulk boiling in the fuel pool. If both trains are rendered inoperable, the fuel pool is sized such that it can retain sufficient coverage of the fuel for seven days, and FPC can provide makeup capacity independently of the forced cooling trains.

The FPC System Safety-Category functions during Normal and Off-Normal conditions include providing makeup capacity to the Fuel Pool, maintaining the water level of the Fuel Pool for shielding and providing Fuel Pool Cooling. In addition, Off-Normal makeup piping and valves which are SC3 and Seismic Category A are provided to allow remote addition of water to the Fuel Pool during Off-Normal conditions through temporary means thereby ensuring spent fuel is cooled and fuel pool water levels are maintained.

In the event of an aircraft impact, there are provisions to refill the pool utilizing Emergency Mitigating Equipment (EME) or other equipment and water sources in a suitable time frame such that adequate cooling of the spent fuel is maintained based on keeping the fuel covered with sufficient water.

The Fuel Pool is designed to dissipate the maximum spent fuel decay heat through heat-up and boiling of the fuel pool water. The most conservative heat load for the Fuel Pool occurs when the Fuel Pool contains spent fuel from the normal decay heat load that occurs when an accumulation of spent fuel equal to 8 years is in the fuel pool, with the newest batch having just been placed in the pool during refueling within the first 72 hours after shutdown.

The Fuel Pool is located in the Reactor Building which provides protection against natural phenomena; supporting the ability of the FPC System to perform its Safety-Category functions.

9A.1.3.11 References

- 9A.1.3-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

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Table 9A.1.3-1: Fuel Pool Cooling and Cleanup System Interfaces

Interfacing System	Interface Description	Interface Boundary
Reactor Building, Fuel Pool, Reactor Cavity, and Equipment Pool	Overflow from the Fuel Pool and Reactor Cavity Pool weirs enters skimmer surge tanks and is pumped through cleanup and heat removal subsystems, then returned to the bottom of the Equipment Pool, Reactor Cavity Pool, and Fuel Pool as required.	Boundary from the Reactor Building exists at overflow from pools to Skimmer Surge Tanks. Boundary at supply exists at sparger nozzle(s) submerged in pools.
Liquid Waste Management System (LWM)	Condensate Storage Tank provides additional coolant inventory or storage volume for overboarding to reduce inventory. LWM also supplies liquid for back wash and flushing of cleanup systems.	Boundary exists at first isolation valve at each connection point.
Solid Waste Management System (SWM)	Repository for spent resin from demineralizers and backwash of filter elements. Flushed via piping to SWM using LWM inventory.	Boundary exists at first isolation valve at each connection point.
Offgas System (OGS)	Vent used to allow compressed air circulation of demineralizer resin and venting of volume.	Boundary exists at first isolation valve at each connection point.
Plant Cooling Water System	PCW supply for heat exchangers.	Boundary exists at first isolation valve at each connection point.
Plant Pneumatics System (PPS)	Air supply to allow mixing of demineralizer resin. Air Operated Valve operation as required.	Boundary exists at first isolation valve at each connection point.
Safety Class 2 and 3 Instrumentation and Control	Provides instrumentation and control for Safety Class 3 functions of FPC system.	Boundary exists at each rack prior to multiplexer.
Safety Class 2 and 3 Electrical Distribution System	Electricity supplied to pumps, motorized valves, and instrumentation/control throughout system.	Boundary located at individual component sub feed.
Process Radiation and Environmental Monitoring System (PREMS)	PREMS provides area radiation detector near the demineralizer, process sampling for radiological analysis and chemistry control.	Area radiation detector, process sampling routed to sample station.
Fire Protection System	Provides additional coolant inventory in Off-Normal conditions.	Boundary exists at first isolation valve at each connection point.

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Interfacing System	Interface Description	Interface Boundary
EME	Although not a standalone system, a connection is included to provide makeup capacity from remote location.	Boundary at final valve of EME connection point.

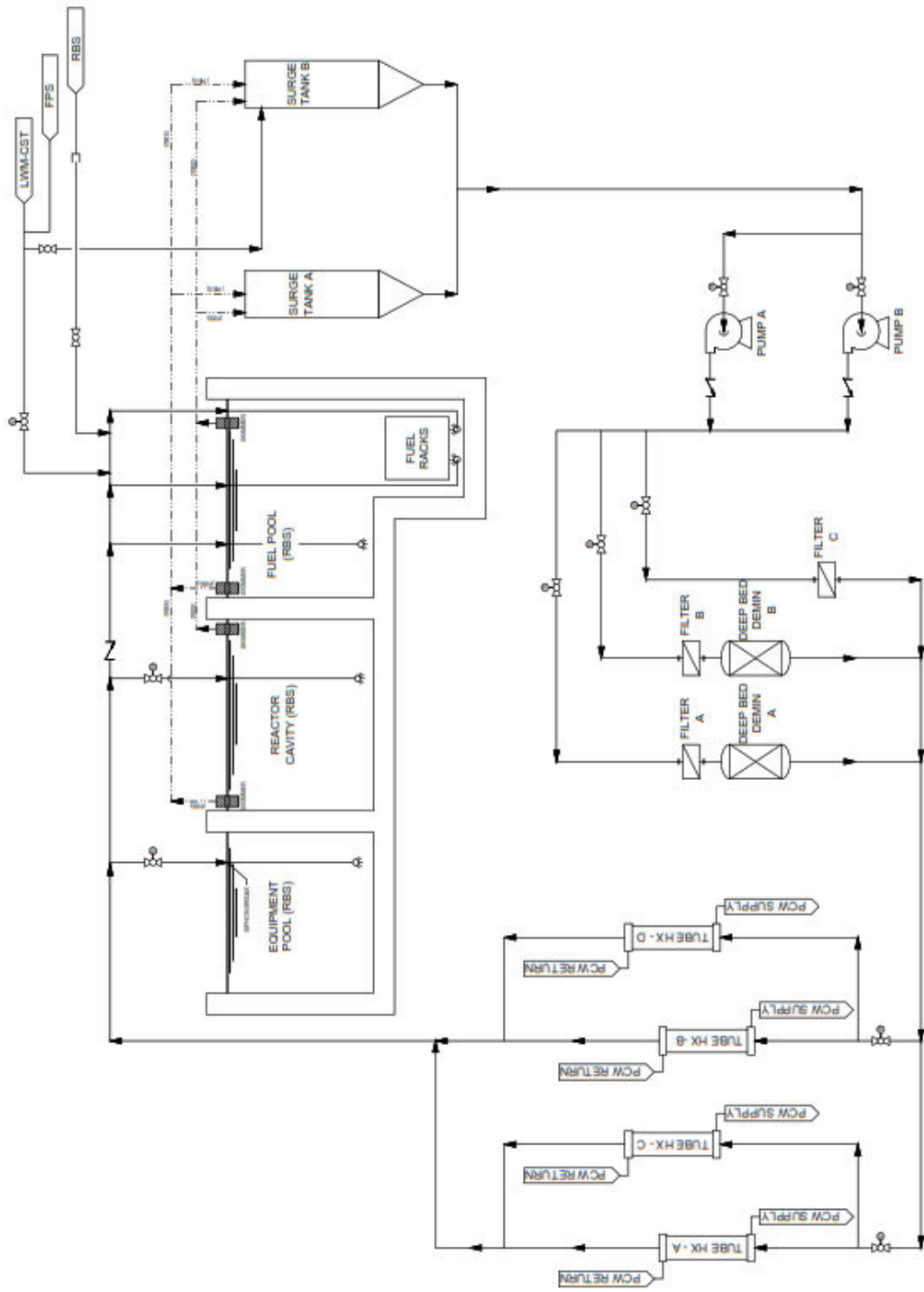


Figure 9A.1.3-1: Fuel Pool Cooling and Cleanup System

9A.1.4 Handling Systems for Fuel Cask Loading

9A.1.4.1 System and Equipment Functions

The system and equipment functions associated with the handling system for fuel cask loading, provide the means to transfer spent fuel located in the fuel pool into a spent fuel storage cask and transport of the spent fuel storage cask to a location for long-term storage. The following information is provided relative to demonstrating compliance to the requirements CNSC REGDOC-2.5.2 Section 8.12.1 (Reference 9A.1.4-1) as it pertains to fuel cask loading.

9A.1.4.2 Safety Design Bases

1. Fuel handling devices have provisions to avoid dropping or jamming of spent fuel assemblies during transfer operations.
2. Handling equipment used to raise and lower spent fuel has a limited maximum lift height so that the minimum required depth of water shielding is maintained.
3. Criticality during fuel handling operations is prevented by maintaining a geometrically safe configuration throughout the spent fuel transfer operation. Operators follow a strict movement plan and procedure and that includes pick up and set down locations for each step and those are followed and verified concurrently as each step is started and completed.
4. In the event of a Safe Shutdown Earthquake (SSE), handling equipment cannot fail in such a manner as to prevent required function of Safety Category 1 SSC.
5. Physical safety features are provided for personnel who operate handling equipment.

Refer to Subsection 9A.1.2.2 for the Safety Design Bases associated with the Fuel Pool. Refer to Section 9A.8 for the Safety Design Bases associated with the Cranes, Hoist, and Elevator System.

9A.1.4.3 Description

9A.1.4.3.1 Refueling Platform

The Refueling Platform is a gantry-type crane that spans the reactor vessel cavity, equipment pool, and fuel pool and is employed to handle fuel and perform other ancillary tasks in the reactor building. It is equipped with a traversing trolley on which is mounted a telescoping mast and fuel grapple. A monorail and auxiliary hoist are also provided. The Refueling Platform is a rigid structure built to precise engineering standards to ensure accurate and repeatable positioning during the refueling process. The Refueling Platform does not perform a Safety-Category function but is designed as Seismic Category I.

9A.1.4.3.2 Irradiated Fuel Canister Loading

The spent fuel is stored in the Fuel Pool before being loaded into a canister. Typical loading of a canister includes the following. Loading of the canister is performed in the Fuel Pool. The canister containing the spent fuel is then loaded into a concrete overpack. Canister loading patterns are determined by the age of fuel, exposure, and decay heat. A campaign normally involves loading the spent fuel canister while the unit is on-line.

An empty canister placed in a transfer cask for shielding is moved into the fuel pool. The selected bundles are moved under water from the spent fuel racks to the allotted location in the spent fuel canister. Once a spent fuel canister is filled with spent fuel, a video recording of the bundle serial numbers is performed, and the lid is placed on the canister. The canister is lifted from the fuel pool, drained, rinsed, and set on the cask pad on the refueling floor where the canister is

decontaminated. The canister is vacuum dried and sealed by welding the lid using remote welding techniques to minimize radiation exposure to workers. The canisters are filled and slightly pressurized with an inert gas, such as helium. The sealed canisters are leak tested and a non-destructive examination is performed on the weld. The shielded canister is installed into an overpack and then fitted with an overpack lid and placed in long-term storage.

Each spent fuel canister may contain several damaged fuel bundles; however, the specific loading is vendor dependent. The failed fuel bundles can stay in the fuel pool indefinitely without any special controls. If desired, fuel bundles can be reconstituted with new fuel or dummy rods replacing damaged rods. This is done on the fuel inspection stands in the fuel pool with long handled tools.

9A.1.4.3.3 Irradiated Fuel Canister Movement to ISFSI Pad

Typical canister movement from the fuel pool to the Independent Spent Fuel Storage Installation (ISFSI) pad is performed under the guidance and participation of Security and Health Physics. The RB Polar Crane is used to lift the transfer cask to the refuel floor and subsequently to the truck bay. The canister is transferred from the transfer cask to the storage cask and placed in long-term storage.

9A.1.4.3.4 ISFSI Pad

The ISFSI pad size is related to the vendor selected for the design and manufacturer of the spent fuel casks and the number of fuel assemblies to be stored.

The land usage required to provide long-term dry storage of spent fuel includes the fenced off area necessary to provide an acceptable radiation protection and security zone. Perimeter fencing, intrusion detection system, lighting and cameras are provided as required by CNSC irradiated fuel storage facility requirements.

9A.1.4.4 Materials

Materials selected for use in the Handling Systems for Fuel Cask Loading are chosen based upon the operating conditions to which they are required to function. The fuel storage racks use surveillance coupons for monitoring and evaluation of neutron absorber material.

9A.1.4.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.1.4-1 for Handling System for Fuel Cask Loading interfaces with other equipment or systems.

9A.1.4.6 System and Equipment Operation

The handling of new fuel is described in Section 5 of the Safeguards Annex. The handling of irradiated fuel inside the fuel pool is described in Subsection 9A.1.2. Handling of irradiated fuel from the fuel pool to a storage canister is discussed in Section 5.1 of the Safeguards Annex. Handling of the spent fuel storage canister with respect to the transfer of the spent fuel canister to an overpack is discussed in Subsection 9A.1.4.3. Handling of the overpack during transport to long-term storage is discussed in Subsection 9A.1.4.3.

9A.1.4.7 Instrumentation and Control

Refer to Subsection 9A.1.3.7 for information pertaining to instrumentation and control associated with the fuel pool and Section 9A.8 for information pertaining to instrumentation and control associated with the RB Polar Crane.

9A.1.4.8 Monitoring, Inspection, Testing, and Maintenance

Maintenance of refueling equipment and tooling is performed prior to an outage to promote optimum reliability.

Refer to Subsection 9A.1.2.8 for information pertaining to the Monitoring, Inspection, Testing, and Maintenance of the fuel pool. Refer to Subsection 9A.8.1.8 for information pertaining to the Monitoring, Inspection, Testing, and Maintenance of Cranes, Hoists and Elevators.

9A.1.4.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.1.4.10 Performance and Safety Evaluation

The handling of new fuel is described in Section 5 of the Safeguards Annex. The handling of irradiated fuel inside the fuel pool is described in Subsection 9A.1.2. Handling of irradiated fuel from the fuel pool to a storage canister is discussed in SubSection 9A.1.4.3.2. Handling of the spent fuel storage canister with respect to the transfer of the spent fuel canister to an overpack and transport to the Independent Spent Fuel Storage Installation is discussed in Subsection 9A.1.4.3.3:

1. The fuel handling equipment is designed such that probability of dropping a fuel assembly following a safe shutdown earthquake has been minimized.
2. Each fuel assembly and control rod is placed strategically to maintain a non-critical configuration by following a specific fuel movement plan.
3. Underwater transfer of spent fuel assemblies provides radiation shielding. The fuel handling equipment has provisions to limit maximum height to maintain sufficient water inventory above the top of the fuel assembly.
4. The fuel handling equipment includes controls and interlocks that impose limits upon system operations, ensuring clearance between structures, systems, and components, thereby preventing the potential for mechanical damage to fuel during fuel transfer operations.

9A.1.4.11 References

- 9A.1.4-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

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Table 9A.1.4-1 Handling System for Fuel Cask Loading Interface

Interfacing System	Interface Description	Interface Boundary
Fuel Pool Cooling and Cleanup System	FPC provides water clarity for underwater visibility FPC also provides water cooling to support fuel pool cooling activities.	The Handling System for Fuel Cask Loading equipment is immersed in water treated and cooled by FPC.
Core and Fuel	Supports the transfer of fuel assemblies in and out of the reactor core.	The mast of the Refueling Platform engages the bail handle of the fuel assembly.
Plant Pneumatics System	PPS provides pressurized air for powering pneumatic motors and tooling.	Gate seals, nozzle plugs, and tooling.
Non-Safety Electrical Distribution System	Provides electricity to the Handling System for Fuel Cask Loading equipment.	Inspection tooling, underwater lights, refueling platform, auxiliary platform, and tooling.
Grounding and Lightning Protection System	Provides electrical grounding for equipment.	Equipment and inspection equipment.
Cranes, Hoists and Elevators	The RB Polar Crane is used during the disassembly and re-assembly of the Reactor Pressure Vessel.	The Handling System for Fuel Cask Loading equipment couples to RB Polar Crane hook.
Heating Ventilation and Cooling System	Provides control of heat and humidity to work areas.	Ambient condition suitable work.
Reactor Building Structure	Supports the loads of equipment and provides railing for traversing of the Refueling Platform.	Equipment either rests on the refuel floor or its weight is transmitted to the RB walls through the RB Polar crane.

9A.2 Water Systems

9A.2.1 Plant Cooling Water System

The PCW system provides cooling water to Non-Safety and Safety Class 3 components and provides a barrier against radioactive contamination of the Circulating Water System (CWS) (Chapter 10, Section 10.8). It consists of two piping subsystems, Reactor Component Cooling Water Piping Distribution and Turbine Component Cooling Water Piping Distribution, that provide cooling water to various heat exchangers.

The safety classification of the PCW as well as interfacing Structures, Systems, and Components (SSC) is consistent with the requirements of CNSC REGDOC-2.5.2, Section 7.1 (Reference 9A.2.1-1).

9A.2.1.1 System and Equipment Functions

9A.2.1.1.1 Normal Functions (Non-Safety-Category)

The PCW circulates cooling water to the Reactor Component Cooling Water Piping Distribution and the Turbine Component Cooling Water Piping Distribution during normal operation and during anticipated operational occurrences, including startup, power operation, hot shutdown, cold shutdown, stable shutdown and refueling. The PCW is responsible for rejecting the total heat load associated with the equipment coolers in the Reactor Component Cooling Water Piping Distribution and Turbine Component Cooling Water Piping Distribution.

PCW operation is normally automatic based on plant operational mode but can be started or stopped manually from the Main Control Room and can operate at any time, regardless of the operational status of the generating unit.

The PCW design supports the redundant Reactor Component Cooling Water Piping Distribution and single Turbine Component Cooling Water Piping Distribution.

The cooling water temperature is maintained by a temperature control valve arrangement that bypasses the PCW heat exchangers with a portion of the return water.

9A.2.1.1.2 Normal Functions (Safety-Category)

The system provides the following normal safety functions:

- Control of the fuel pool temperature
- The system provides cooling for the SDC heat exchanger
- A barrier against radioactive contamination of the CWS

9A.2.1.1.3 Off-Normal Functions (Safety-Category)

Upon a Loss-of-Offsite Power (LOOP), the operating PCW pump(s) will trip. The pumps are automatically repowered from the standby diesel generators.

In the event of LOOP, the reactor cooling loops are used to support cooling of the Fuel Pool and Reactor to bring the plant to cold shutdown condition in 36 hours if necessary assuming the most limiting single active failure (single train in use), with use of the Isolation Condenser System (ICS) (Chapter 6, Section 6.2). If both trains are still in use, plant systems provide adequate shutdown cooling capacity to achieve appropriate reactor coolant temperature in 24 hours.

9A.2.1.1.4 Off-Normal Functions (Non-Safety-Category)

The system does not perform any Safety-Category functions during off-normal conditions.

9A.2.1.2 Safety Design Bases

The Safety Design Bases for the PCW includes the following Safety Class 3 functions:

- Temperature control of the fuel pool
- Cooling for the SDC heat exchanger

9A.2.1.3 Description

Figure 9A.2.1-1 depicts the Plant Cooling Water System.

The PCW system consists of two trains, each containing one pump and one heat exchanger, that handle the Reactor Component and Turbine Component cooling loads. One train is normally in operation while the other is on standby, to be started in event the operating train needs to be shutdown.

These independent trains are cross connected using manual cross ties to allow for online maintenance. If necessary, the Turbine Component Cooling Water Piping Distribution can be isolated from each Reactor Component Cooling Water Piping Distribution by closing the supply and return header valves on each train, and each Reactor Component Cooling Water train operates independent of the other.

Cooling water supplied by the PCW is continuously circulated through various auxiliary equipment heat exchangers and rejects the heat transferred to the CWS. The Turbine Component Cooling Water Piping Distribution is a single piping distribution that serves all equipment in the Turbine Building that do not support reactor cooling functions. The Reactor Component Cooling Water Piping Distribution consists of two redundant piping distributions that support all Reactor Building equipment cooling functions, Plant Pneumatic System and any equipment located outside the Reactor Building associated with reactor cooling activities.

Although PCW supports all plant equipment cooling functions, the design redundancy and isolation capability is centered around the ability to provide redundant cooling supply to nuclear systems.

One train of PCW equipment is normally in operation and in the event that train is lost, the standby train is started. During plant shutdown both trains of cooling equipment are used to provide required flow for Turbine Building and Reactor Building cooling functions. The Reactor Component Cooling Water Piping Distribution cools redundant equipment associated with nuclear cooling functions, thus if one train of the Reactor Component Cooling Water Piping Distribution is lost all redundant equipment will operate on the remaining train.

The PCW pumps and heat exchangers are in the Turbine Building. The pumps in each train are powered from separate busses. During a LOOP, the pumps are powered from the two Safety Class 3 standby diesel generators.

The PCW utilizes plate and frame type heat exchangers. This design mitigates cross-contamination of either PCW or CWS.

Temperature control valves in the system are provided to maintain the cooling water temperature within an allowable range.

Surge tanks provide a constant pump suction head and allow for thermal expansion of the PCW inventory. Makeup to the PCW inventory is from the PCW surge tank which is supplied from the Makeup Water System (Subsection 9A.9.5.1) through an automatic level control valve.

A chemical addition tank tie-in allows for manual introduction of corrosion inhibitor and pH control chemicals into the system.

The Turbine Building cooling loop serves but is not limited to the following equipment:

- Feedwater Pump Motors
- Feedwater Pump Variable Frequency Drive
- Condenser Pump Motor
- Vacuum Pump Skid
- Generator Cooler
- Isophase Cooler
- Lube Oil Cooler
- Electro-Hydraulic Control (EHC) Cooler

The redundant Reactor Building cooling loops serve the following equipment:

- Fuel Pool Cooling and Cleanup Heat Exchanger
- Isolation Condenser System Pool Cooling and Cleanup System (ICC) Heat Exchanger
- Shutdown Cooling System (SDC) Heat Exchanger
- Plant Pneumatics System Cooler

PCW equipment is designed for the plant normal operating environmental conditions specified for its location within the Turbine Building and Reactor Building. Refer to Chapter 3, Section 3.9 for information pertaining to Environmental Qualification of BWRX-300 SSC's.

9A.2.1.3.1 Component Description

PCW Pumps

The PCW pumps are designed to meet the requirements of ANSI/HI 1.3 (Reference 9A.2.1-2).

The pumps are constant speed, electric motor driven, horizontal centrifugal pumps. Pump impellers are less than the maximum diameter available for the pump casing. The pumps' mechanical seals minimize the potential for leakage and reduce the need to perform maintenance on shaft seal packings.

PCW Heat Exchangers

The PCW heat exchangers are designed to meet the requirements of ASME BPVC, Section VIII.

PCW is capable of performing design functions during all modes of operation.

Plate and frame heat exchangers are used to minimize the probability of possible cross-contamination between PCW and CWS. Strainers are installed on the raw water side of the exchangers due to the narrow passages plate and frame heat exchangers present. Leakage through holes or cracks in the plates is not considered credible based on industry experience with plate type heat exchangers. In addition, the heat exchangers are designed such that any gasket leakage from either PCW or CWS drains to the EFS (Subsection 9A.9.3).

The CWS water temperature on the other side of the heat exchanger to absorb the heat load. There is a bypass line provided for the PCW Heat Exchangers with an air-operated bypass temperature control valve. The flow of cooling water through the PCW Heat Exchangers is controlled by the PCW Heat Exchanger temperature control system aiming to achieve target

temperature based on the PCW Heat Exchangers outlet temperature readings sent to the Distributed control and Information System.

PCW Surge Tanks

The PCW surge tanks are atmospheric corrosion-resistant tanks which are designed to meet the requirements of American Water Works Association D100. The surge tanks are sized to provide a pump suction head and to allow for thermal expansion. The surge tanks are located above the highest points in the PCW system. The makeup water supply is provided to maintain the water level in the tanks within the correct operating conditions.

Piping and Valves

All Non-Safety-Category function piping is designed to meet the requirements of ASME B31.1, Power Piping.

Isolation valves are provided at the inlet and outlet of each heat exchanger and pump for maintenance and efficiency purposes. Isolation valves are provided at the interfaces with the components being cooled by PCW to allow for maintenance on those items without impact to the PCW system operation. These isolation valves add the ability to remove heat exchangers from the system cooling requirements without shutting down the entire PCW system.

For the PCW pump and heat exchanger train section, there are isolation valves around the pumps and heat exchangers. Each pump discharge line is provided with a check valve to prevent backflow through the pump.

Vents are located in high points and drains are located in all low points. This ensures that the system is completely filled with water and that there are no air pockets. Vents reduce the chance for water hammer after a pump start. Valve opening and closing times are selected to minimize water hammer effects.

Each flow path to the interface system heat exchangers is designed to have flow balancing features that may include fixed plate orifices and/or control or manual valves.

Chemical Pot Feeder

The chemical pot feeder is designed to meet ASME BPVC, Section VIII requirements. The chemical pot feeder is used to manually add corrosion inhibitors, pH control chemicals, and biocides into the system on a periodic basis. Chemicals are compatible with radwaste processing equipment if PCW water becomes contaminated.

9A.2.1.4 Materials

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials through material chemistry, heat treatment, contamination, and material processes controls.

9A.2.1.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.2.1-1 for PCW interfaces with other equipment or systems.

9A.2.1.6 System and Equipment Operation

9A.2.1.6.1 Normal Operational Concept

For normal start up, the PCW is filled through the surge tank. When surge tank levels reach low level, the Makeup Water System (Subsection 9A.9.5.1) refills the surge tank through an automatic level control valve.

For normal operation, one train which consists of one pump and one heat exchanger is in operation while the other train is on standby. If one heat exchanger fails, the standby heat exchanger has enough capacity to handle cooling requirements.

Plant cooling water flow through to heat exchangers and coolers throughout the Turbine Building and Reactor Building by way of the Turbine Closed Cooling Water Subsystem piping and Reactor Closed Cooling Water Subsystem piping.

The CWS water flows on the other side of the PCW main heat exchanger to absorb the heat load. There is a bypass line provided for the PCW Heat Exchangers with an air-operated bypass temperature control valve. The flow of cooling water through to the PCW Heat Exchangers is controlled by the PCW Heat Exchanger temperature control system based on the PCW Heat Exchangers outlet temperature.

9A.2.1.6.2 Off-Normal Operational Concept

For off-normal modes of operation, the PCW system (Reactor Component Cooling Water Piping Distribution) provides Defence-in-Depth (D-in-D) functions to remove decay heat from the reactor, fuel pool, and isolation condenser pools. PCW piping trains are isolated to provide fully independent cooling water loops to both trains of equipment. Both trains of PCW are capable of supporting D-in-D cooling requirements associated with the plant.

In the event of a loss of one train of the PCW cooling system equipment, the standby train is placed into service thereby precluding plant shutdown.

Upon a LOOP, the PCW pumps trip and restart upon power from the standby diesel generators.

Any leakage resulting in a low surge tank level signal, the standby train of PCW isolates the Reactor Component Cooling Water Piping Distribution line and automatically places any non-operating equipment in bypass. The Water, Gas and Chemical Pads (WGC) system provides makeup water to the surge tanks. The train that continues to lose water will eventually trip on low suction head, leaving at least one train of PCW operational to support nuclear D-in-D functions.

In cases of leak detection the two isolation valves responsible for the Reactor Component Cooling Water Piping Distribution train isolation automatically close and the standby train starts and proceeds with cooling of the redundant Reactor Component Cooling Water Piping Distribution coolers.

9A.2.1.7 Instrumentation and Control

In normal operation one cooling train is operating while being controlled by the Distributed Control and Information System in order to achieve the desired target temperature for each cooled line in the Reactor Component Cooling Water Piping Distribution and Turbine Component Cooling Water Piping Distribution. In events of failure of the operating train or pump discharge pressure drop, an automated process switches operations to the standby train.

Temperature is monitored downstream of each system heat exchanger and readings are reported back to the main control room to monitor the heat exchanger performance.

For designated heat exchangers, bypass and discharge temperature control valves operating in split range control mode are used to control temperature. Instrumentation Temperature monitoring is provided, but not limited to the following locations:

- At the outlets of all equipment coolers
- Downstream and upstream of each PCW heat exchanger

Pressure monitoring is provided, but not limited to the following locations:

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- At the suction and discharge of each PCW pump
- Downstream of the PCW heat exchangers

Differential Pressure monitoring provided, but not limited to the following locations:

- Across each PCW heat exchanger
- Across equipment coolers
- Across the pump suction strainers

Flow monitoring provided, but not limited to the following locations:

- At the outlet of each PCW heat exchanger
- At the inlets of selected system heat exchangers as needed

Level monitoring is provided at the following locations:

- On each surge tank

Temperature Control Valves are provided at the following locations:

- Downstream and Across (bypass) each SDC heat exchanger
- Downstream and Across (bypass) the Generator Coolers
- Downstream and Across (bypass) the Lube Oil Coolers
- Downstream and Across (bypass) each PCW heat exchanger

9A.2.1.7.1 Controls

The PCW system is monitored from the Main Control Room. PCW controls and interlocks for the main components are described below:

1. Temperature monitoring is provided downstream of the PCW heat exchangers on the main cooling water supply line, that send a signal to the DCIS to control cooling water temperature by bypassing a portion of the water around the PCW heat exchangers. This is accomplished using an air-operated temperature control valve on the bypass line and an air-operated temperature control valve at the heat exchanger discharge header, and both are controlled by the DCIS. These valves regulate the supply temperature. The bypass and discharge valve have the ability to be manually controlled. The bypass valve fails closed and the discharge valve fails open.
2. The PCW surge tank makeup flow is controlled by an air-operated block valve. The valve automatically opens and closes and can be manually controlled. The block valve opens when the PCW surge tank level drops to a predetermined low level. The block valve closes when the surge tank level rises to a predetermined high level. The surge tank makeup water inlet block valve fails closed.
3. The cooling water temperature to the Turbine Lube Oil coolers is regulated by two temperature control valves, one at the Turbine Lube Oil cooler discharge and one on the bypass around the cooler, which operate in split range control mode. This type of flow will naturally balance the flow around the coolers. The bypass and discharge valves have the ability to be manually controlled. The bypass valve fails closed and the discharge valve fails open.

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4. The cooling water temperature to the Generator Air Coolers is regulated by two temperature control valves, one at the Generator Air Coolers discharge and one bypass around the coolers, which operate in split range control mode. This type of control will naturally balance the flow around the coolers. The bypass and discharge valves have the ability to be controlled manually. The bypass valve fails closed and the discharge valve fails open.
5. The cooling water temperature to the Shutdown Cooling Heat Exchangers is regulated by two temperature control valves, one at the Shutdown Cooling Heat Exchanger discharge and one bypass around the cooler, which operate in split range control mode. This type of control will naturally balance the flow around the heat exchangers. The bypass and discharge valves have the ability to be controlled manually. The bypass valve fails closed and the discharge valve fails open.
6. Upon a low surge tank level signal, the standby train of PCW isolates the Reactor Component Cooling Water Piping Distribution line and automatically places any non-operating equipment in bypass. The train that continues to lose water will eventually trip on a low suction head signal, but at least one train of PCW is preserved in order to support nuclear defense in depth functions.
7. Normally one PCW pump and heat exchanger train is in service with the other on standby. A pump trip signal results in the starting of the standby PCW train and CWS train.
8. The PCW Heat Exchangers interfaces directly with the CWS system. PCW trains cannot be started until the associated CWS train is running. The system connects such that an automated standby train of either CWS or PCW results in automatic realignment. The isolation valves are automatic with manual overrides available.

PCW controls, displays, and alarms include, but are not limited to, the following:

Main Control Room Panel Controls:

- PCW pump(s) start/stop and pump selection controls
- PCW heat exchanger(s) outlet valve open/close controls
- PCW supply temperature control valve(s) controls
- Surge Tank Makeup water valve open/close control
- PCW Heat Exchanger temperature control Air Operated Valves (AOVs)

Main Control Room Displays:

- PCW pump discharge pressures
- PCW pump operation status
- PCW operating temperatures (All heat exchanger outlet temperatures)
- PCW surge tank levels
- AOV open/close status
- PCW system flow rates

Main Control Room Alarms:

- PCW pump header low/high discharge pressure

- PCW cooling water supply high outlet temperature
- PCW cooling water supply low outlet temperature
- PCW surge tank high and low levels
- PCW heat exchanger high differential pressure

9A.2.1.8 Monitoring, Inspection, Testing, and Maintenance

The PCW is designed such that major equipment is provided adequate equipment removal paths and personnel access points.

The equipment and components of the PCW are designed for inspection and maintenance during plant operation without requiring complete loss of the Turbine Building cooling loop or the Reactor Building cooling loops.

Provisions to isolate the Turbine Building cooling loop or the Reactor Building cooling loops using AOV is provided. This design feature supports inspection, testing and maintenance activities.

Provisions for PCW drainage to the Equipment and Floor Drain System (EFS) are provided due to its high probability of having chemical content. If the PCW captured water is not chemically contaminated, the water is recycled back to the PCW surge tank.

Routine testing of the PCW system is conducted in accordance with normal power plant requirements for demonstrating system and component functionality and integrity. This includes testing for heat exchanger performance, surge tank levels and water quality standards.

The PCW design includes provisions to take periodic samples for analysis to ensure the water quality meets the chemistry specifications.

9A.2.1.9 Radiological Aspects

Chapter 12, Section 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are as low as reasonably achievable in operational states and in accident or post-accident conditions.

9A.2.1.10 Performance and Safety Evaluation

System reliability is enhanced through the use of periodic system testing, and preventive maintenance.

The PCW includes the components necessary to provide cooling water in support of the redundant Reactor Component Cooling Water Piping Distribution and the Turbine Component Cooling Water Piping Distribution. Portions of the PCW necessary to support Safety Class 3 functions to provide temperature control of the fuel pool, and cooling for the SDC heat exchanger, are designed to Non-Seismic Category, Quality Group D requirements. The redundant design elements of the PCW design ensures that the safety function of the system is maintained.

9A.2.1.11 References

- 9A.2.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.2.1-2 ANSI/HI 1.3, "Rotodynamic Centrifugal Pumps for Design and Application," Hydraulic Institute.

Table 9A.2.1-1: Plant Cooling Water System Interfaces

Interfacing System	Interface Description	Interface Boundary
Safety Class 2 and 3 Instrumentation and Control System	Provides SC3 Instrumentation and Control. Pump logics, valve logics, instrumentation (pressure, flow, temperature, level) etc.	Safety Class 3 Instruments
Non-Safety Instrumentation and Control System	Provides Non-Safety Class (SCN) Instrumentation and Control Pump logics, valve logics, process variables, etc.	(SCN) Instruments
Process, Radiation and Environmental Monitoring System	Provides sampling points for chemical analysis	PCW Pump Common header
ICS Pool Cooling and Cleanup System	Provides cooling water to ICS pool cooler to ICS pool cooler from Reactor Component Cooling Water Piping Distribution	ICS Pool Cooler
Shutdown Cooling System	Provides cooling water to SDC cooler from Reactor Component Cooling Water Piping Distribution	SDC Cooler
Fuel Pool Cooling and Cleanup System	Provides cooling water to FPC cooler from Reactor Component Cooling Water Piping Distribution	FPC Cooler
Condensate and Feedwater Heating System (CFS)	Provides cooling water to Condensate and Feedwater Pumps and their Variable Frequency Drives from Turbine Component Cooling Water Piping Distribution	Condensate and Feedwater Variable Frequency Drive and Motors
Main Turbine Equipment (MTE)	Provides cooling water to Lube Oil and Electro-Hydraulic Cooler (EHC) from Turbine Component Cooling Water Piping Distribution	Lube Oil and EHC Coolers
Generator, Exciter, and Isophase Bus Ducts	Provides cooling water to the Generator cooler and Isophase Cooler from the Turbine Component Cooling Water Piping Distribution	Generator and Isophase Coolers
Main Condenser and Auxiliaries	Provides cooling water to the Vacuum Pumps skids from Turbine Component Cooling Water Piping Distribution water loop	Vacuum Pump Skids
Circulating Water System	Circulating Water acts as a heat sink to PCW heat exchangers	PCW Heat Exchangers

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Interfacing System	Interface Description	Interface Boundary
Plant Pneumatic System	Provides cooling water to the Plant Pneumatics System from Reactor Component Cooling Water Piping Distribution PPS Supplies instrument air to PCW Valves	PPS coolers Valve actuators
Safety Class 2 and 3 Electrical Distribution System	Provides PCW components power	SC3 pumps and valve motors
Non-Safety Electrical Distribution System	Provides Non-Safety PCW components power	SCN motors
Equipment and Floor Drain System	Provides drainage and collection for contaminated or potentially contaminated waste	Drain hub
Water, Gas, and Chemical Pads	Water, Gas and Chemical Pads provides demineralized makeup water to the PCW surge tanks	Surge Tank

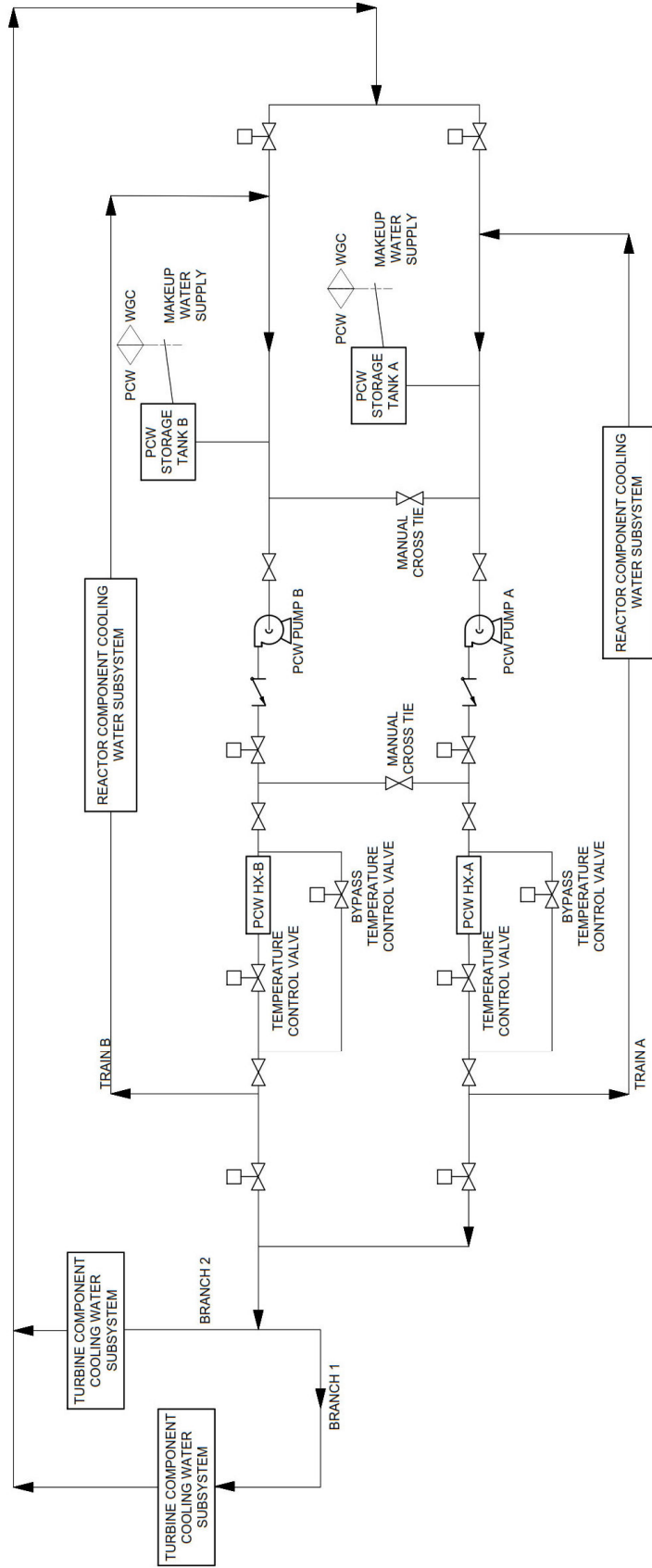


Figure 9A.2.1-1: Plant Cooling Water

9A.2.2 Reactor Water Cleanup System

The Reactor Water Cleanup System (CUW) provides blowdown-type cleanup flow for the Reactor Pressure Vessel (RPV) during the reactor power operating mode. Cleanup or filtration and ion removal is performed by the CFD (Chapter 10, Subsection 10.3.1). The CUW provides an overboarding flow path to the condenser hotwell (condensate pump suction) or Liquid Waste Management System (LWM) (Chapter 11, Section 11.2) directly from the RPV lower region. CUW piping can be utilized to reduce reactor temperature stratification with reverse flow from the Shutdown Cooling System (Subsection 9A.2.3).

9A.2.2.1 System and Equipment Functions

Refer to Chapter 1, Section 1.8 for a description of plant operating modes.

The CUW performs the following functions during normal and off-normal conditions.

9A.2.2.1.1 Normal Functions (Non-Safety-Category)

- Mode A
 - Normal Reactor Water Cleanup (1% Flow) Plant mode 1, 2, 3, 4)
- Mode B – Overboarding
 - Mode B1-A – Overboarding from Nuclear Boiler System (NBS)(Chapter 5) to Main Condenser and Auxiliaries (MCA) (Chapter 10, Section 10.5) (Plant mode 1, 2, 3, 4)
 - Mode B1-B – Overboarding from NBS to Liquid Waste Management System (LWM) (Chapter 11, Section 11.2) (Plant mode 1, 2, 3, 4)
 - Mode B2-A – Overboarding from SDC through CUW to MCA (Plant mode 2, 3, 4, 5)
 - Mode B2-B – Overboarding from SDC through CUW to LWM (Plant mode 2, 3, 4, 5)
- Mode C
 - RPV thermal stratification reduction (plant mode 2)

Overboarding

During overboarding, the heat exchanger is in service to cool the reactor water to minimize flashing and two-phase flow in the pressure reducing components and downstream piping. Flow can be directed to the condenser hotwell, or in the event high radiation is detected, the overboarding flow is manually shifted to the LWM.

Thermal Stratification Reduction

The system is designed to provide flow through the bottom head connections in the CUW during startup operations to reduce thermal stratification caused by the continuous input of cold Control Rod Drive (CRD) flow through the control rod drives.

Cleanup Flow to Condenser

During Plant Operational Modes 1, 2, 3, and 4, CUW provides the equivalent of one percent of Feedwater (FW) nominal flow to downstream of a condensate pump for remixing and filtration.

9A.2.2.1.2 Normal Function (Safety-Category)

To ensure the safety of the plant, the system continuously monitors for leakage utilizing density compensated differential flow measurements. If a leak is detected, the containment isolation

function isolates the system from the Reactor Coolant Pressure Boundary (RCPB) through the closure of the containment isolation valve.

9A.2.2.1.3 Off-Normal Functions (Non-Safety Category)

The CUW system does not perform any Non-Safety-Category functions during off-normal conditions.

9A.2.2.1.4 Off-Normal Functions (Safety Category)

The CUW system performs Leak Detection and Isolation functions in Off-Normal conditions. Off-Normal functions for leak detection are the same as described in Subsection 9A.2.2.1.3.

Containment isolation is provided in accordance with REGDOC 2.5.2, Section 8.6.6 (Reference 9A.2.2-1). Leak detection is provided in accordance with Chapter 6, Subsection 6.3.2.8. Chapter 6, Subsection 6.3.4 describes the BWRX-300 containment isolation and containment isolation valves.

9A.2.2.2 Safety Design Bases

The following represents the CUW safety design bases:

1. DL3 leak detection actuates CUW isolation on CUW line break indication in Modes 1, 2, 3, and 4.
2. As part of DL3, the CUW provides a containment isolation valve (Safety Class 1) on piping that penetrates the containment boundary. The containment isolation valves are designed to close upon receiving an isolation signal from Safety Class 1 Instrumentation and Control System.
3. As part of DL3, upon detecting a CUW line break in Modes 1, 2, 3, and 4 the Safety Class 1 Instrumentation and Control System actuates CUW isolation. The CUW isolation valves are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System.
4. Defense Line 4a (DL4a) leak detection actuates CUW line isolation on break indication in the CUW lines in Modes 1, 2, 3, and 4. Upon detecting a CUW line break in modes 1, 2, 3, and 4 the Safety Class 2 and 3 Instrumentation and Control System actuates CUW isolation. The CUW isolation valves are designed to close upon receiving an isolation signal from Safety Class 2 and 3 Instrumentation and Control System.

9A.2.2.3 Description

Figure 9A.2.2-1 depicts a simplified flow diagram of the CUW and interfacing systems.

The CUW system consists of one train. The CUW receives flow through two nozzles located on the RPV. The train's inlet is independently connected to RPV penetrations located at about the mid-vessel height which take inlet flow from nozzles located near the RPV bottom head. This piping up to the RPV isolation valves are a part of the NBS. The inlet piping connects to the reactor vessel and combines inside containment to form one discharge line. This line is provided with a containment isolation valve where it penetrates containment. This valve will receive a signal from the CUW leak detection system and will close upon a detected leak. CUW continues through a regenerative heat exchanger and a pressure reduction station. The heat exchanger and pressure reduction station are designed to condition the water to acceptable temperatures and pressures for processing to the condensate system or overboarding. The heat exchanger is designed to recover heat back to the condensate and feedwater system. Discharge piping is

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connected either to a condensate line for the normal CUW function or routed to the condenser hotwell or LWM for overboarding.

The leak detection and isolation subsystem perform both a DL3 and DL4a function to identify leaks and line breaks in the CUW and to initiate automatic response actions by other systems. For DL3, in MODES 1-4, the detection of a leak or line break in the CUW initiates isolation of the CUW system. For DL4a, in MODES 1-4, the detection of a leak or line break in the CUW initiates isolation of the CUW system.

The leak detection and isolation subsystem utilize a single Flow Element (FE) in the intake flow path and a single FE in each discharge flow path. Each FE serves independent, redundant sets of divisional Flow Transmitters (FTs), three FTs each for the DL3 and the DL4a line break detection functions. The basis for this arrangement is that a leak or break near the end of a discharge flow path may not be detectable by a single FE/FT feature located in the intake location of the system due to the physical distance between those locations. The design is intended to ensure that a leak or break can be identified by any of the FE/FTs at a detection point, or by differential flow between an upstream and downstream FE/FT detection point. Hence the use of detection features at the intake and at the discharge flow paths allows for detection of differential flow between the intake and any discharge flow path, or detection of a leak or break within detection proximity of any single FE.

The FT readings are density compensated by sets of three independent, redundant SC1 Temperature Transmitters. This feature will account for density changes in the water across the heat exchanger that is between the FE located in the intake flow path and the FEs located in each discharge flow path. Density compensation is used to avoid spurious isolation of the CUW and spurious generation of a signal that would initiate unwanted response actions by other systems.

Environmental Qualification of Safety-Class SSC's is provided as applicable as discussed in Chapter 3, Section 3.9.

9A.2.2.3.1 Component Description

The following provides description of CUW components.

Piping and Valves

The system piping is sized to ensure it can withstand the maximum pressure and temperature combination while keeping the maximum flow velocities within the acceptable limits.

The piping and the wetted parts of the valves are stainless steel to help with ALARA (Chapter 12, Section 12.3) concerns since this system is in contact with potentially radioactive water. Smooth bends are used instead of welded fittings where practical. At certain locations within the system, stainless steel pipes with electro-polished finish are used to reduce corrosion product buildup.

All valves in the system are used for isolation (on/off) except for a throttling valve in the pressure reduction station. This valve is used to throttle the system to achieve desired flow rates.

The CIVs for CUW fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.

- Overpressure protection is provided per the applicable design code, as follows:
 - ASME Boiler and Pressure Vessel Code, Section III for Class 1, 2, and 3 piping and components
 - ASME B31.1, Power Piping for Non-Safety-Category function piping
 - ASME BPVC, Section VIII for pressure vessels

Radioactive equipment and piping are designed to minimize crud buildup and to provide for easy decontamination and maintenance.

Regenerative Heat Exchanger

The CUW heat exchanger is a regenerative heat exchanger cooled by the Condensate and Feedwater Heating System flow (Chapter 10, Subsection 10.3.2). The heat exchanger is designed to reduce the temperature of the inlet flow. Most of the heat of the CUW water is transferred to the condensate returning to the reactor, thus losing almost no thermal value.

The CUW heat exchanger is designed, manufactured, installed, and tested in accordance with applicable codes and standards, including Thermal Exchanger Manufacturers Association, API-662, ASME BPVC, Section VIII, Division 1. The heat exchanger tubes are seamless and constructed of stainless steel or titanium to minimize the possibility of failure and reduce maintenance requirements. The heat exchanger is designed with provisions for tube replacement as well as individual tube plugging. An excess of tubes factored into the design of the heat exchanger accommodates tube plugging.

System Pressure Reduction Station

At the pressure reduction station, a breakdown orifice (or pressure reduction valve) is used to reduce pressure before the water enters the CFS downstream of the condensate pumps, where it is mixed with the condensate and cleaned in the CFD cleanup filters and demineralizers. This pressure is greater than condensate pressure at the connection point to ensure positive flow. The breakdown orifice (or pressure reduction valve) and associated instrumentation are located downstream of the heat exchanger and upstream of the condensate pump discharge tie-in location in the TB.

9A.2.2.4 Materials

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from IGSCC (as applicable) through material chemistry, heat treatment, contamination and material processes controls.

9A.2.2.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.2.2-1 which provides the description and boundary for each interfacing system.

9A.2.2.6 System and Equipment Operation

9A.2.2.6.1 Initial Configuration (Pre-Startup)

The CUW is initially idle with the system isolation valves closed. The CUW is shutdown water solid, but because it is idle for long periods, the system pressure may need to be checked to ensure there was no leakage while shutdown. If necessary, the system piping is filled and vented to ensure there is no water hammer upon CUW startup.

9A.2.2.6.2 System Startup

When the CUW is actuated from the main control room, it first enters a stage of pre-warming. The system SSC are at ambient temperature due to inactivity in Plant Operating Mode 1. The water coming from the RPV is expected to be much warmer; therefore, this pre-warming stage is important to allow the system to gradually heat up to prevent over stressing the components due to thermal growth. The pre-warming is controlled by slowly initiating flow through the CUW through the use of the flow control valve and eventually ramping up to full flow before transitioning to normal operations.

9A.2.2.6.3 Normal Operations

Operations under normal conditions are described below. Table 9A.2.2-2 presents the plant operating Modes and corresponding CUW operational modes.

Mode A – Power Operation:

During power operation (plant Mode 1), the CUW system is in-service, taking input from the bottom of the RPV through two inlet lines. Water flows through piping internally mounted inside the RPV downcomer region and exits the vessel about four metres above top of active fuel. At the top of the piping outside the RPV, there are double isolation valves integral to the vessel which close to prevent RPV inventory from exiting the CUW line in case of a pipe break. Except for the containment isolation and leak detection related functions, the CUW is an SCN system, and all other system functions defined below are SCN.

The two inlet lines join into a single line outside the RPV where it exits the containment vessel, routes through piping in the steam tunnel and into the Turbine Building. The system contains a regenerative heat exchanger in which the CUW water is cooled by the Condensate and Feedwater Heating System. Most of the heat of the CUW water is transferred to the condensate returning to the reactor. A breakdown orifice is used to reduce pressure before the water enters CFS downstream of the condensate pumps where it is mixed with the condensate and cleaned in the CFD (Chapter 10, Subsection 10.3.1) cleanup filters and demineralizers.

During plant mode 2, 3, & 4 CUW can operate in Mode A. However, due to relying on the reactor for motive pressure for flow through CUW, sufficient reactor pressure is required to overcome system flow resistance for CUW flow to be achieved, with flow being maintained through throttling as reactor pressure increases.

Mode B1-A:

When there is sufficient pressure in the RPV for flow in CUW, CUW can overboard to the condenser hotwell. In this mode CUW operates the same as Mode A, with the exception of flow being diverted to the condenser hotwell. This function is available in plant modes 1, 2, 3, & 4.

Mode B1-B:

When there is sufficient pressure in the RPV for flow in CUW, CUW can overboard to LWM. In this mode CUW operates the same as mode A, except for flow being diverted to LWM. This function is available in plant Modes 1, 2, 3, & 4.

Mode B2-A:

When there is insufficient pressure in the RPV for flow in CUW, SDC can overboard through the CUW to the condenser hotwell. SDC interfaces with CUW in the steam tunnel, and continues on the same path as described in mode B1-A. Mode B2-A is available in plant modes 2, 3, 4, & 5.

Mode B2-B:

When there is insufficient pressure in the RPV for flow in CUW, the SDC can overboard through the CUW to LWM. SDC interfaces with CUW in the steam tunnel, and continues on the same path as described in mode B1-B. Mode B2-B is available in plant modes 2, 3, 4, & 5.

Mode C:

To help reduce vessel stratification, caused by continuous input of cold CRD flow through the control rod drives, while shut down and during preparation for startup, the SDC water can be routed to the RPV lower region through the normal CUW inlet lines. The SDC thermal stratification

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reduction line interfaces with the CUW system between the regenerative heat exchanger and the isolation containment valve. Mode C is only available in plant mode 2.

Mode 1 – Power Operation:

During power operation, the CUW system is in service in system Mode A, with flow from the bottom of the RPV through two inlet lines. Water flows through piping internally mounted inside the RPV downcomer region and exits the vessel about four metres above top of active fuel. At the top of this piping outside the RPV, there are double isolation valves integral to the vessel, which prevents RPV inventory from exiting the CUW line in case of a pipe break.

The two inlet lines join into a single line outside the RPV, and the CUW water then is routed through piping in the steam tunnel. There is a regenerative heat exchanger in which the CUW water is cooled by the CFS. Most of the heat of the CUW water is transferred to the condensate returning to the reactor, thus losing almost no thermal value. A breakdown orifice is used to reduce pressure before the water enters CFS downstream of the condensate pumps, where it is mixed with the condensate and cleaned in the CFD cleanup filters and demineralizers.

Mode 2 – Startup:

During startup Mode, CUW operation is the same as during power operation. However, due to relying on reactor pressure for flow through CUW, sufficient reactor pressure is required to overcome system flow resistance for CUW flow to be achieved, with flow increasing as reactor pressure increases.

During startup or shutdown, CUW can function to let down (overboard) the excess reactor inventory if there is enough pressure (CUW Mode B1-A and B1-B). If not, the SDC system can be used to let down the excess reactor inventory (CUW Mode B2-A and B2-B). The SDC letdown line interfaces with the CUW system upstream of the regenerative heat exchanger. Overboard flow can be directed to the condenser hotwell or LWM. In this mode, reactor vessel water level is maintained by the control rod drive water.

To help reduce vessel stratification during preparation for startup, SDC water can be routed to the RPV lower region through the normal CUW inlet lines (CUW Mode C).

Mode 3 – Hot Shutdown:

During hot shutdown operation, the CUW system is in-service if the RPV is pressurized. This mode is with reactor temperature $>215.6\text{ }^{\circ}\text{C}$. Vessel letdown can be performed as discussed above under the startup mode.

Mode 4 – Stable Shutdown:

During stable shutdown operation, the CUW system is isolated from the reactor. This mode is with reactor temperature $\leq 215.6\text{ }^{\circ}\text{C}$ and $>93.3\text{ }^{\circ}\text{C}$. If CUW is isolated at the vessel, SDC can be used via CUW if overboarding is required.

Mode 5 – Cold Shutdown:

During cold shutdown operation, the CUW system is isolated at the vessel. This mode is with reactor temperature $\leq 93.3\text{ }^{\circ}\text{C}$. SDC can be used via CUW if overboarding is required.

Mode 6 – Refueling:

During refueling mode, the CUW is not in service.

9A.2.2.6.4 Off-Normal Operations

No manual operator actions are required to initiate or assure the completion of fundamental safety functions. The RPV isolation valves and containment isolation valve fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.

For large line breaks, it is noted that for FW line breaks and reactor water cleanup breaks outside of containment, leak detection systems identify the break condition and provide signals to close the affected system isolation valves.

9A.2.2.6.5 System Shutdown

The system is shut down in plant operating Mode 6.

Upon shutdown, the system isolation valves are closed. The system remains water solid while shutdown.

9A.2.2.7 Instrumentation and Control

During plant operation and after plant shutdown (with sufficient reactor pressure), CUW removes water from the bottom of the reactor and routes it to the condensate system or overboards excess "swell", when required, to the condenser hotwell or liquid radwaste. These functions require both monitoring and control.

The input signals to the control functions are triply redundant and the actuators receive two out of three voted commands from the CUW controller. The CUW controller is triply redundant and is dual ported to the plant balance of plant segment network. As with all Triple Modular Redundant (TMR) controllers, extensive hardware and software diagnostics are provided for operator monitoring and alarms. The CUW controller is also designed to support plant automation. Refer to Chapter 7 for information pertaining to TMR.

The CUW can be controlled from the main control room. The CUW controller can receive commands from the Plant Automation System (PAS) or be manually operated. The controller automates the operation of the system.

9A.2.2.7.1 Flow

There is one SC1 flow element along with triply redundant (SC1) transmitters on the system supply line at the containment boundary inside the steam tunnel. This flow indication is used in conjunction with identical flow elements and transmitters located at the end of each flow path. When used as a pair they monitor for system leakage. Main Steam Reactor Isolation Valves (MSRIV) and CUW Isolation are actuated on CUW line break indication in Plant Modes 1-4.

There are triply redundant SC2 transmitters independent from the SC1 flow element on the system supply line at the containment boundary. This flow indication is used in conjunction with identical flow elements and transmitters located at the end of each flow path. When used as a pair they monitor for system leakage. MSRIV Isolation and CUW Isolation are actuated on CUW line break indication in Plant Modes 1-4.

There are triply redundant SCN transmitters used for the control of the pressure reduction station.

9A.2.2.7.2 Temperature

There are triply redundant SC1 temperature sensors that accompany the SC 1 flow transmitters on the containment boundary and at the end of each flow path. These temperatures are used to provide input for the density compensation of the flow transmitters.

There are triply redundant Safety Class 2 (SC2) temperature sensors that accompany the SC1 flow transmitters on the containment boundary and at the end of each flow path.

There is one SCN temperature sensor downstream of the heat exchanger. This temperature is measured prior to entrance of the pressure reduction station to ensure that the temperature of the fluid is acceptable after the heat exchanger.

9A.2.2.7.3 Pressure

Pressure Reduction Station Inlet:

One set of triply redundant SCN pressure instruments are provided at the inlet of the pressure reduction station. These pressure instruments are used to control the pressure reduction station.

Pressure Reduction Station Outlet:

Three SCN pressure instruments are provided at the discharge of the pressure reduction station. These pressure instruments are used to monitor the discharge pressure of the pressure reduction station, to monitor status and performance.

9A.2.2.7.4 Radiation

CUW has a radiation monitor downstream of the heat exchanger provided by the PREMS. This monitor provides input to the Process and Radiation Monitoring Subsystem.

9A.2.2.8 Monitoring, Inspection, Testing and Maintenance

Maintenance and testing support equipment reliability. SSCs are designed to facilitate operation and maintenance. Maintenance activities, including post-maintenance and post-modification testing, are controlled by plant procedures. Maintenance activities restore SSCs to their original condition or involve a temporary alteration in accordance with plant procedures.

Maintenance practices consider industry best practices such as Institute of Nuclear Power Operations (INPO, etc.) and operating experience, and conform with plant safety requirements to minimize potential for personnel injury. Maintenance activities comply with Technical Specifications as required. Maintenance activities implement plant ALARA practices to minimize work activity dose. Maintenance activities involving critical plant equipment may require involvement of vendors or industry specialists.

The CUW is provided with chemical cleaning and decontamination connections that can utilize condensate to flush piping and equipment prior to maintenance to provide decontamination of the regenerative heat exchanger, pressure reduction station, and associated valves.

Inspections, checks, and tests conducted during the performance of maintenance activities are considered "In-Process Maintenance Tests" and are governed by the in-use maintenance procedures and processes.

Inspections, checks, and tests conducted following the conclusion of maintenance activities are considered "Post-Maintenance Tests" and are governed by plant Post-Maintenance Test procedures. ISI and IST requirements are established and include inspection/test frequency for SSCs. Remote monitoring of key parameters assists in the trending of component performance as part of condition-based predictive maintenance.

9A.2.2.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions. To control and

minimize personnel exposure, radioactive equipment is designed to minimize crud buildup and provide for decontamination and maintenance.

9A.2.2.10 Performance and Safety Evaluation

System reliability is enhanced through the use of periodic system testing, preventive maintenance, and application of redundant instrumentation.

The CUW SC1 functions during normal conditions and off-normal conditions include continuous monitoring for leakage utilizing density compensated differential flow measurements. If a leak is detected, in either normal or off-normal conditions the system is isolated from the reactor coolant pressure boundary using system isolation and containment isolation valves.

The arrangement of SSCs minimize the possibility of compromising the functionality of both the SC1 functions and their backup systems and equipment for any credible events that could cause damage to one region of the plant. The portions of the CUW responsible for performing containment isolation are located in the Reactor Building which provides protection against natural phenomena ensuring the ability of the CUW to perform its Safety-Category functions.

9A.2.2.11 References

- 9A.2.2-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

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Table 9A.2.2-1: Reactor Water Cleanup System Interfaces

Interfacing System	Interface Description	Interface Boundary
Nuclear Boiler System	Locations where flow from the Reactor Pressure Vessel enter the CUW. The RPV isolation valves for reactor water cleanup will fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means	Physical boundary is at the dual NBS reactor isolation valves on two penetrations exiting the RPV
Shutdown Cooling System	Connection from SDC to CUW for overboarding flow at low pressures or flow to the vessel to reduce temperature stratification	The system interface is at the isolation valve between SDC and CUW
Condensate and Feedwater Heating System	Normal CUW flow to Condensate System. All cleanup of CUW injection flow is performed by CFD filters and demineralizers. Coolant flow for CUW regenerative heat exchanger from Feedwater	Downstream of a Condensate pump prior to filters and demineralizers CUW regenerative heat exchanger
Plant Pneumatics System	Provide air or nitrogen to operate air operated valves	The system interface is located at each valve actuator
Safety Class 2 and 3 Electrical Distribution System	Provide power to CUW instrumentation for system operation and line break detection	The system interface is at each powered component
Safety Class 1 Instrumentation and Control System	CUW instrumentation for leak detection provide input to Safety Class 1 Instrumentation and Control System. Safety Class 1 Instrumentation and Control System is the safety class instrumentation system	
Safety Class 2 and 3 Instrumentation and Control System	CUW instrumentation for system operation and leak detection provide input to Safety Class 2 and 3 Instrumentation and Control System	
Liquid Waste Management System	Overboarding to LWM to reduce vessel level if required and activity (radiation level) in the process flow may be high	The system boundary is after the overboard isolation valve

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Interfacing System	Interface Description	Interface Boundary
Main Condenser and Auxiliaries	Overboarding to MCA to reduce vessel level if required and process flow activity (radiation level) is normal	The system boundary is after the overboard isolation valve
Process Radiation and Environmental Monitoring System	Monitor radiation levels of cleanup flow Process sampling for radiological and chemistry control	Radiation monitor/transmitter on process flow line PREMS isolation valve

Table 9A.2.2-2: Reactor Water Cleanup System Operating Modes

Mode	Title	Reactor Mode Switch Position	CUW System Modes
1	POWER OPERATION (10% - 100% Reactor Power)	RUN	Mode A, B1-A, and B1-B
2	STARTUP	REFUEL ⁽¹⁾ or STARTUP	All Modes
3	HOT SHUTDOWN ⁽¹⁾	SHUTDOWN	Mode B1-A, B1-B, B2-A, and B2-B
4	STABLE SHUTDOWN ⁽¹⁾	SHUTDOWN	Mode B1-A, B1-B, B2-A, and B2-B
5	COLD SHUTDOWN ⁽¹⁾	SHUTDOWN	Modes B2-A and B2-B
6	REFUELING ⁽²⁾	SHUTDOWN or REFUEL	NA

(1) All reactor pressure vessel head closure bolts fully tensioned

(2) One or more reactor pressure vessel head closure bolts less than fully tensioned

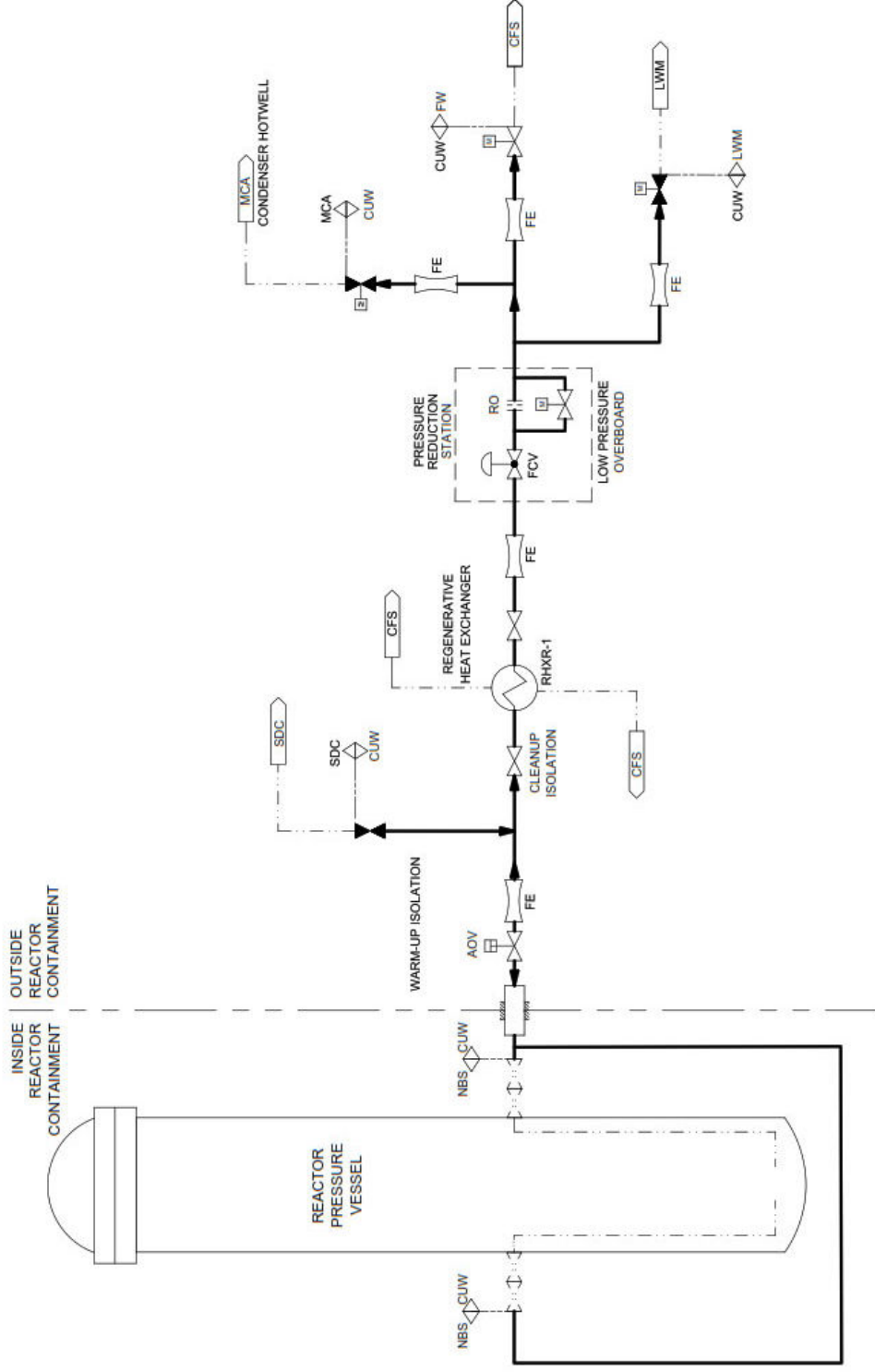


Figure 9A.2.2-1: Reactor Water Cleanup System

9A.2.3 Shutdown Cooling System

The SDC provides for decay heat removal when shutting down the plant. The system is also used to reduce reactor pressure vessel inventory and can be used in conjunction with CUW (Subsection 9A.2.2) piping to reduce reactor pressure vessel thermal stratification.

9A.2.3.1 System and Equipment Functions

9A.2.3.1.1 Normal Functions (Non-Safety-Category)

Overboarding

During reactor startup, it is necessary to remove the CRD purge water injected into the RPV as well as the excess reactor water volume arising from thermal expansion due to reactor heat-up. The SDC accomplishes these volume removals and thereby maintains proper reactor level until the reactor pressure is sufficient to allow use of the CUW overboard flow path.

During overboarding, the heat exchanger is in service to cool the reactor water to minimize flashing and two-phase flow in the downstream piping. The preferred overboarding destination is the hotwell that is part of the CFS (Chapter 10, Subsection 10.3.2); however, in the event high radiation is detected, the overboarding flow is manually shifted to the LWM (Chapter 11, Subsection 11.3).

Overboarding is considered part of SDC Mode B1 and is available in plant operating Modes 2-6.

RPV Thermal Stratification Reduction

The SDC is designed to provide flow through the CUW inlet lines into the bottom head region of the RPV during startup operations to reduce thermal stratification caused by the continuous input of cold CRD flow through the control rod drives.

Thermal stratification reduction is considered part of SDC Mode B2 and is available in plant operating Mode 2.

9A.2.3.1.2 Normal Functions (Safety-Category)

The SDC system for the BWRX-300 has two Safety Class functions:

Decay Heat Removal (Defense Line 2/Safety Class 3)

In conjunction with the heat removal capacity of either the main condenser and/or the isolation condensers, the SDC is used to reduce the RPV pressure and temperature during cooldown operation from the rated design pressure and temperature to below saturation temperature at atmospheric pressure in less than one day.

The shutdown cooling function of the SDC provides decay heat removal capability at normal reactor operating pressure as well as at lower reactor pressures. The redundant trains of SDC permit shutdown cooling even if one train is out of service; however, cooldown time is extended when using only one train.

Decay heat removal is considered part of SDC Mode A1 and is available in plant operating Modes 2-6.

Leak Detection and Isolation (Defense Line 3/Safety Class 1 and Defense Line 4a/Safety Class 2)

When the SDC is operational, the system continuously monitors for leakage utilizing differential flow measurements. If a leak is detected, the system is isolated from the reactor through a

number of system isolation and containment isolation valves to prevent the excessive reduction of reactor water inventory.

9A.2.3.1.3 Off-Normal Functions (Non-Safety Category)

If feedwater flow is lost in operating Modes 2-4, the operator initiates either the SDC or ICS (Chapter 5, Section 5.8) to reduce reactor water temperature and pressure.

Upon identification of a small line break inside containment with or without preferred power during operating Modes 3-4, the SDC is initially used for decay heat removal until the containment is isolated at which time the ICS is initiated to remove the decay heat.

9A.2.3.1.4 Off-Normal Functions (Safety-Category)

The system does not perform any safety category functions during off-normal conditions.

The design of the SDC system meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.6.6 (Reference 9A.2.3-1) as related to the design of containment isolation valves. Refer to Chapter 6, Subsection 6.3.4 for a discussion related to BWRX-300 containment isolation and containment isolation valves.

9A.2.3.2 Safety Design Bases

The SDC has two Safety-Class functions, Decay Heat Removal, and Leak Detection and Isolation. Refer to Subsection 9A.2.3.1.2 for discussion of SDC Safety-Category functions.

Plant systems retain sufficient heat removal capacity to prevent boiling in the reactor during periods of plant shutdown with the Reactor Pressure Vessel head removed, and to simultaneously prevent boiling in the fuel pool, assuming a Loss of Preferred Power event and the most limiting single active failure.

Defense lines associated with Leak Detection and Isolation are noted below.

1. As part of Defense Line 3 (DL3) 3, the Shutdown Cooling isolation valves are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System.
2. As part of DL4a, the SDC isolation valves are designed to close upon receiving an isolation signal.
3. As part of DL2, the SDC is designed to initiate upon receiving an initiation signal from Safety Class 2 and 3 Instrumentation and Control System.
4. DL2 operators are able to manually initiate SDC train in Modes 2-6.
5. DL3 leak detection actuates MSRV Isolation on FW and SDC Line break indication in Modes 1, 2, 3, and 4.
6. DL3 leak detection actuate FW and SDC line isolation on FW and SDC line break indication in Modes 1, 2, 3, and 4.
7. DL4a Leak Detection actuates MSRV Isolation on FW line break indication / SDC Line break indication in Mode 1-4.
8. DL4a Leak Detection actuates FW Isolation/SDC line isolation on FW line break indication/SDC Line break indication in Mode 1-4.

9A.2.3.3 Description

The SDC comprises two independent pump and heat exchanger trains. These trains together provide redundant decay heat removal capacity such that each train is designed to remove 100%

of decay heat as soon as 4 hours after reactor shutdown. The major components of each train are a pump and a Heat Exchanger (HX), along with valves, piping, instrumentation and controls, and power inputs. The two trains operating in parallel provide the systems full rated shutdown cooling performance. Bypass lines and valves are included around the tube side of each HX to allow bypassing of the HX for SDC functions such as reducing RPV thermal stratification.

Each train's suction is independently connected to a separate ICS (Chapter 5, Section 5.6) condensate return line outside of containment, and downstream of the ICS containment isolation valves. Each SDC train's return piping is independently connected to a separate CFS (Chapter 10, Subsection 10.3.2) feedwater line outside of the containment isolation valves. This arrangement helps to prevent short circuiting of the flow inside of the RPV since the SDC suction is taken from the ICS return nozzle which originates in the chimney region over the core and the SDC return flow utilizes the CFS return which terminates outside this region in the downcomer area.

Fittings are registered in compliance with Technical Standards and Safety Authority.

Refer to Chapter 3, Section 3.9 for information pertaining to equipment qualification of BWRX-300 SSCs.

9A.2.3.3.1 Decay Heat Removal Subsystem

The subsystem comprises the flow path from the ICS to the SDC pump, through the tube side of the SDC heat exchanger, and then returned to the RPV through the CFS as depicted in Figure 9A.2.3-1. The subsystem is available in Plant Modes 2 through 6, though the condenser and ICS are expected to be used initially to cool the reactor after shutdown. The subsystem is initially operated at low flow to allow the system Structures, Systems, and Components, to be brought up to operating temperature without overstressing the components. Once the system is up to temperature, the flow rate increases and is controlled to maintain PCW (Subsection 9A.2.1) heat exchanger cooling water exit temperature within the interface requirements of 54.4 °C for two SDC trains running and 60.0 °C for one train running.

9A.2.3.3.2 Overboarding Subsystem

The overboarding subsystem supports RPV level control during Plant Modes 2 through 6 when the fuel pool gate is installed and the CUW (Subsection 9A.2.2) is not available for use. The subsystem comprises the flow path from the ICS to the SDC pump, through the tube side of the SDC heat exchanger, and then through the overboard flow path (see Figure 9A.2.3-2). The figure shows multiple overboard flow pathways highlighted; however, only one overboard path is implemented at a time by controlling the position of the valves on each flow path. The SDC trains share the overboard control valves and each train has its own isolation valve to prevent inadvertent overboarding. Each overboard flow path has been designed to provide adequate overboarding flow to support the worst-case scenario (fastest) heat-up rate of the reactor combined with the expected water input from the CRD (Chapter 4, Section 4.5).

The preferred overboard flow path is from the SDC to the CFS hotwell; however, in the event that high radiation is detected in the effluent, the flow path is manually switched from the CFS to the LWM system. The overboarding flow pathways through the CUW are utilized in the event of failure of any of the other overboard pathway isolation valves, or if the pressure in the SDC is too high for the interfacing system, in which case, the CUW pressure reduction station is utilized to reduce the fluid pressure to protect the interfacing system.

Each overboard flow pathway from the SDC has two fail-closed valves in series. There are two valves in series to prevent exposing a lower design pressure system or subsystem to a higher design pressure system in the event of an inadvertent opening of a single valve.

9A.2.3.3.3 Thermal Stratification Reduction Subsystem

The subsystem comprises the flow path from the ICS to the SDC pump, through the SDC heat exchanger bypass line, and then through the overboard flow path through the CUW and back to the RPV through the NBS as depicted in Figure 9A.2.3-3. The subsystem provides approximately 200 gpm to the bottom region of the RPV to counter act the constant 20 gpm flow from the CRD through the control rod drives. The 10:1 flow ratio is expected to provide sufficient heat to overcome the cooling effect of the CRD flow. Also, the velocity at the discharge of the piping into the RPV is sufficient to promote mixing in the bottom head region to minimize thermal stratification.

9A.2.3.3.4 Leak Detection and Isolation Subsystem

The leak detection and isolation subsystem are comprised of three flow elements per train along with two sets of triply redundant transmitters per flow element to support both DL3 and DL4a leak detection requirements. The readings are utilized to account for temperature change across the heat exchanger. This is done to avoid spurious isolation of the SDC which would reduce the SDC capacity, or if both trains are isolated, would require the restart of ICS. Restart of ICS is only possible if the reactor is still pressurized.

The control system compares the flow readings from the supply flow transmitters against flow readings from the return and overboard flow transmitters. If the system detects a flow differential greater than the acceptable leakage value, the control system actuates MSRIV Isolation, Feedwater Isolation, and SDC Isolation.

9A.2.3.3.5 Component Description

Shutdown Cooling System Piping and Valves

The SDC piping material is 304L stainless steel. The pipe wall thickness has been sized to ensure it can withstand the maximum pressure and temperature combination while the diameter has been chosen to ensure the maximum flow velocities stay within the acceptable limits.

The piping and the wetted parts of the valves are stainless steel which supports ensuring ALARA goals since this system is in contact with potentially radioactive water.

All valves in the system are used for isolation (on/off) except for the heat exchanger bypass valve. The bypass valve is used to throttle the bypass around the heat exchanger and can be used in conjunction with the SDC pump speed control to control the exit temperature of the heat exchanger.

SDC decay heat removal piping and valves are designed, manufactured, and tested in accordance with ASME BPVC Section III, Class 3. SDC overboard piping and valves are designed, manufactured, and tested in accordance with ASME B31.1.

Shutdown Cooling System Pumps and Adjustable Speed Drives

Each train of the SDC contains one SC3 horizontal centrifugal pump sized to allow sufficient flow to remove 100% of the decay heat generated by the core 4 hours after shutdown. The SDC pumps are designed to pump the maximum flow required by the system to meet SDC cooldown requirements for decay heat removal. The pump is installed below the RPV inlet nozzle which provides adequate NPSH at maximum flow conditions. Each pump is paired with an adjustable speed drive to allow for control of the cooldown rate and the reactor temperature during shutdown.

The adjustable speed drive is necessary to allow the flow to be adjusted to meet all expected system flow design points.

SDC pumps are designed, manufactured, and tested in accordance with ASME BPVC Section III, Class 3. SDC pump wetted parts are manufactured from stainless steel.

Shutdown Cooling System Heat Exchangers

Each train of the system contains one SC3 shell and tube heat exchanger sized with sufficient capacity to remove 100% of the decay heat generated by the core 4 hours after shutdown. The heat is transferred from the SDC process fluid to the PCW.

SDC heat exchangers are designed, manufactured, installed, and tested in accordance with ASME BPVC Section VIII, Class 3. SDC heat exchanger tubes are constructed of stainless steel.

9A.2.3.4 Materials

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from IGSCC through material chemistry, heat treatment, contamination, and material processes controls.

9A.2.3.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.2.3-1 for SDC interfaces with other equipment or systems.

9A.2.3.6 System and Equipment Operation

Plant cooldown following shutdown is accomplished using a combination of the main condenser, ICS, and SDC systems. Although the SDC system is qualified for high temperature and pressure operation, the main and ICS condensers are the preferred cooldown sources immediately following shutdown.

9A.2.3.6.1 Initial Configuration (Pre-Startup)

The system is initially idle with the system isolation valves closed. When the system is shutdown, it is water solid, but because it is idle for long periods, the system pressure may need to be checked to ensure there was no leakage while shutdown. If necessary, the system piping is filled and vented to ensure there is no water hammer upon system startup.

9A.2.3.6.2 System Startup

When the system is actuated from the control room, it first enters a stage of pre-warming. It is expected that the system SSCs are at ambient temperature due to inactivity in Plant Mode 1, but the water coming from the RPV is expected to be much warmer; therefore, this pre-warming stage is important to allow the system to gradually heat up to prevent over stressing the components due to thermal growth. The pre-warming is controlled by initiating low flow through the SDC using the adjustable speed drive and eventually ramping up to full flow before transitioning to normal operations.

9A.2.3.6.3 Normal Operations

The normal and off-normal operational modes of the SDC are described below. This includes how the plant transitions from a shutdown condition to full power, how steady-state full power operation is maintained, how the transition from full power to shutdown is achieved, and how a refueling outage is performed. Table 9A.2.3-2, "BWRX-300 Mode Table" lists the plant operational modes and the corresponding operational modes of SDC. The SDC system modes and functions are discussed in more detail below.

Plant Mode 1 - Power Operation

During power operation, the SDC is not in-service.

Plant Mode 2 - Startup Mode

The SDC is a manually initiated system.

During startup operation, the SDC is initially in-service to provide a reactor water reject flow path for RPV level control. The SDC is eventually taken out of service when the RPV pressure is high enough to establish CUW flow and when the SDC has inadequate net positive suction head due to voiding in the chimney section.

If the SDC is required to reduce vessel temperature stratification, SDC Mode B2 is utilized. After flow from SDC enters the CUW it is routed reverse the normal flow direction and discharges from the CUW inlet lines into the RPV bottom head region.

Plant Mode 3 - Hot Shutdown Operation

During hot shutdown operation, the SDC is not normally in-service since heat removal by steam condensation is more efficient. The SDC requires adequate water level in the chimney after shutdown before it is placed into service.

Plant Mode 4 - Stable Shutdown Operation

During hot shutdown operation, the SDC is not normally in-service since heat removal by steam condensation is more efficient. The SDC requires adequate water level in the chimney after shutdown before it is placed into service.

When RPV pressure drops to a point that the normal reactor water reject flow path is ineffective, the SDC can be placed into service to provide this function through an interface connection to the CFS and LWM.

Plant Mode 5 - Cold Shutdown Operation

During cold shutdown operation, the SDC is in-service. Each train of SDC can separately remove 100% of the decay heat generated 4 hours following reactor shutdown. Both trains of the SDC are normally available in cold shutdown conditions, but one train may be taken out of service at a time for maintenance if needed.

With the RPV depressurized, the SDC also provides water reject flow through an interface connection to the CFS and LWM.

Plant Mode 6 - Refueling Mode

At any time when the fuel pool gate is installed and water level is being held constant, the SDC is overboarding to the LWM to offset the CRD purge flow.

During refueling operation, the SDC system is in service. Each train can separately remove 100% of the decay heat 4 hours following reactor shutdown. The SDC system reject flow path is coordinated with the water level in the RPV, position of the FP gate, and availability of the Fuel Pool Closed Cooling and Cleanup System. When fully flooded and connected to the fuel pool, the Fuel Pool Cooling and Cleanup System can maintain a constant level. Refer to table 9A.2.3-3 for information pertaining to SDC configuration during refueling mode.

9A.2.3.6.4 Off-Normal Operation

Loss of Feedwater Flow Plant Modes 2-4

Operator initiates SDC or ICS.

Small Line Break Inside Containment Plant Modes 3-4

On identification of a small line break inside containment with or without preferred power during Plant Modes 3-4, the SDC is used for decay heat removal until the containment is isolated at which time the ICS is initiated to remove the decay heat.

9A.2.3.6.5 System Shutdown

The SDC is shutdown as the plant heats up in Plant Mode 2. The cooling function is no longer required since heat is being intentionally added to the RPV and the overboarding function transitions from the SDC to the CUW when the RPV pressure is sufficient to drive the necessary flow through the CUW to the overboarding destination. The thermal stratification reduction function won't be necessary once the CUW is overboarding since any cold spots developing near the piping terminations are overboarded or mixed with the flow entering the core.

Upon shutdown, the pump is turned off and the system isolation valves closed. The system remains water solid while shutdown.

9A.2.3.7 Instrumentation and Control

The SDC is a manually initiated system but is designed for automatic control of functions such as SDC system piping pre-warming, adjusting the system heat removal rate to stay within vessel and HX cooling water limitations, as well as allowing the operator to input a temperature rate setpoint that allows automatic vessel water temperature cooldown rate. Additionally, during reactor heat-up, the SDC controllers receive flow demand signals from the RLC to allow "swell" to be overboarded to maintain reactor level. The SDC can be controlled from the Main Control room as well as the Secondary Control Room.

The SDC has its own (triply redundant) sensors for control and its own actuators. It is possible to validate the temperature signals used for control with reactor water temperature measurements and (when steaming) reactor pressure measurements. The SDC controllers are triply redundant and are dual ported to the plant nuclear segment network. As with all Triple Modular Redundant controllers, extensive hardware and software diagnostics are provided for operator monitoring and alarms. Refer to Chapter 7 for information pertaining to TMR.

9A.2.3.7.1 Flow

Supply Flow

There is one SC1 flow element along with triply redundant SC1 transmitters as well as an independent set of triply redundant Safety Class 2 transmitters on the system supply line upstream of each SDC pump. The instruments are used in conjunction with the return flow and overboard flow instruments to support the DL3 and DL4a function of monitoring for system leakage. MSRV Isolation, Feedwater Isolation, and SDC Isolation are actuated on SDC line break indication in Plant Modes 1-4.

The Safety Class 2 transmitters are also used to monitor the system flow rate to help the operator determine that the pump is functioning properly and as input into the control system for automatic control of the pump.

Return Flow

There is one SC1 flow element along with triply redundant SC1 transmitters as well as an independent set of triply redundant Safety Class 2 transmitters on the system return line downstream of the overboarding branch connection. This flow indication is also used in conjunction with the supply flow and overboard flow instruments to support the DL3 and DL4a function of monitoring for system leakage respectively.

Overboard Flow

There is one SC1 flow element along with triply redundant SC1 transmitters as well as an independent set of triply redundant SC2 transmitters on the system overboard line upstream of the point where the two system trains connect. This flow indication is also used in conjunction with the supply flow and return flow instruments to support the DL3 and DL4a function of monitoring for system leakage respectively.

Pump Seal Purge Flow

There is one Non-Safety Class flow element along with one Non-Safety Class flow transmitter on the pump seal purge line. This flow instrument is used to monitor flow from the CRD system to the SDC pump to ensure seal purge flow is within acceptable limits.

9A.2.3.7.2 Temperature

Heat Exchanger Inlet Temperature

There are triply redundant SC1 as well as independent triply redundant SC2 temperature sensors upstream of the pump. This temperature is used to provide input for the SC1 and SC2 supply flow transmitters respectively.

The SC2 temperature sensors along with the heat exchanger discharge temperature is used to monitor the performance of the SDC heat exchanger. In Plant Modes 3 and 4, when the SDC is operational, these temperature elements are used to determine the average reactor coolant temperature.

Heat Exchanger Discharge Temperature

There are triply redundant SC1 as well as independent triply redundant SC2 temperature sensors on the discharge of the heat exchanger. This temperature is used to provide input for the overboard flow and return flow transmitters.

The SC2 temperature sensors along with the heat exchanger inlet temperature is used to monitor the performance of the SDC heat exchanger. This temperature is also used to modulate the heat exchanger bypass valve to allow the combined discharge temperature from the heat exchanger and bypass to meet a pre-set value.

Overboard Temperature

There is one SCN temperature sensor on the overboard line. This temperature is measured prior to discharge to LWM, and CUW to ensure that the temperature of the fluid is acceptable for the interfacing system.

9A.2.3.7.3 Pressure

SDC Pump Suction Pressure

There is one SC3 pressure instrument provided at the suction of each SDC pump. This pressure instrument is used to monitor the suction pressure of the SDC pumps, in order to monitor pump status and performance.

SDC Pump Discharge Pressure

There is one SCN pressure instrument provided at the discharge of each SDC pump. This pressure instrument is used to monitor the discharge pressure of the SDC pumps, in order to monitor pump status and performance.

Heat Exchanger Differential Pressure

There is one SC3 differential pressure instrument provided to measure the pressure drop across the tube side of the heat exchanger. This pressure drop is monitored to help inform the operator of needed maintenance.

9A.2.3.7.4 Radiation

Overboard Radiation Monitoring

There is one SCN radiation monitor installed on the SDC overboard line to monitor radiation levels. Refer to Chapter 11, Section 11.5 for information pertaining to the functions of the instrument.

9A.2.3.8 Monitoring, Inspection, Testing, and Maintenance

Maintenance and testing support equipment reliability. SSCs are designed to facilitate operation and maintenance. ISI and IST requirements are established and include inspection/test frequency for SSCs. Remote monitoring of key parameters assists in the trending of component performance as part of condition-based predictive maintenance. Testing is performed to ensure required functional operability is maintained under design conditions. The Safety-Category Functions of the SDC that support decay heat removal are tested in accordance with ASME BPVC Section XI as applicable. Testing is performed in accordance with plant procedures. Testing in support of plant pre-operational testing, startup, and commissioning is addressed in Chapter 14, Section 14.3.

Maintenance practices consider industry best practices and operating experience and conform with plant safety requirements to minimize potential for personnel injury. Maintenance activities implement ALARA practices to minimize work activity dose. Maintenance activities involving plant equipment may require involvement of vendors or industry specialists.

9A.2.3.9 Radiological Aspects

Chapter 12, Section 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions. In addition, the following design features are employed to reduce dose:

1. Process piping from the containment penetrations to the overboard lines should be routed through shielded areas or pipe chases to reduce the dose rates to the general areas of the reactor building.
2. Components within a compartment are arranged to facilitate rapid maintenance and inspection and allow for the addition of temporary shielding. Reducing the time for maintenance and inspection activities minimizes the duration and intensity of radiation exposure and assists with meeting ALARA goals.
3. Field located instruments and/or indicators are visibly and conveniently mounted outside of secondary shielding walls. This arrangement avoids unnecessarily exposing personnel to radiation during routine monitoring and maintenance.
4. SDC heat exchangers are manufactured with all-welded construction with flanged connections where accessibility to internal components is required for maintenance, inspection, or replacement. Maximizing welding reduces crevices in the heat exchanger and flanging the connections for accessibility helps reduce maintenance time – both of which support ALARA goals.

9A.2.3.10 Performance and Safety Evaluation

System reliability is enhanced through the use of periodic system testing, preventive maintenance, and application of redundant instrumentation.

The SDC Safety-Category functions during normal conditions include decay heat removal and, leak detection and isolation as discussed in Subsection 9A.2.3.1.2. There are no Safety-Category functions associated with Off-Normal conditions.

Containment isolation is provided in accordance with CNSC REGDOC-2.5.2 (Reference 9A.2.3-1) Chapter 8, Subsection 8.6.6. Refer to Chapter 6, Subsection 6.3.4 for a discussion related to BWRX-300 containment isolation and containment isolation valves.

The SDC is located in the Reactor Building which provides adequate protection against natural phenomena ensuring the ability of the SDC to perform its Safety-Category functions.

9A.2.3.11 References

- 9A.2.3-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

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Table 9A.2.3-1: Shutdown Cooling System Interfaces

Interfacing System	Interface Description	Interface Boundary
Process Radiation and Environmental Monitoring System	The Shutdown Cooling System provides a sample line to the PREMS for sampling of reactor coolant when SDC is in service. Radiation Monitor on the SDC overboard line provided by PREMS.	The systems connect downstream of the SDC isolation valve. Radiation Monitor on the SDC overboard line.
Isolation Condenser System	The Isolation Condenser System provides a suction flow path from the RPV via the condensate return lines to the Shutdown Cooling System for decay heat removal and overboarding.	The systems connect downstream of the ICS containment isolation valves.
Control Rod Drive System	The Control Rod Drive system provides water to the Shutdown Cooling System pumps for motor cooling and seal purge flow.	The system interface is on the SDC pump purge water line upstream of SDC isolation valves.
Reactor Water Cleanup System	The Shutdown Cooling System flow can be routed to the Reactor Water Cleanup System for overboarding or into the RPV through the CUW inlet lines for thermal stratification reduction.	The system interface is on the SDC overboard lines at the isolation valve between SDC and CUW.
Liquid Waste Management System	The Shutdown Cooling System can overboard to the Liquid Waste Management System reactor water storage tank if overboard flow is contaminated.	The systems interface on the SDC overboard lines downstream of the isolation valve between SDC and LWM.
Condensate and Feedwater Heating System	The Shutdown Cooling System discharges into the Condensate and Feedwater Heating System lines A and B which discharge inside the RPV to complete a closed cooling loop.	The system interface is on the upstream side of the isolation valve between CFS and SDC.
Main Condenser and Auxiliaries	The Shutdown Cooling System can overboard to the Main Condenser and Auxiliaries if available for overboard flow.	The system interface on the SDC overboard lines downstream of the isolation valve between SDC and MCA.
Plant Cooling Water System	The Plant Cooling Water System provides cooling water to the Shutdown Cooling System heat exchangers.	The system interface is located at the SDC heat exchanger nozzles.
Plant Pneumatics System	The Plant Pneumatics System provides air or nitrogen to operate the Shutdown Cooling System air operated valves.	The system interface is located at each valve actuator.

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Interfacing System	Interface Description	Interface Boundary
Safety Class 2 and 3 Electrical Distribution System	The Safety Class 2 and 3 Electrical Distribution System provides power to the Shutdown Cooling System instrumentation and ASD.	The system interface is at each powered component.
Reactor Building Structure	The Reactor Building Structure provides structural support to the Shutdown Cooling System SSCs.	The system interface is at each component support.
Safety Class 1 Instrumentation and Control System	The Shutdown Cooling System includes three divisions of SC1 leak detection instrumentation to provide inputs required by the Safety Class 1 Instrumentation and Control System Distributed Control and Information System.	Input/Output termination cabinets in the Digital Control and Information System cabinet rooms.
Safety Class 2 and 3 Instrumentation and Control System	The Safety Class 2 and 3 Instrumentation and Control System DCIS interfaces with Shutdown Cooling System instrumented and controlled components.	Input/Output termination cabinets in the Digital Control and Information System cabinet rooms.

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Table 9A.2.3-2: BWRX-300 Shutdown Cooling System Mode Table

Mode	Title	Reactor Mode Switch Position	Shutdown Cooling System Modes
1	POWER OPERATION (10% - 100% Rx Pwr)	RUN	N/A
2	STARTUP	REFUEL ⁽²⁾ or STARTUP	Modes A1 and/or B1, B2
3	HOT SHUTDOWN ⁽¹⁾	SHUTDOWN	Modes A1 and/or B1, B2
4	STABLE SHUTDOWN ⁽¹⁾	SHUTDOWN	Modes A1 and/or B1, B2
5	COLD SHUTDOWN ⁽¹⁾	SHUTDOWN	Modes A1 and/or B1, B2
6	REFUELING ⁽²⁾	SHUTDOWN or REFUEL	Modes A1 and/or B1, B2

(1) All reactor pressure vessel head closure bolts fully tensioned

(2) One or more reactor pressure vessel head closure bolts less than fully tensioned

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Table 9A.2.3-3: Shutdown Cooling Configuration During Refueling Mode

Reactor Water Level	FPC Assist	SDC System	FPC System
Reactor Cavity Pool Filled	Yes – Gate Removed	SDC normally in service (FPC can be evaluated to perform function as needed)	FPC cooling normally in service (SDC can be evaluated to perform function as needed)
Reactor Cavity Pool Filled	No – Gate Installed	SDC train in service (number of trains in service dependent upon the needed cooldown rate/decay heat)	One FPC cooling train normally in service, one in Standby
Reactor Cavity Pool Drained	No – Gate Installed	SDC train in service (number of trains in service dependent upon the needed cooldown rate/decay heat)	One FPC cooling train normally in service, one in Standby

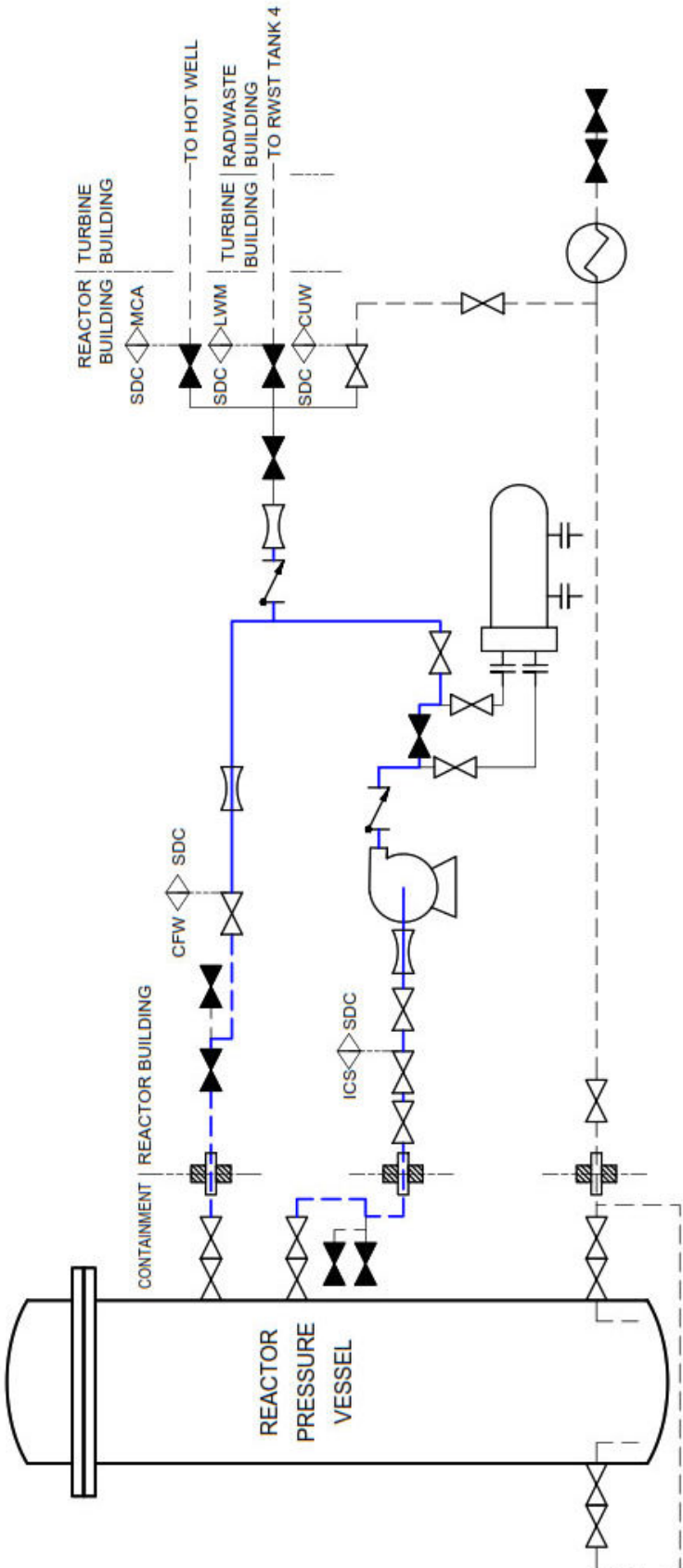


Figure 9A.2.3-1: Decay Heat Removal Subsystem

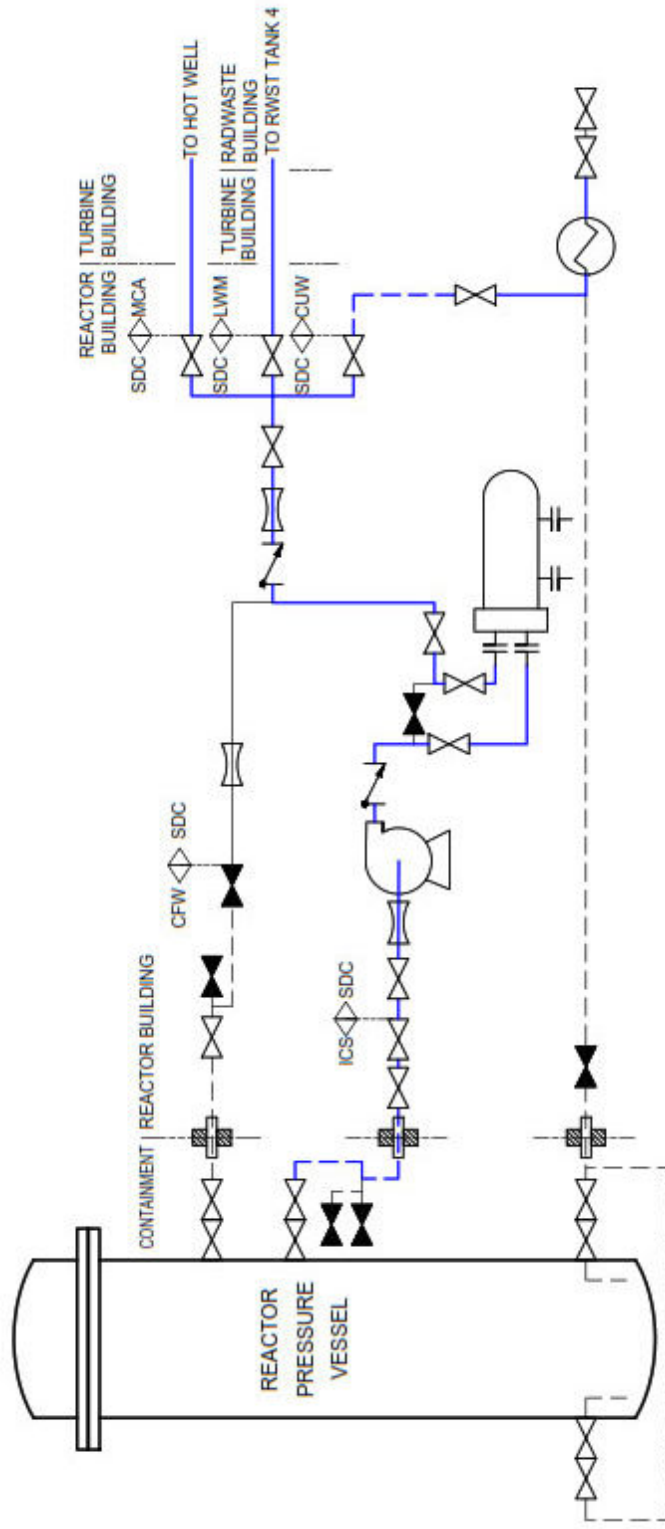


Figure 9A.2.3-2: Overboard Subsystem

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9A.2.4 Chilled Water Equipment System

The Chilled Water Equipment (CWE) provides chilled water cooling to the Heating Ventilation and Cooling Systems throughout the plant, to the offgas cooler condenser, to the charcoal adsorber vault Fan Coil Units (FCUs), and to the Containment Cooling System (CCS) in the Reactor Building.

The safety classification of the CWE as well as interfacing SSC is consistent with the requirements of REGDOC 2.5.2, Section 7.1 (Reference 9A.2.4-1).

9A.2.4.1 System and Equipment Functions

System and equipment functions associated with the CWE are specified as follows. Information pertaining to Defense Lines (DL) is discussed in Chapter 3, Subsection 3.1.6.

9A.2.4.1.1 Normal Functions (Non-Safety-Category)

CWE provides the following Non-Safety-Category functions:

1. Provides chilled water to cooling coils in HVAC equipment in the following buildings or areas: Control Building (CB), Reactor Building, Turbine Building (TB), and Radwaste Building (RWB)
2. Provides chilled water to selected equipment coolers

9A.2.4.1.2 Normal Functions (Safety-Category)

The CWE provides the following Safety-Category functions:

1. As part of a Safety-Category (TBD) function, the CWE supplies chilled water to the CCS Air Handling Units (AHUs)
2. The CWE supplies chilled water to the Safety Category (TBD) Instrumentation and Control System, Defense Line 2, Room A, and Safety-Category (TBD) Instrumentation and Control System, Defense Line 2, Room B, Safety Category (TBD) Fan Coil Units
3. The CWE supplies chilled water to the Defense Line 4a Room Fan Coil Units in the Control Building

9A.2.4.1.3 Off-Normal Functions (Non-Safety-Category)

1. During a LOOP, one chiller/pump set from each chiller train continues to operate and provide cooling to critical Fan Coil Units and AHUs throughout the plant using backup power provided by the standby diesel generators.
2. The rooftop chillers are designed and fabricated to withstand the effects of high winds, earthquakes, and other adverse weather conditions, with the exception of tornado and tornado missiles.
3. During a Toxic Gas Event, the CWE system provides chilled water associated the normal Control Building supply AHU, which continues to operate during the event.
4. In the event of a fire outside the Control Building, the outside air through the normal operating AHU will be secured and the CWE system provides chilled water to operating AHU, which continues to operate in recirculation mode only.

9A.2.4.1.4 Off-Normal Functions (Safety-Category)

1. As part of Defense Line 3, CWE containment isolation valves are designed to close upon receiving an isolation signal from the Safety Category (TBD) Instrument and Control System.

Containment isolation is provided in accordance with CNSC REGDOC-2.5.2 Section 8.6.6. Refer to Chapter 6, Subsection 6.3.4 for a discussion related to BWRX-300 containment isolation and containment isolation valves.

9A.2.4.2 Safety Design Bases

As part of DL3, CWE containment isolation valves on piping that penetrate the containment boundary are designed to close upon receiving an isolation signal from the Safety Category (TBD) Instrument and Control System.

9A.2.4.3 Description

CWE is a closed loop chilled water system that supplies chilled water to various AHU cooling coils and plant equipment coolers in the Turbine Building, Radwaste Building, Reactor Building, and Control Building. Heat absorbed by the CWE is rejected from the CWE chiller condensers mounted on the Radwaste Building roof to atmosphere.

CWE is comprised of four (4) CWE air-cooled chillers, four (4) CWE pumps, two (2) expansion tanks (one per train), four (4) air separators, one (1) chemical bypass feeder, one (1) glycol auto-fill unit, piping, valves, instruments, and controls.

The CWE is split into two trains which allows for the isolation of the two different sets of AHUs and FCUs supporting redundant defence-in-depth equipment. Each CWE train consists of two chillers, two pumps, two air separators, and an expansion tank, while the glycol auto-fill unit and chemical bypass feeder are shared among both trains.

Both CWE trains are cross-tied together with air operated valves, which are normally open, to allow for the three active chillers to evenly share the heat load; however, these valves can be closed for train separation.

Figure 9A.2.4-1 depicts a simplified flow diagram of the CWE System.

Four 33 1/3% capacity air-cooled chillers are provided to reject heat from the closed chilled water loop to the environment. During normal operation, three chillers are in service while the fourth is in standby mode. Each chiller consists of an evaporator section, condenser section, refrigerant compressor, controls, and an integrated waterside economizer which allows for the chillers to run at reduced electrical loads at lower ambient temperatures. The CWE chillers utilize R-134a to exchange heat with the propylene glycol/water mixture. Since these units are mounted outside, the use of R-134a does not pose a risk to plant personnel. Each CWE chiller is provided with built-in protection against freezing.

The four 33 1/3% capacity in-line chiller pumps deliver chilled water from the CWE return header to the dedicated chiller. During normal operation, three pumps are in service while the fourth is in standby mode. Each pump discharge line is provided with a check valve to prevent backflow through the pump.

The CWE system trains are powered by separate generator load groups as part of the defense in depth protection function.

Four air separators are provided to remove entrained air. An air separator is connected to the chilled water return header upstream of each chiller pump. The air that is removed by the air

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separators is piped to the train specific expansion tank to provide the gas cushion for chilled water expansion and contraction.

The two expansion tanks are connected to the two main chilled water return lines and provide a reservoir of demineralized makeup water to account for small amounts of system leakage and accommodate thermal expansion and contraction of water within the system while maintaining the system pressure. The expansion tank level is also used to determine the need for additional makeup fluid. Each expansion tank is shrouded to prevent rain, snow, and other debris from collecting in the expansion tank curbed area. The shroud is fitted with sight glasses for operators to manually inspect the curbed area for any liquid that has been collected.

The glycol auto-fill unit has a connection to the Water, Gas, Chemical Pads System (Subsection 9A.9.5) to provide demineralized water to maintain a predetermined propylene glycol/water mixture.

Isolation valves are provided at the interfaces with the components being cooled by CWE to allow for maintenance on those items without impact to the CWE system operation.

The supply and return line penetrations have isolation valves with electrical division 1 and division II connections outside and inside containment respectively, and ASME Class 2 piping into containment. The return piping inside of primary containment is provided with relief valves to protect against overpressure caused by thermal expansion. The system is designed to meet mechanical and electrical separation and redundancy requirements to ensure that the consequences of a single failure, pipe whip, jet forces, missiles, and the effect of failure of any non-Seismic Category I components do not compromise the ability of the system to meet its safety category requirements.

The CWE containment isolation valves are designed to operate in environments associated with the normal and accident conditions in the RB to which they are exposed. Refer to Chapter 3, Subsection 3.9.4 for information pertaining to Environmental Qualification.

The CWE expansion tanks, glycol auto-fill unit, chemical bypass feeder, and chillers pumps are all surrounded by a permanent curb to prevent the accidental excursion of propylene glycol into the Equipment and Floor Drainage System (Subsection 9A.9.3) sumps. To minimize dose exposure to plant personnel, the isolation valves for the FCUs located inside the bioshield area of the Turbine Building are positioned outside the bioshield area.

9A.2.4.3.1 Component Description

Air-Cooled Chillers

Four (4) 33% capacity air-cooled chillers are provided. Each chiller consists of an evaporator section, condenser section, refrigerant compressor, and controls. The chillers have individual temperature controls.

The CWE chillers are tested in accordance with Air Conditioning and Refrigeration Institute (AHRI) Standard 550 (Reference 9A.2.4-2) and American Society of Heating, Refrigerating, and Air Conditioning Engineers (ASHRAE) Standard 30 (9A.2.4-3).

Refrigerant piping is designed and fabricated per CSA B52 (Reference 9A.2.4-5).

Each chiller is equipped with pressure relief valves which relieves excess pressure of the refrigerant charge to the atmosphere in accordance with ASHRAE 15 Safety Code and ASME BPVC, Section VIII.

The chiller capacities include a 15% margin to accommodate unaccounted hot surfaces, higher than anticipated latent loads, and other unknown cooling loads

Pumps

Four (4)x33% capacity centrifugal, electric motor driven pumps are provided with variable frequency drivers.

Pumps are factory tested per ANSI/HI 14.6.

The pumps are sized such that two (2) pumps are capable of supplying the necessary flow during a Loss-of-Offsite Power.

Chemical Bypass Feeder

A carbon steel chemical bypass feeder tank is provided to treat the glycol water mixture and prevent the development of organics within the closed cooling water loop.

The bypass feeder tank is designed to meet the requirements of ASME BPVC, Section VIII.

Expansion Tanks

The two atmospheric vessels are designed to meet the requirements of API 650.

Glycol Auto Fill Unit

The packaged, glycol auto fill unit maintains the glycol system pressure by providing glycol make-up automatically upon a drop in system pressure. The glycol auto-fill unit has a connection to the Water, Gas and Chemical Pads system to provide demineralized water to maintain a predetermined propylene glycol/water mixture. Upon a drop in liquid level in the expansion tanks, the integrated side suction peripheral pump starts adding fluid from the glycol auto fill tank to the CWE System expansion tanks until the setpoint level is reached.

Piping and Valves

All non-refrigerant piping is designed in accordance with the requirements of ASME B31.1 with the exception of the containment penetration portion. The containment isolation valves and lines that supply and return chilled water to the containment cooling system cooling coils are Safety Class 1, designed to the requirements of BPVC Section III, Division 1- Subsection NCD; and are Seismic Category I. The containment isolation valves are designed to fail-closed on a loss of air or signal.

Refrigerant piping inside the air-cooled chillers is designed and fabricated in accordance with the requirements of ASHRAE 15(Reference 9A.2.4-2).

Overpressure relief valves are provided on the CWE return lines located both inside and outside of containment.

CWE piping is provided with insulation to prevent pipe sweating as required. The pipe insulation meets the combustibility requirements in compliance with fire protection codes and standards.

9A.2.4.4 Materials

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials through material chemistry, heat treatment, contamination, and material processes controls.

Materials are selected in accordance with applicable codes, standards, and industry practice for the design, service, and test conditions and expected ambient conditions. Materials are compatible with the internal process and external environmental conditions during normal, abnormal, accident, and beyond design basis accident conditions as appropriate. Building

construction utilizes noncombustible materials as defined in the fire protection Codes and Standards.

9A.2.4.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.2.4-1 for Chilled Water Equipment interfaces with other equipment or systems.

9A.2.4.6 System and Equipment Operation

9A.2.4.6.1 Normal Operational Concept

Up to three of the four chiller pump sets are placed in normal operation based on the cooling load demand. One chiller pump set is always available in standby. The CWE is designed to operate with N+1 redundancy. Where N is the number of pieces of equipment to meet the baseload demand. N+1 indicates there is one extra piece of equipment provided. Hence during normal operation 3 chillers/pumps will be in service while the 4th be in standby mode.

The chilled water control valves modulate in response to room or supply air temperature controllers for the Control Building and Reactor Building, which are part of the HVS except for the Containment Cooling System (Subsection 9A.5.6) AHUs. The temperature control valve downstream of the CCS AHU cooling coils modulate in response to the air discharge temperature of the CCS AHUs. As the cooling loads decrease, the valves modulate to decrease the chilled water flow causing a differential pressure increase across the supply and return mains.

Chillers are automatically started or stopped, based on the CWE supply header temperature.

The glycol auto-fill unit will automatically be switched on and off based on the level transmitter in the expansion tanks. Upon receiving a low level signal, the glycol auto-fill unit will switch on and add additional fluid to the CWE expansion tanks.

Normally, chiller train A, which consists of chillers and pumps 1 and 2, and expansion tank A; and train B, which consists of chillers and pumps 3 and 4 and expansion tank B are cross-tied together such that flow from the RWB, TB, TB switchgear rooms A and B, RB loops A and B, and CB loops A and B, will be shared proportionally between the two trains based on the number of active chillers in each train.

Depending on the ambient conditions, the chillers are able to operate in one of three different modes: mechanical cooling only, hybrid cooling, or free cooling only. When the ambient temperature is too warm to provide free cooling, an integrated three-way valve at the chiller inlet allows the glycol/water mixture to only run through the condenser coils. When the temperature reaches an ambient temperature where some free cooling is feasible, the three-way valve diverts some flow through the free cooling coils which then enters the chiller evaporator. If further fluid temperature reduction is required, the chillers will perform the remaining mechanical cooling. As the ambient temperature continues to decrease, there will be a point where the supply temperature can be met by free cooling operation alone and the mechanical cooling can be shut off completely. During free cooling mode, the chillers operate using significantly less power because the waterside economizer does not use a refrigerant loop and the section requires fewer hot air discharge fans. When one or multiple FCUs located inside the Turbine Building bioshield area are manually isolated because of a leak, the chilled water trapped between the isolation valves is drained to prevent damage to the CWE distribution pipe or FCUs tubes due to thermal expansion.

9A.2.4.6.2 Off-Normal Operational Concept

During a LOOP and turbine trip, one CWE chiller and pump from each chiller train will trip and restart upon receiving power from the standby diesel generators. The inlet guide vane of the non-

operational chillers, and the block valves in the TB, excluding the TB switchgear rooms, and RWB return headers close automatically, so that the CWE flow will only flow through the equipment required for the continued safe operation of the plant.

Upon station blackout, the CWE is not operating.

9A.2.4.7 Instrumentation and Control

9A.2.4.7.1 Instrumentation

CWE contains sufficient instrumentation to:

1. Verify chilled water pump performance, by monitoring pump suction and discharge pressure.
2. Verify chiller performance, by monitoring the inlet and outlet pressure, CWE flowrate, and the inlet and outlet temperatures for each chiller.
3. Verify chilled water return and supply header temperatures.
4. Verify pressure loss through the system.
5. Provide remote status indication of the air-operated containment isolation valve to the MCR and SCR.

9A.2.4.7.2 Controls

System and component operating status, including the state of manual overrides, and the state of the A and B trains are provided at the Main Control Room (MCR) and Secondary Control Room (SCR). Manual initiation and shutdown of CWE is provided from the MCR.

The following CWE primary displays and alarms are provided:

1. MCR and SCR Major Indications:
 - a. Chiller operating status
 - b. Chiller inlet and outlet pressures and temperatures
 - c. CWE system operating pressures and temperatures
 - d. CWE pump suction and discharge pressures
 - e. CWE pump flow rates
 - f. Air-operated containment isolation valve status
 - g. Expansion tank liquid level
 - h. CB, RB, and TB switchgear room loops misaligned (only possible when the chiller headers are isolated)
2. MCR and SCR Alarms:
 - a. Loss of CWE flow to any operating chiller
 - b. Chiller trip and chiller trouble
 - c. High CWE return water temperature
 - d. High or low chiller outlet temperature
 - e. High or low expansion tank liquid level

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The CWE is expected to run continuously with three of the four chillers and pumps running while the other chiller and pump are on standby. The operator manually brings the chiller/pump combo online by selecting an operating chiller to put into standby. Upon receiving the standby command, the standby pump will start, followed by the chiller. Once the MCR or SCR receives confirmation that the pump and chiller are both on, the operating chiller will stop, followed by the pump. The chillers and chiller support systems are capable of automatically starting and stopping based on the CWE supply header temperature; however, the chillers and CWE pumps can be operated remotely from the MCR.

A chilled water temperature control valve located in the return piping at each AHU cooling coil bank, is designed to maintain air discharge temperature of the upstream AHU. The chilled water control valves modulate in response to room or supply air temperature controllers, which are part of the HVS except for the CCS AHUs. The temperature control valves downstream of the CCS AHU cooling coils modulate in response to the air discharge temperature of the CCS AHUs. As the cooling loads decrease, the valves modulate to decrease the chilled water flow causing a differential pressure increase across the supply and return mains.

The automatic block valve in each building's chilled water return line is modulated, during plant startup, to balance pressure loss in each building chilled water loop so that flow is adequately distributed to every building.

The temperature control valves in each building's chilled water return line can be modulated to balance pressure loss in each building's chilled water loop so that flow is adequately distributed to every building.

During normal operation, the chillers operate to automatically maintain the chilled water supply temperature at a predetermined value.

The standby chiller pump will auto start upon trip of any operating pump. The CWE pump is tripped on occurrence of any of the following:

- Low CWE pump suction pressure
- A manual stop command
- A protective device trip

The chiller is tripped on occurrence of any of the following:

- Loss of CWE flow
- A failure of the chiller automatic start sequence
- A manual stop command
- High system pressure

The containment isolation valves are automatically closed upon receiving a containment isolation signal and also automatically fail-closed on loss of air or signal.

9A.2.4.8 Monitoring, Inspection, Testing, and Maintenance

Pre-operational testing is performed in accordance with applicable codes and manufacturer recommendations.

The CWE chillers are tested in accordance with AHRI Standard 550 (Reference 9A.2.4-3) and ASHRAE Standard 30 (Reference 9A.2.4-4).

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Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package and are a part of a maintenance program conforming to ASME Code OM 2020.

Areas requiring inspection are provided with access and removable insulation.

CWE piping and valves for the containment penetrations are tested in accordance with 10CFR50, Appendix J and CNSC REGDOC-2.5.2. Test and vent connections are provided at the containment isolation valves in order to verify that the valves meet the local leak rate limits.

The containment isolation valve closure time is monitored during the valve operability test and the leakage is monitored or verified during the valve leakage test as specified in the containment leakage testing program. Leak detection and inspection for primary containment isolation features is designed to ASME BPVC Section III, Division 1, Class 1. A test connection is provided to support local leak rate testing of the primary containment boundary. The test connections minimize the amount of water that must be drained to permit periodic testing of containment isolation valves that include operability testing, leak rate testing, valve status verification, and test frequency.

Containment isolation valve testing, stroke time testing, and leakage rate testing are incorporated to ensure proper and safe functionality of the valves.

Isolation valves are provided at each piece of equipment, control valve, and piping circuit so they can be isolated for maintenance and repair.

Chillers and pumps are arranged to provide adequate floor space and unobstructed clearance to permit monitoring, maintenance, and inspection of each unit.

Isolation valves are provided at the interfaces with the components being cooled by CWE to allow for maintenance on those items without impact to the CWE system operation.

9A.2.4.9 Radiological Aspects

Chapter 12 provides information pertaining to design measures that are taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are As Low As Reasonably Achievable (ALARA).

9A.2.4.10 Performance and Safety Evaluation

The CWE is split into two trains which allows for the isolation of the two different sets of AHUs and FCUs supporting redundant defense in depth equipment.

CWE is designed with sufficient redundancy to assure that chilled water is normally available during all modes of plant operation, including startup and shutdown. Certain equipment needed during a Loss-of-offsite power are provided Diesel Generator backup power.

As part of DL3, the CWE performs a containment isolation Safety-Category function. The CIVs are designed to close upon receipt of an isolation signal from the Safety Category (TBD) Instrument and Control System. During normal operations the CWE supplies chilled water to the CCS AHUs, DL2 Room A and Room B FCU's, and chilled water to the DL 4a Room FCU in the Control Building.

9A.2.4.11 References

- 9A.2.4-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.2.4-2 ASHRAE Standard 15, "Safety Standard for Refrigeration Systems."

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- 9A.2.4-3 AHRI Standard 550/590 (I-P), "Performance Rating of Water-chilling and Heat Pump Water-heating packages using the Vapor Compression Cycle," Air Conditioning, Heating, & Refrigeration Institute.
- 9A.2.4-4 ASHRAE Standard 30, "Method of Testing Liquid Chillers."
- 9A.2.4-5 CSA B52, "Mechanical Refrigeration Code," CSA Group.

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Table 9A.2.4-1: Chilled Water Equipment System Interfaces

Interfacing System	Interface Description	Interface Boundary
Safety Category (TBD) Instrument and Control System	Provides all Safety Category (TBD) I&C control for containment isolation valve logic	At Input/Output termination
Safety Category (TBD) Instrumentation and Control System	Provides Safety Category (TBD) I&C controls including pump logics, valve logics, excluding the CIV, instrumentation (pressure, flow, temperature, level) etc.	At Input/Output termination
Non-Safety Instrumentation and Control System	Provides Non-Safety-Category controls	At Input/Output termination
Offgas System	CWE provides chilled water for the offgas cooler condenser and charcoal adsorber vault FCU cooling coils	Offgas cooler condenser flanges and isolation valves for the charcoal adsorber vault FCU
Plant Pneumatics System	PPS provides instrument air/nitrogen to pneumatically operated valves	At CWE air operated valves
Safety Category (TBD)Electrical Distribution System	Provides normal and standby diesel backed power to the CWE chillers and pumps	At CWE equipment
Non-Safety Electrical Distribution System	Provides Non-Safety-Category power to CWE equipment	At CWE equipment
Containment Cooling System	CWE provides chilled water for containment cooling system chilled water coil loads	CCS AHU flanges
Heating, Ventilation, and Cooling System	CWE provides chilled water for the HVS chilled water cooling coil loads	Isolation valves at AHU and FCU
Equipment and Floor Drain System	The condensate collected from the pump baseplate drain is piped to the EFS sumps. The liquid connection from the pressure relief valves inside containment discharge to the containment sump, while the pressure relief valve in the RB, outside of containment, is piped to the floor drain	Drain pipe from chiller pumps and liquid connection from RB PRVs
Water, Gas, Chemicals Pads	Water, Gas, Chemicals System supplies makeup water to the glycol auto-fill unit	Isolation valve at glycol auto-fill unit

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Interfacing System	Interface Description	Interface Boundary
Process Radiation and Environmental Monitoring System	PREMS provides process sampling for chemistry monitoring	Grab sample station(s)

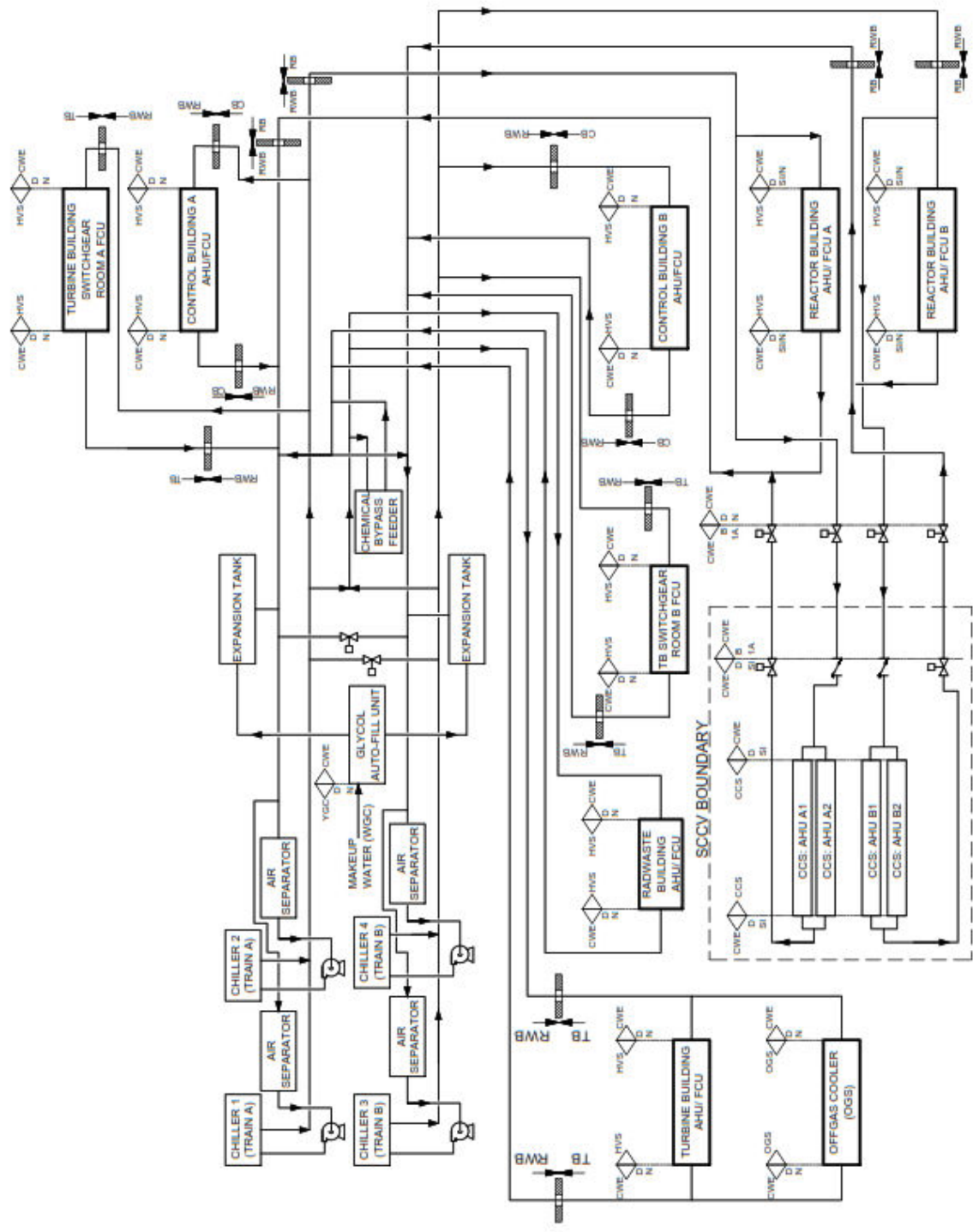


Figure 9A.2.4-1: Chilled Water Equipment

9A.2.5 Normal Heat Sink System

The Normal Heat Sink (NHS) provides heat rejection for the CWS and the PCW. Refer to Chapter 10, Sections 10.8, and Subsection 9A.2.1 for information related to the CWS and PCW respectively.

Refer to Subsection 9B.3.5 for information pertaining to the Pumphouse, Forebay, and Tunnels.

9A.2.5.1 System and Equipment Functions

Normal Functions (Non-Safety-Category)

The NHS provides the supply of cooling water for the circulating water pumps, which removes heat from the condenser and PCW heat exchangers and accepts the circulating water return flow. NHS performs this function at all times for all plant modes of operation.

Normal Functions (Safety-Category)

The NHS performs the following Safety-Category functions during normal conditions:

- Provides cooling water as a means to reject heat from the MCA to the environment
- Provides cooling water as a means to reject heat from the PCW heat exchangers to the environment

Off-Normal Functions (Non-Safety Category)

The system does not perform any Non-Safety-Category functions during off-normal conditions.

Off-Normal Functions (Safety-Category)

Upon a LOOP, NHS continues to provide a cooling water source and heat rejection functions to support CWS interface with PCW.

9A.2.5.2 Safety Design Bases

The NHS provides cooling water to the two CWS (Chapter 10, Subsection 10.8) plant cooling water supply pumps (2A and 2B), which in turn are relied upon to provide cooling water to the PCW heat exchangers, to support Safety-Category functions for FPC (Subsection 9A.1.3) and SDC (Subsection 9A.2.3).

9A.2.5.3 Description

The NHS is a once-through cooling water system design. Water will flow into the intake structure forebay from Lake Ontario by means of an intake tunnel. The water is strained prior to pumping by the CWS pumps. Water flows through the CWS system where it absorbs the heat from the main condenser and the PCW heat exchangers. The water is discharged back into Lake Ontario through the discharge structure. A recirculation line from the CWS discharge line to the intake is provided to moderate NHS temperature at the Forebay of the NHS, as needed, during cold weather conditions.

9A.2.5.4 Materials

Material and process control requirements for the BWRX-300 structures and components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from corrosion (as applicable) through material chemistry, heat treatment, contamination, and material processes controls.

9A.2.5.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.2.5-1 for system interfaces.

9A.2.5.6 System and Equipment Operation

The NHS system operates under all normal operating modes of the plant. The Off-normal operational modes include operations following a LOOP event.

9A.2.5.6.1 Normal Operational Concept

The NHS has its water pumped by the circulating water pumps to the circulating water supply in the Turbine Building where it absorbs the heat from the main condenser and the PCW heat exchangers. Next, the CWS dedicated PCW and Condenser return lines discharge the circulating water back to the heat sink.

9A.2.5.6.2 Off-Normal Operational Concept

Upon a LOOP, NHS continues to provide a cooling water source and heat rejection functions to support CWS interface with PCW.

9A.2.5.7 Instrumentation and Control

The NHS has no instrumentation and controls associated with the system. All NHS parameters are measured within CWS.

9A.2.5.8 Monitoring, Inspection, Testing, and Maintenance

Maintenance and inspection is based upon integrating proactive, reactive, preventive, and predictive maintenance and operating experience. Implementation of maintenance and inspection increases the probability that the NHS structures, systems, and components function in the required manner over their design life cycle. Operational testing is performed in accordance with plant procedures.

9A.2.5.9 Radiological Aspects

Chapter 12, Subsection 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.2.5.10 Performance and Safety Evaluation

The NHS is capable of performing its design functions during all modes of operation.

Upon a LOOP, NHS continues to provide a cooling water source and heat rejection functions to support CWS interface with PCW.

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Table 9A.2.5-1: Normal Heat Sink Interfaces

Interfacing System	Interface Description	Interface Boundary
Circulating Water System	The NHS provides the source of water used as the cooling medium to the CWS pumps and accepts the return flow from the CWS pumps. When required for cold weather operation, CWS is designed to maintain a minimum intake basin temperature through a recirculation line from the CWS discharge line	At the intake and discharge structures

9A.2.6 Isolation Condenser System Pool Cooling and Cleanup System

The primary function of the Isolation Condenser System Pool Cooling and Cleanup System (ICC) is to remove heat from the Isolation Condenser System (ICS) (Chapter 6, Subsection 6.2) pools such that the bulk temperature of water in the pools is maintained below Technical Specifications limits and thereby ensure the readiness of the ICS to perform its Safety-Category function. Secondary functions of the ICC include maintaining the cleanliness of the ICS pool water and providing the capability to add clean makeup water during normal reactor operations to offset the minor routine loss of water inventory due to evaporation.

9A.2.6.1 System and Equipment Functions

The ICC performs the following functions during normal and off-normal conditions.

9A.2.6.1.1 Normal Functions (Non-Safety-Category)

1. The ICC extracts heat from the ICS Pool Water to maintain IC (Isolation Condenser) pool temperature within TS limits for maximum pool temperature, and thereby maintains ICS pool readiness for IC deployment.
2. The ICC removes soluble and insoluble impurities from the ICS Pool Water to comply with plant water quality requirements.
3. The ICC provides the capability of adding makeup demineralized water to the ICS pools during normal reactor operations to replace water inventory lost by evaporation from the ICS pool surface.

9A.2.6.1.2 Normal Functions (Safety-Category)

The Safety-Category functions provided by the ICC are associated with the suction and return pipe penetrations into the ICS pools. The Suction Surge Tank is one of two primary interfaces with the Reactor Building Structure (RBS) ICS Pool Structure and is categorized as SC1 and as a Seismic A component. The Return Guard Pipe is the second of two primary interfaces with the RBS ICS Pool Structure and is likewise categorized as an SC1 and Seismic A component. The principal function of these two components is to prevent draining of the ICS Outer Pools in the event of a break in ICC piping below the ICS pools.

9A.2.6.1.3 Off-Normal Functions (Non-Safety-Category)

The ICC does not provide any Non-Safety-Category functions during off-normal conditions.

9A.2.6.1.4 Off-Normal Functions (Safety-Category)

The ICC does not provide any Safety-Category functions during off-normal conditions.

The design of the ICC meets requirements specified in CNSC REGDOC-2.5.2 Section 7.1 (Reference 9A.2.6-1) as related to ensuring that the portions of the ICC, namely the suction and return pipe penetrations into the ICS pools, that interact with the SC1 ICS are also designed to SC1 requirements. Considering that the ICC is of a lower safety classification than the ICS, designing the suction and return pipe penetrations to SC1 requirements ensures that the failure of a lower safety class SSC cannot propagate to an SSC belonging to a higher safety class.

9A.2.6.2 Safety Design Description

The Safety-Category functions provided by the ICC are associated with the suction and return pipe penetrations into the ICS pools.

The Suction Surge Tank and the Return Guard Pipe are designed to prevent draining of the ICS Outer Pools in the event of a break in ICC piping below the ICS pools. The probability of such

breaks occurring is minimized by locating ICS pool isolation valves as close to the ICS pool penetrations as practicable. In the event of a non-isolable break in the ICC suction piping, it is possible for the ICS Cubicles (i.e., Inner Pools) to drain to the bottom of the connecting weir in the partition wall separating the ICS Cubicles from the ICS Outer Pools, and for the ICS Outer Pools to drain to the top of the Suction Surge Tank. In the event of a non-isolable break in the ICC return piping, the design of the Return Guard Pipe prevents draining the ICS Outer Pools and Cubicles because the top of the Return Guard Pipe is above the maximum pool water level.

ICC suction and return connection interfaces with the Reactor Building Structure are designed such that any failure in the ICC pressure boundary cannot result in draining the ICS Pools.

9A.2.6.3 Description

The ICC is classified as Non-Safety Class, Non-Seismic Category and Quality Group D. The ICC has no Defense Line Functions. Figure 9A.2.6-1 depicts the ICC.

The ICC consists of two independent and identical 50% capacity trains which service the three Isolation Condenser Cubicle Pools. The two identical thermal processing trains can be operated together or operated with one train and the other train shutdown as needed to maintain conditions in the ICS pools.

Both ICC trains take suction from a single penetration in the Isolation Condenser Outer Pool A. An air-operated ICS Pool Isolation Valve is installed in the main suction pipe to stop flow from Outer Pool A. Processed ICS pool water is returned directly to the three IC cubicles. Return water can be directed to individual IC cubicles by controlling the flow of ICC effluent by aligning IC cubicle isolation valves as desired. The ICS Pool suction and return isolation valves are equipped with manual overrides and handwheels, which allows the valves to be manually operated if the valve actuator fails, thereby increasing system reliability.

Each ICC pump/heat exchanger train can be isolated as needed to maintain ICS readiness to allow both trains to operate simultaneously, or so that one train can operate separately to allow maintenance to be performed on the other train for continuous operation. Each ICC train includes a centrifugal pump with Adjustable Speed Drive (ASD), frame-and-plate HX, associated piping, sensors, and valves. PCW (Subsection 9A.2.1) flows through the HX to extract heat from the ICS pool water which is rejected to the environment. The discharge from the two ICC pump/HX trains is processed by a skid-mounted demineralizer before being returned to the IC Cubicle Pools. All ICC flow passes through the demineralizer except when bypassed. The demineralizer can be bypassed when the temperature of the discharge exceeds the maximum temperature allowed for the demineralizer resins or when chemicals are injected into the water to inhibit corrosion and biological growth in the IC pools and ICC components. The ICC is equipped with a dosing pot for injecting chemicals for corrosion and biological control into the process fluid.

Because the ICC is of a lower safety class and quality group than the ICS and pools, design provisions are provided to ensure that a pressure boundary failure in its piping or components cannot adversely impact the higher safety class system, specifically draining of the ICS pools. This is accomplished by a surge volume (Suction Surge Tank) that is a component of the seismically designed reactor building IC pools structure, from which the ICC takes suction. In the event of a failure in the ICC that creates a drainage path, only the surge volume can be drained with no impact on the safety class IC pools volume, and with minimal impact on the ICC components from flooding. Similarly, the discharge or return piping to the IC pools is routed through the seismic Return Guard Pipe, and then routed near the ceiling of the ICS Outer Pools and across the ICS cubicle wall through the connecting weir at which point the return pipes are submerged and routed to a distribution sparger that discharges the cooled and purified water

directly to the ICS Cubicle near the bottom of the IC. The high point of the return piping contains an anti-siphon feature to prevent backflow in the event of an ICC pressure boundary failure. Additionally, ICS pool isolation valves are located as close as possible to the pipe penetrations to minimize the impact to the ICS pools for any failure of ICC piping beyond the isolation valves.

9A.2.6.3.1 Component Description

The following information is provided relative to the major components in the ICC system.

Suction Surge Tank

The Suction Surge Tank is one of the primary interfaces with the ICS Pools and, therefore, is an ICC component with SC1 and Seismic A classifications. The Suction Surge Tank consists of flanged (removable) pipe spools. All Suction Surge Tank components are made of Type 304/304L or 316/316L stainless steel. The main suction pipe penetration into the ICS Outer Pool A inside the Suction Surge Tank is designed to provide a leakage-tight barrier between the ICS Outer Pool A and the underlying ICC Equipment Area. A cover plate with vent tube for the Suction Surge Tank is provided to permit the Suction Surge Tank to be drained for downstream maintenance activities without draining the ICS Outer Pools.

Return Guard Pipe

The Return Guard Pipe is the second of two primary interfaces with the ICS Pools and, therefore, is an ICC component with SC1 and Seismic A classifications. The Return Guard Pipe consists of flanged (removable) pipe spools. Return Guard Pipe components are made of Type 304/304L or 316/316L stainless steel. The return pipe penetration into the ICS Outer Pool A, inside the Return Guard Pipe is designed to provide a leakage-tight barrier between the ICS Outer Pool A, and the ICC Equipment Area. The Return Guard Pipe includes an attached cover that seals the Return Guard Pipe to prevent water accumulation and debris from entering the Return Guard Pipe.

The Return Guard Pipe cover plate assembly is designed to provide lateral support for the anti-siphon devices in the return piping to meet seismic structural requirements.

Filtration

A cylindrical metallic screen is installed at the main suction inlet for foreign material exclusion. The foreign material exclusion screen is connected to the main suction pipe by a flanged connection. The foreign material exclusion screen is designed to facilitate periodic cleaning to remove accumulated debris on the outer surface of the screen. In-line filters with replaceable filter cartridges are installed at the inlets to the heat exchangers on both the hot side and the cold side to remove minute suspended particulates to prevent clogging the narrow flow channels between plates in the heat exchangers for maximum heat transfer effectiveness. All filtration components and surfaces in contact with the process fluid is made of Type 304/304L or 316/316L stainless steel.

Adjustable Speed Drive Pump Motors

The ICC pumps are driven by ASDs with electric motors to provide operational flexibility for: a) optimal performance in restoring ICS Pool Water temperature to TS requirements following deployment of one or more ICs; and b) efficient and economical performance in maintaining ICS Pool Water temperature and cleanliness during normal reactor operations.

Pumps

Pump suction and discharge connections are flanged. All pump components and surfaces in contact with the process fluid are made of Type 304/304L or 316/316L stainless steel. The ICC pumps are designed to comply with the requirements of ANSI/HI 1.3 (Reference 9A.2.6-2). Each

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pump is powered by an Adjustable Speed Drive with electric motor. The pumps have mechanical seals to minimize the potential for leakage and reduce the need to perform maintenance on shaft seal packings. Pumps are installed within catch basins to contain minor leakage. Moisture sensors are installed in the catch basin to inform reactor operators of potential maintenance issue due to water leakage.

Heat Exchangers

Frame-and-plate HXs are used to reject heat from the process fluid in the ICC loop to the PCW. HX components and surfaces in contact with the process fluid are made of Type 304/304L or 316/316L stainless steel. HX inlet and discharge connections are flanged to facilitate removal as necessary for maintenance, repair, and replacement. Drainage taps are provided in HX or piping connections to allow the HX to be drained to facilitate maintenance or repair activities.

Demineralizer

An integrated skid-mounted demineralizer is used to clean the process fluid in the ICC loop. Demineralizer components and surfaces in contact with the process fluid are made of Type 304/304L or 316/316L stainless steel. A drainage tap is provided to allow the Demineralizer tank to be drained to facilitate maintenance or repair activities. The Demineralizer is sized to process the effluent from a single ICC pump train. ICC Demineralizer spent resin media is discharged to the Solid Waste Management System (Chapter 11, Section 11.4).

Dosing Pot

A dosing pot allows chemical injection into the process fluid to inhibit corrosion and biological growth in ICC piping and components and the ICS pools. All dosing pot components and surfaces in contact with the process fluid are made of Type 304/304L or 316/316L stainless steel. A drainage tap is provided to allow the dosing pot to be drained when not in use.

Backflow Preventer

Back flow preventers are installed in the return piping to the IC cubicles to prevent inadvertently draining the IC cubicle pools in the event of an ICC piping failure below the ICS pools. The back flow preventers are installed at the highest elevation of the ICC piping to function as anti-siphon devices. Back flow preventer components are made of Type 304/304L or 316/316L stainless steel. The Back Flow Preventer vent hole is covered by a shield plate located normal to the axis of the vent hole to limit the flow of return processed fluid out of the Back Flow Preventer.

Distribution Spargers

The return water distribution spargers are installed in the IC cubicles near the ICs. The spargers are installed at or near the bottom of the Isolation Condenser cubicles in a manner that promotes bulk flow of the water in the cubicle, particularly with respect to the Isolation Condenser HX. The distribution spargers are fabricated using seamless Type 304/304L or 316/316L stainless steel pipe. The distribution spargers include a flanged end connector to facilitate removal as necessary for maintenance, repair, and replacement.

ICC Piping

Piping in contact with ICS pool water is made of Type 304/304L or 316/316L seamless, stainless steel. ICC piping in contact with PCW on the cold side of the HXs is made of seamless Grade B carbon steel. The ICC piping is designed to comply with ASME B31.1 (Reference 9A.2.6-3) requirements. Drainage taps are provided at low points in the ICC piping to allow the pipes to be drained to facilitate maintenance or repair activities. Vents are provided at the high points in the ICC piping to allow air to be purged from the ICC piping when filling with pool water and to allow

the pool return pipes to be drained to facilitate maintenance or repair activities. Structural supports for piping systems and components inside the ICS Pool Compartment are in accordance with the rules and requirements of ASME BPVC-III NF, Subsection NF, "Supports" (Reference 9A.2.6-4).

ICC Valves

Valve surfaces in contact with ICS pool water are made of Type 304/304L or 316/316L stainless steel. The ICC valves are designed to comply with ASME B31.1 and ASME B16.34 (Reference 9A.2.6-5) requirements. Valve internals in the ICC piping in contact with PCW is made of Type 304/304L or 316/316L stainless steel to maintain leak-tightness. Valves are provided with flanged inlet and outlet connections to facilitate removal as necessary for maintenance, testing, and replacement. The ICS pool suction isolation valve and pool return isolation valves are equipped with manual override and handwheel.

Instrument Requirements

ICC instrument housings and internals in contact with ICS Pool Water are made of Type 304/304L or 316/316L stainless steel. Where possible, instrument housings are provided with flanged connections to facilitate removal as necessary for maintenance, testing, and replacement.

9A.2.6.4 Materials

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, and corrodents through material chemistry, heat treatment, contamination, and material processes controls.

Components and surfaces that come in contact with ICS pool water are manufactured using corrosion-resistant material. Use of corrosion-resistant material reduces iron particulate introduced into the system and Flow Accelerated Corrosion (FAC).

9A.2.6.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.2.6-1 for ICC interfaces with other equipment or systems.

9A.2.6.6 System and Equipment Operation

The ICC provides active cooling and purification of the ICS pools when it is operating.

The ICC cooling function is required when the ICs are in standby mode due to constant thermal leakage caused by a relatively small amount of condensing steam in the IC during normal reactor operations. The Adjustable Speed Drives allow the ICC to operate at a lower system flow rate during normal operations when the ICs are in standby mode for improved overall plant economy.

The ICC purification function is provided by directing ICC flow through the ICC Demineralizer, which is the normal flow path during routine system operation. The Demineralizer is bypassed for the following conditions:

1. The Demineralizer is bypassed if the temperature of the process fluid exceeds 53.3 °C to prevent thermal degradation of the demineralizer resins and early desorption of adsorbed contaminants that occurs with aging.
2. The Demineralizer is bypassed during the injection of chemicals into the process fluid to inhibit corrosion and biological growth in the ICS Pool Cooling and Cleanup System equipment, piping, and IC pools to prevent chemical degradation of the demineralizer resins.

9A.2.6.6.1 Normal Operation

Refer to Table 9A.2.6-2, "BWRX-300 Reactor and ICC Operating Modes" which provides a summary of the plant operating modes with corresponding ICC operating modes.

Mode 1: Power Operation

During Mode 1 power operation, the ICC is in service for all system operating modes:

- ICC Mode A2 – Both ICC trains A and B are operating for maximum heat removal from the ICS pools
 - The Demineralizer is bypassed during Mode A2
- ICC Mode A1a – Only ICC Train A is operating to purify ICS pool water and remove heat
 - The Demineralizer is bypassed if the HX effluent temperature exceeds 53.3 °C
- ICC Mode A1b - Only ICC Train B is operating to purify ICS pool water and remove heat
 - The Demineralizer is bypassed if the HX effluent temperature exceeds 53.3 °C
- ICC Mode A2c – Both ICC trains are operating for injection of water treatment chemicals
 - The Demineralizer is bypassed during chemical injection

One ICC pump/HX train can be isolated as needed for maintenance during power operation. Demineralized makeup water is automatically added to maintain ICS pool inventory during normal reactor operations.

Processed water is returned to individual ICS cubicles as required by aligning the ICS pool isolation valves to the desired ICS cubicle pools.

Mode 2: Startup Operation

During startup operation, the ICC is in service for all system operating modes described for power operation.

Mode 3: Hot Shutdown Operation

During hot shutdown operation, the ICC is in service for all system operating modes described for power operation.

Mode 4: Stable Shutdown Operation

During stable shutdown operation, the ICC is in service for all system operating modes described for power operation. In addition, the ICC can be completely removed from service during stable shutdown operation for system maintenance.

Mode 5: Cold Shutdown Operation

During cold shutdown operation, the ICC is in service for all system operating modes described for power operation. In addition, the ICC can be completely removed from service during cold shutdown operation for system maintenance.

Mode 6: Refueling Operation

During refueling operation, ICC is in service for all system operating modes described for power operation. In addition, ICC can be completely removed from service during refueling operations for system maintenance.

9A.2.6.6.2 Off-Normal Operations

ICC does not have any Off-Normal operating modes.

9A.2.6.6.3 System Shutdown

The ICC can be completely removed from service for system maintenance during the following plant operating modes:

- Mode 4: Stable Shutdown
- Mode 5: Cold Shutdown
- Mode 6: Refueling

9A.2.6.7 Instrumentation and Control

Non-Safety-Category function instrumentation and controls (Chapter 7) provides control signals to ICC pumps and valves to initiate system operation and direct flow through system piping branches as needed. The Non-Safety function instrumentation and controls also receives data signals from instruments installed in the ICC piping to measure local process flow variables: water level, temperature, pressure, differential pressure, and conductivity. A local instrumentation monitoring panel is provided in the ICC equipment area for maintenance, diagnostic, and functional testing of the ICC and individual components.

Where possible, instrument housings are provided with flanged connections to facilitate removal as necessary for maintenance, testing, and replacement. Instruments are installed such that there is space surrounding the instrument to facilitate ready access to the instrument and the ability to perform maintenance or replacement.

ICC Control and Alarm and Monitoring functions are identified in Table 9A.2.6-3.

9A.2.6.8 Monitoring, Inspection, Testing, and Maintenance

Testing of the ICC is performed to demonstrate proper system and component functioning. ICC functionality is continuously demonstrated during normal plant operation. Provisions are made for periodic inspection of major components to ensure the capability and integrity of the system. Service platforms are installed to facilitate inspection of the Surge Tank and Return Guard Pipe. The design of the ICC allows for the inspection and testing of components under normal operating conditions.

Material and equipment selection for the ICC is based on a 60-year design life, with appropriate provisions for maintenance and replacement. Components that require replacement prior to the end of the 60-year design life include, but are not limited to, electrical and electronic equipment, gaskets and seals, lubricants, valve disks and internal components such as seats and packing, and bearings.

Isolation valves are provided near the ICC piping penetrations to the ICS Outer Pool A to permit maintenance and repair activities to be performed on pipe section below the ICS pools. Valves are installed with flanged connections to facilitate removal as needed for testing or repair. In addition, valves and other components are installed with plenty of space surrounding the component to facilitate ready access and the ability to perform maintenance, repair, or replacement activities.

Maintenance isolation valves and flanged connections are provided for the pumps and HXs so they can be isolated and removed for repair or replacement as required. Drain taps are provided at low points in piping sections to evacuate process water for maintenance and repair activities.

A manual crossover feature allows for simultaneous maintenance on the pump of one train with the HX on the other train without interrupting normal cooling and cleanup operation. This design feature adds flexibility for maintenance purposes thereby increasing ICC reliability and availability. The design of the ICC provides adequate equipment removal paths and personnel access for maintenance and repair activities and equipment replacement.

Testing is performed to ensure required functional operability is maintained under design conditions. Testing is performed in accordance with plant procedures. Testing in support of plant pre-operational testing, startup, and commissioning is addressed in Chapter 14, Section 14.3.

9A.2.6.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.2.6.10 Performance and Safety Evaluation

Two features have been incorporated into the ICC design to increase system reliability.

1. A crossover pipe with manual isolation valve is installed between the A and B trains pump discharge to allow the system to remain operable in the event Pump A and HX B are simultaneously out of service for maintenance activities, or if Pump B and HX A are simultaneously out of service.
2. Manual overrides with handwheels are specified for the main suction and for the return Air-Operated Block Valves to enable manual operation of these valves in the event of actuator failure.

Safety Category functions provided by the ICC are associated with the suction and return pipe penetrations into the ICS pools. The Suction Surge Tank and Return Guard Pipe are categorized as SC1 and Seismic A components and function to prevent draining of the ICS Outer Pools in the event of a break in ICC piping below the ICS pools.

9A.2.6.11 References

- 9A.2.6-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.2.6-2 ANSI/HI 1.3, "Rotodynamic Centrifugal Pumps for Design and Application," Hydraulic Institute.
- 9A.2.6-3 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 9A.2.6-4 ASME BPVC-III NF, "BPVC Section III – Rules for Construction of Nuclear Facility Components – Division 1 – Subsection NF – Supports," American Society of Mechanical Engineers.
- 9A.2.6-5 ASME B16.34, "Valves – Flanged, Threaded, and Welding End," American Society of Mechanical Engineers.

Table 9A.2.6-1: Isolation Condenser System Pool Cooling and Cleanup System Interfaces

Interfacing System	Interface Description	Interface Boundary
Non-Safety Instrumentation and Control System	Provides control logic for initiation and control of the ICC. Receives data signals from ICC instruments for display on MCR and local panels Input to ICC: Control signals to pumps and valves Output from ICC: Instrumentation data	Remote Multiplexer Unit associated with ICC instrument and control equipment
Process Radiation and Environmental Monitoring System	Provides process sampling for water quality to detect potentially radioactive contaminated Demineralizer resin before discharge to SWM Output from ICC: Process water samples for analysis for water quality and monitoring Demineralizer performance	ICC main Return Pipe Tie-ins ICC Demineralizer
Isolation Condenser System	Provides process fluid from ICS pools. Provides pool temperature and water level conditions via instruments in the ICS Outer Pool at ICC suction interface Input to ICC: Process fluid and inlet conditions (pool temperature and water level) Output from ICC: Processed fluid is returned to the ICS cubicles	ICS Pool Piping Penetrations ICS A, B, and C Temperature Elements ICS Outer Pool TE and Level Transmitter
Control Panel System	Provides control logic for initiating operation and control of the ICC. Receives data signals from ICC instruments for display on MCR and local panel Input to ICC: Control signals to pumps and valves Output to from ICC: Instrumentation data	Remote Multiplexer Unit associated with ICC instrument and control equipment
Solid Waste Management System	Receives discharged demineralizer resin from ICC Demineralizer Output from ICC: Discharge of potentially radioactively contaminated spent resin media from the Demineralizer	ICC Demineralizer Interface is at downstream side of ICC isolation valve

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Interfacing System	Interface Description	Interface Boundary
Plant Pneumatics System	Provides compressed air for ICC Instrumentation and Control (I&C) components, AOVs, and Demineralizer Input to ICC: Compressed air service for: ICC I&C; Demineralizer resin media redistribution and exchange evolutions	ICC AOVs ICC Demineralizer Interface for compressed air is at upstream side of ICC isolation valve
Non-Safety Electrical Distribution System	Provides Non-Safety-Category AC electrical power to ICC equipment Input to ICC: Non-Safety-Category electrical power to equipment electrical loads in ICC	Interface to be located at local breaker box
Heat, Ventilation, and Cooling System	Provides HVAC for environmental temperature control of ICC Equipment Room Input to ICC: HVAC air flow Output from ICC: Heat load from ICC equipment	Interface at ICC Equipment Room
Reactor Building Structure	Provides space, structural and infrastructure support, and protection for ICC piping and equipment Input to ICC: ICC equipment is situated within dedicated space provided by the RBS	Interfaces at: (1) ICC Equipment Room, (2) ICS piping penetrations, (3) ICS pools (pipe supports)
Water, Gas, and Chemical Pads	Provides makeup demineralized water for ICS pools during normal reactor operations Input to ICC: Demineralized water to replace routine pool losses due to evaporation	Interface for makeup water is at upstream side of ICC isolation valve
Water, Gas, and Chemical Pads	Provides demineralized water to ICC Demineralizer. Input to ICC: Demineralized water to facilitate Demineralizer resin media exchange evolutions	Demineralizer Interface for demineralized water is at upstream side of ICC isolation valve

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**Table 9A.2.6-2: Isolation Condenser System Pool Cooling and Cleanup System
Operating Modes**

Mode	Title	Reactor Mode Switch Position	ICC Modes	Description
1	Power Operation (10 – 100% Rated Power)	RUN	A2	Cooling (A & B Trains Operating)
			A1a	Cleanup (A Train Operating)
			A1b	Cleanup (B Train Operating)
			A2c	Chemical Injection
2	Startup	STARTUP or REFUEL ⁽¹⁾	A	All ICC Operating Modes
3	Hot Shutdown ⁽¹⁾	SHUTDOWN	A	All ICC Operating Modes
4	Stable Shutdown ⁽¹⁾	SHUTDOWN	A	All ICC Operating Modes
			B	ICC Shutdown for Maintenance
5	Cold Shutdown ⁽¹⁾	SHUTDOWN	A	All ICC Operating Modes
			B	ICC Shutdown for Maintenance
6	Refueling ⁽²⁾	SHUTDOWN or REFUEL	A	All ICC Operating Modes
			B	ICC Shutdown for Maintenance

(1) All RPV head closure bolts fully tensioned

(2) One or more RPV head closure bolts less than fully tensioned

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**Table 9A.2.6-3: Isolation Condenser System Pool Cooling and Cleanup
System Instrumentation**

Instrument	Location	Function
Temperature Element	ICS Cubicle Pools	Control Temperature of ICS Cubicle Pools used to initiate ICC A2 operation
Temperature Element	Suction Surge Tank	Alarm and Monitoring
Level Transmitter	Suction Surge Tank	Control ICS Pool Water Level used to control automatic addition of makeup demineralized water and shutoff pumps if water level is too low
Pressure Transmitter	Pump Suction	Alarm and Monitoring Suction Pressure
Pressure Transmitter	Pump Discharge	Alarm and Monitoring Discharge Pressure
Pressure Differential Transmitter	Pump Discharge	Control Pressure Drop across Flow Element used to control pump speed to adjust Flow Rates in A and B Trains
Flow Element (Orifice Plate)	Pump Discharge	Alarm and Monitoring Flow Rate in A and B Trains
Temperature Element	HX Inlet (Hot Side)	Alarm and Monitoring Temperature of PCW entering HX Hot Side
Pressure Differential Transmitter	HX Inlet and Outlet (Hot Side)	Alarm and Monitoring Pressure Drop across HX Hot Side
Temperature Element	HX Outlet (Hot Side)	Alarm and Monitoring/HX Condition Monitoring
Temperature Element	Main System Return	Control Temperature of HX effluent into Demineralizer
Conductivity Transmitter	Main System Return	Alarm and Monitoring Pre- and post- Demineralizer Conductivity of return fluid
Pressure Differential Transmitter	Demineralizer Inlet and Outlet	Alarm and Monitoring Pressure Drop across Demineralizer
Pressure Differential Transmitter	Return pipes to Individual IC Cubicles	Alarm and Monitoring Pressure Drop across Flow Element
Flow Element (Orifice Plate)	Returns to Individual IC Cubicles	Alarm and Monitoring Flow Rate in Return Piping

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Instrument	Location	Function
Pressure Transmitter	Makeup Water Line	Alarm and Monitoring Pressure of Makeup Demineralized Water
Pressure Differential Transmitter	Makeup Water Line	Alarm and Monitoring Pressure Drop across Flow Element on Makeup Demineralized Water
Flow Element (Orifice Plate)	Makeup Water Line	Alarm and Monitoring Flow Rate on Makeup Demineralized Water
Pressure Differential Transmitter	HX Inlet (Cold Side)	Alarm and Monitoring Pressure Drop across Flow Element on HX Cold Side
Flow Element (Orifice Plate)	HX Inlet (Cold Side)	Alarm and Monitoring Flow Rate on HX Cold Side
Temperature Element	HX Inlet (Cold Side)	Alarm and Monitoring Temperature entering HX Cold Side
Pressure Differential Transmitter	HX Inlet / Outlet (Cold Side)	Alarm and Monitoring Pressure Drop across HX Cold Side
Temperature Element	HX Outlet (Cold Side)	Control Temperature exiting HX Cold Side used to control Demineralizer Bypass and flow of PCW into HX
AOV Actuators	All AOV Actuators	Alarm and Monitoring Valve Position Indication Limit switches to detect current position (open/closed) of AOVs

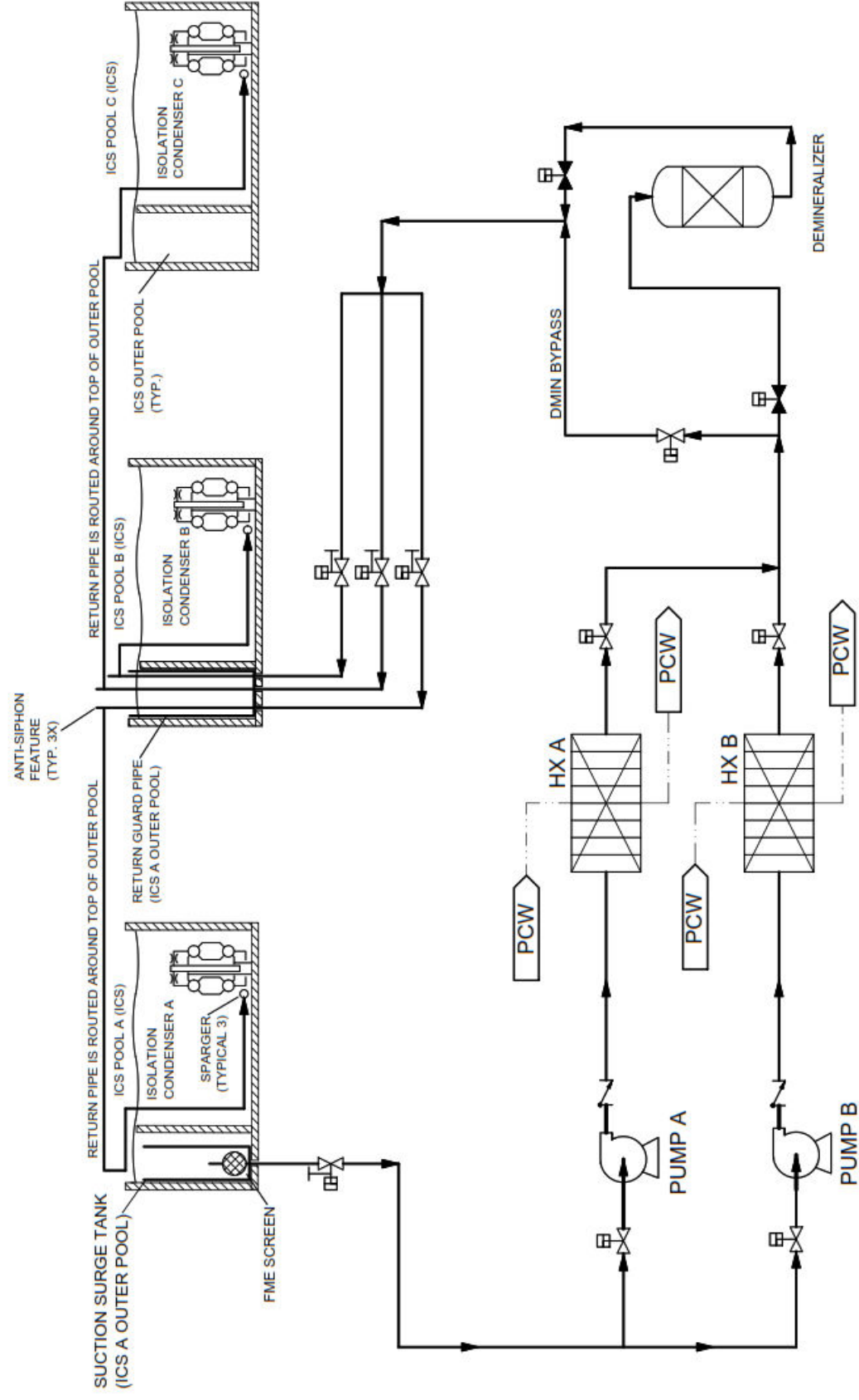


Figure 9A.2.6-1: Isolation Condenser System Pool Cooling and Cleanup System

9A.3 Process Auxiliary Systems

9A.3.1 Process Sampling Systems

The Process Sampling System is a subsystem of the Process Radiation and Environmental Monitoring System. In addition to the Process Sampling subsystem, the PREMS is comprised of the following subsystems, Process Radiation Monitoring (Chapter 11, Subsection 11.5), Area Radiation Monitoring (Chapter 12, Subsection 12.3.14), and Containment Monitoring (Chapter 12, Subsection 12.3.14).

The Process Sampling subsystem is included in the design of the BWRX-300 based upon the requirements of REGDOC 2.5.2 (Reference 9A.3.1-1) Section 8.13.3 "Radiation Monitoring" as applicable.

The BWRX-300 design does not employ a Post-Accident Sampling System for assessing core damage as per REGDOC 1.1.2 (Reference 9A.3.1-2). In its place is a collection of Post-Accident Monitoring (PAM) inputs utilizing real-time measurements of plant parameters which are deemed more accurate and/or more timely to measure core damage. The sensors and instruments designated for PAM are treated differently by the plant I&C architecture to ensure availability during and after the most severe accident or Beyond Design Basis Accident. Many of the other Post-Accident Sampling System samples were not needed for short severe accident scenario or the information is available from post-accident monitoring instruments such as radiation monitors, (containment, main steam and offgas) temperature and water level as discussed in Reference 9A.3.1-3.

9A.3.1.1 System and Equipment Functions

The Process Sampling subsystem collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance. The Process Sampling subsystem performs the following functions during normal and off-normal conditions.

9A.3.1.1.1 Normal Function (Non-Safety Category)

The Process Sampling subsystem collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance. This process subsequently guides changes to operating parameters. This subsystem is designed to function during all plant operational modes under individual system requirements. The Process Sampling subsystem does not perform or ensure any Safety Class function. Therefore, this system has no Safety Class design basis.

9A.3.1.1.2 Normal Function (Safety Category)

This subsystem does not perform any Safety-Category functions during normal conditions.

9A.3.1.1.3 Off-Normal Functions (Non-Safety-Category)

This subsystem does not perform any Non-Safety-Category functions during Off-Normal conditions.

9A.3.1.1.4 Off-Normal Functions (Safety-Category)

This subsystem does not perform any Safety-Category functions during Off-Normal conditions unless designated a Post-Accident Monitoring function.

9A.3.1.2 Safety Design Bases

The Process Sampling subsystem does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

9A.3.1.3 Description

The Process Sampling subsystem collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance.

Process streams not requiring continuous sampling or computer monitoring include provisions for obtaining grab samples. Sample lines are routed to a common sample station, local sample station (or simply local sample taps) depending on plant layout and process fluid.

The Process Sampling subsystem consists of the following: permanently installed sample lines, sampling panels with analyzers and associated sampling equipment, provisions for grab sampling, and permanent shielding. The division of sample lines versus grab samples are assessed during detailed design activities based on the availability of automation in the instrumentation, the need to limit personnel access (ALARA), the frequency of needed measurements, and other considerations. Sampling stations and associated process samples are summarized by location below.

Sample Stations:

- Reactor Water Cleanup System
- Shutdown Cooling System
- Control Rod Drive System
- Fuel Pool Cooling and Cleanup System
- Isolation Condenser System
- Isolation Condenser Pool Cooling and Cleanup System
- Condensate and Feedwater System
- Moisture Separator Reheater System
- Heater Drain and Vent System (Condensate and Feedwater subsystem)
- Nuclear Boiler System Main Steam Lines
- Condensate Filter and Demineralizer System
- Main Turbine Equipment
- Main Condenser
- Liquid Waste Management
- Solid Waste Management

Local Grab Sample Stations:

- Plant Cooling Water System
- Water, Gas, and Chemical Pads (Plant Service Water and Makeup Water are subsystems)
- Chilled Water System

- Circulating Water System
- Equipment and Floor Drain System

Containment isolation function is applied to all mechanical instrument sensing line penetrations of the containment boundary in a manner that provides the highest reliability of maintaining instrument function while limiting potential radioactive release if an instrument line is ruptured outside the containment boundary.

The Process Sampling subsystem monitors plant liquids and gases utilizing components designed and installed in such a way as to minimize interconnection of radioactive and non-radioactive systems. To the extent practicable, Process Sampling subsystem interconnections to non-radioactive systems are limited to purge air, purge water and makeup water for filling loop seals. The designs of these interconnections prevent contamination of the non-radioactive system or process due to leakage, spillage, valving errors, or other operating conditions.

The Process Sampling subsystem is designed to minimize facility and environmental contamination and minimize the generation of radioactive waste. These objectives are supported by the procedures for operation, and are controlled as follows:

- Providing atmospheric purging of the internal portion of air sampling skids as necessary
- Providing the ability for liquid flushing of the internal portions of liquid sampling skids as necessary
- Designing the interior portions of liquid and gaseous sampling chambers to minimize the plate out of radioactive material
- Designing sample extraction points such that they minimize the potential for spillage and contamination of adjacent areas
- Minimizing the generation of liquid radioactive waste by minimizing the amount of a sample that needs to be extracted, consistent with laboratory and sensitivity requirements

9A.3.1.3.1 Component Description

The Process Sampling subsystem consists of sample lines, sampling panels with analyzers and associated sampling equipment, provisions for grab sampling, and shielding.

9A.3.1.4 Materials

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials specifically from IGSCC (as applicable) through material chemistry, heat treatment, contamination and material processes controls.

9A.3.1.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.3.1.1-1 which provides the description and boundary for each interfacing system.

9A.3.1.6 System and Equipment Operation

The Process Sampling subsystem operates continuously or in grab sample mode when the interfacing system being monitored is operational.

9A.3.1.6.1 Initial Configuration (Pre-Startup)

The Process Sampling subsystem operates continuously or in grab sample mode.

9A.3.1.6.2 System Startup

The Process Sampling subsystem operates continuously or in grab sample mode.

9A.3.1.6.3 Normal Operations

Sampling equipment operates continuously with capability for grab samples from each process stream.

Root valves for process system sampling points are included in the Process Sampling subsystem. The safety requirements for root valves are determined by the safety requirements for the particular system.

9A.3.1.6.4 Off-Normal Operations

There are no mitigating system functions tied to the Process Sampling subsystem.

9A.3.1.6.5 System Shutdown

Process Sampling subsystem is performed in accordance with detailed operation procedures.

9A.3.1.7 Instrumentation and Control

The Process Sampling subsystem is supported by the SCN Distributed Control and Information System (Chapter 7, Subsection 7.3.4). SCN classified functions are implemented on a hardware/software platform to provide the reliability needed to prevent controller failure from becoming an Anticipated Operational Occurrence (AOO).

9A.3.1.8 Monitoring, Inspection, Testing, and Maintenance

The Process Sampling subsystem design provides continuous availability regardless of the plant operating mode; therefore, Process Sampling subsystem functionality is continuously demonstrated during all phases of normal, shutdown, and off-normal conditions.

Routine maintenance and cleaning of the sampling subsystem is performed. The design accommodates easy access to sensors, instruments, and panels for maintenance. Additionally, devices for lifting or hoisting the Process Sampling subsystem components is provided to facilitate replacement if required.

Instrumentation is designed to facilitate calibration checks and troubleshooting.

Sampling racks and electronic modules are serviced and maintained in accordance with the operational instructions to ensure reliable operation. Such maintenance includes servicing and replacement of defective components and adjustments as required. Periodic testing or calibration checks is performed as part of the maintenance plan.

9A.3.1.9 Radiological Aspects

Chapter 12, Subsection 12.3.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.3.1.10 Performance and Safety Evaluation

The system design provides continuous availability regardless of the plant operating mode. SCN functions are implemented on the Non-Safety Instrumentation and Control System which provides the reliability needed to prevent controller failure from becoming an AOO. The Process Sampling subsystem performs no Safety-Category functions and is not required to prevent or mitigate the consequences of a design basis accident, to shut down the reactor and maintain safe shutdown

conditions, or to maintain the integrity of the reactor coolant pressure boundary. Therefore, a nuclear safety evaluation is not required.

9A.3.1.11 References

- 9A.3.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.3.1-2 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 9A.3.1-3 NEDO-32991-A, "Regulatory Relaxation for BWR Post- Accident Sampling Stations (PASS)," GE Nuclear Energy.

Table 9A.3.1-1: Process Sampling Subsystem Interfaces

Interfacing System	Interface Description	Interface Boundary
Distributed Control and Information Systems	DCIS Non-Safety Class instrumentation communication	Instrumentation communicates with the SCN DCIS system
Nuclear Boiler System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Isolation Condenser System	Process sampling for chemistry control	Local Grab Sample and/or Routed to Sample Station
Control Rod Drive System	Process sampling for chemistry control	Routed to sample station
ICS Pool Cooling and Cleanup System	Process sampling for radiological analysis, chemistry control and demineralizer performance monitoring	Routed to Sample Station
Shut Down Cooling System	Process sampling for chemistry control	Routed to Sample Station
Reactor Water Cleanup System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Fuel Pool Cooling and Cleanup System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Liquid Waste Management System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Solid Waste Management System	Process sampling for radiological analysis, chemistry control, and demineralizer performance monitoring	Routed to Sample Station
Condensate and Feedwater System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Condensate Filters and Demineralizers System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Main Turbine Equipment	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Moisture Separator Reheater System	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Main Condenser and Auxiliaries	Process sampling for radiological analysis and chemistry control	Routed to Sample Station
Circulating Water System	Process sampling for chemistry control	Local grab sampling
Chilled Water Equipment	Process sampling for chemistry control	Local grab sampling

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Interfacing System	Interface Description	Interface Boundary
Plant Cooling Water System	Process sampling for chemistry control	Local grab sampling
Equipment and Floor Drain System	Process sampling for radiological analysis and chemistry control	Local grab sampling
Water, Gas, and Chemicals Pads	Process sampling for Demineralized Water Tank chemistry control	Local grab sampling

9A.4 Air and Gas System

9A.4.1 Plant Pneumatic System

The Plant Pneumatics System provides a continuous supply of compressed air for the majority of the plant air demands. In addition, PPS supplies oil-free air to service boxes for use of portable breathing air filtration systems. The PPS also distributes nitrogen to users inside containment.

9A.4.1.1 System and Equipment Functions

Normal Functions (Non-Safety-Category)

PPS provides the following Non-Safety-Category functions:

1. The PPS provides dry, oil-free, filtered compressed air for valve actuators, instrument control functions, air operated tools, and miscellaneous equipment/services outside of containment in Mode 1, 2, 3, 4, 5, and 6.
2. The PPS provides the compressed air for tank sparging, filter/demineralizer backwashing, air-operated tools and other services requiring air of lower quality in Mode 1, 2, 3, 4, 5, and 6.
3. The PPS provides the distribution of gaseous nitrogen, supplied by the Containment Inerting System (CIS) (Subsection 9A.4.2), to primary containment for valve actuators and other primary containment users in Mode 1, 2, 3, 4, and 5.
4. The PPS supplies instrument-quality air to nitrogen consumers when primary containment is open for personal access (e.g., major outages).
5. The PPS supplies oil-free air to service boxes for breathable air.

Normal Functions (Safety-Category)

The PPS provides dry, oil-free, filtered compressed air for Safety-Category valve actuators and instrument control functions outside of containment (Safety Class 3).

The PPS provides the distribution of gaseous nitrogen, supplied by CIS, to Primary Containment for Safety-Category valves, controls, instrumentation, and other Primary Containment users (Safety Class 3).

While these functional requirements are categorized as SC3, not all associated air demands are expected to be SC3.

Off-Normal Functions (Non-Safety-Category)

The power supply to the PPS automatically switches to standby AC power during a LOOP.

Off-Normal Functions (Safety-Category)

As part of Defense Line 3 the PPS provides containment isolation valves on piping that penetrates the containment boundary. These valves are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System (Chapter 7, Subsection 7.3.1).

Containment isolation is provided in accordance with CNSC REGDOC-2.5.2 Section 8.6.6 (Reference 9A.4.1-1). Refer to Chapter 6, Subsection 6.3.4 for information pertaining to containment isolation. In addition, the design of the PPS meets the requirements specified in CNSC REGDOC-2.5.2 Section 7.1 as related to ensuring that the Non-Safety-Category function portions of the PPS are isolated from the Safety-Category function portions of the PPS.

9A.4.1.2 Safety Design Bases

The PPS is Safety Class 3, providing integral support functions for DL2 functions:

As part of the PPS Safety-Category 3 functions the PPS provides dry, oil-free, filtered, compressed air for valve actuators, Safety-Category instrument control functions, and general instrumentation services outside of the Steel-plate Composite Containment Vessel (SCCV). The PPS provides the distribution of gaseous nitrogen, supplied by the CIS, to SCCV for valve actuators and other SCCV users.

Containment Isolation Valves are provided on piping that penetrates the containment boundary. These valves are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System.

9A.4.1.3 Description

Figure 9A.4.1-1 depicts the PPS.

The PPS consists of two 100% air compressors, two 100% capacity dryer trains, two service air receivers operating in parallel upstream of the dryers, and two instrument air receivers operating in parallel downstream of the dryers. The PPS provides service and instrument air for the majority of plant air needs.

Each compressor takes suction through an air intake filter/silencer which brings air from outside the TB. The air inlet is fitted with rain hood and bird screen. The lead compressor supplies air to both service air receivers, which will flow through the lead dryer train, to both instrument air receivers.

Distribution piping in the plant is by ring headers which allows for flexibility in distribution locations, maintenance, and ease of accommodating future requirements. The service air receivers distribute air to the service air header demands. The service air subsystem also connects to a separate utility containment penetration to provide service air to the service boxes inside containment. These service boxes are used to supply air to portable breathing air filtration systems to provide breathing air hookups to personnel entering containment when the atmosphere is not yet fully habitable or for areas where high levels of airborne contaminants cannot be eliminated efficiently by the HVAC.

The instrument air receivers will distribute air to the instrument air header demands. The instrument air subsystem also includes piping penetrating containment, connecting the CIS to pneumatic valves and other compressed gas users inside containment. During plant outages, the PPS supplies the pneumatic valves and other compressed gas users inside containment with instrument-quality air while the nitrogen supply from the CIS is isolated from inside containment.

During normal operation, the air supply is isolated from the primary containment vessel via double block and bleed valve configuration so that nitrogen from the CIS can be supplied to nitrogen users inside containment using the PPS primary containment piping penetrations. The block and bleed valves on the air supply prevent air from entering containment during normal operation and posing an explosion and fire risk. During plant shutdown, the nitrogen supply is isolated from the primary containment vessel via double block and bleed valve configuration, so that in the case of a leak in the PPS supply piping to primary containment, the primary containment does not turn into a confined space or pose an asphyxiation risk to plant personnel.

PPS is designed to meet the applicable requirements of Instrument Society of America 7.0.01 (Reference 9A.4.1-2).

9A.4.1.3.1 Component Description

The following paragraphs provide information that pertains to the major equipment items in the PPS system.

Air Compressors

The two (2) compressors are oil-free, rotary screw, water-cooled, electric motor driven compressors. Each compressor can supply 100% of the system's continuous air requirements. Noise reduction methods are incorporated to ensure that noise levels in habitable areas near the compressors are within the guidelines for industrial environments. The air compressor units are powered from the two (2) separate electrical busses. The service air piping includes a connection for adding an additional compressor. These compressors are commercially available skid packages.

Air Receivers

The system includes two service air receivers in parallel and two instrument air receivers in parallel. The design capacity of the receivers includes adequate reserve capacity to provide at least 10 minutes of operating time following a trip with no air compressors in operation before the low pressure set point alarms are sounded in the control room.

The receivers are ASME VIII designed and stamped.

A pressure-relief safety valve is installed on each receiver.

Dryer/Filter Trains

The PPS utilizes two (2) 100% dryer trains. The dryers are regenerative desiccant air dryers.

The dryer trains include coalescing pre-filters and after-filters. The 100% capacity pre-filters are included to remove water and oil droplets, rust, dust, and other solid objects suspended in the air. The 100% capacity after-filters are included to remove particles of desiccant that may be carried away from the dryer. During normal operation, one dryer/filter train is continuously operating, and the other train is in standby. If both compressors are operating, both dryers are placed in service, if available.

The dryer filter trains deliver compressed air that meets ISA 7.0.01 (Reference 9A.4.1-2) Quality Standards for Instrument Air.

Piping and Valves

All PPS piping is welded stainless steel material

The piping, other than that required for containment isolation, is designed to meet the requirements of ASME B31.1.

To ensure containment integrity at the Steel Composite Containment Vessel penetrations, the containment piping and isolation valves are designed to Seismic Category A and B, respectively; ASME BPVC, Section III, Division 1-Subsection NCD; and Quality Group B requirements.

PPS containment isolation piping includes an AOV outside of containment and a check valve inside of containment. During a Design Basis Accident (DBA), if the AOV fails to close leakage out from inside containment is stopped by the air system if an air compressor is operating or by the check valve if an air compressor is not operating.

9A.4.1.4 Materials

The PPS is designed utilizing materials that ensure that the functional requirements of the system are achieved. Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials through material chemistry, heat treatment, contamination, and material processes controls.

9A.4.1.5 Interfaces with Other Equipment or Systems

Table 9A.4.1-1 identifies PPS interfaces with other BWRX-300 Systems.

9A.4.1.6 System and Equipment Operation

The PPS operates in the normal plant environment and provides Non-Safety and SC3 functions. With the exception of the containment isolation valves the PPS is not required to operate after a design basis accident.

The system is designed for continuous operation during all modes of plant operation:

Normal Operational Concept

During normal operation, one 100% capacity air compressor operates to supply both service air receivers, one dryer train, both instrument air receivers, and both instrument and service air system distribution piping.

A compressor is chosen as the lead, continuously operating compressor. The other compressor serves as standby.

The standby compressor automatically starts upon trip of the lead compressor, or when the air pressure in the system drops below the predetermined pressure set point. The assignment for lead and standby air compressors is switched periodically by operators in the MCR.

One 100% dryer/filter train operates, and one serves as standby. Automatic start of the standby dryer skid occurs when the differential pressure across a dryer filter, discharge pressure or the dewpoint reach predetermined setpoints. The assignment for lead and standby dryers is switched periodically by operators in the MCR.

Off-Normal Operational Concept

The PPS containment isolation valves and associated piping are designed to be functional after a safe shutdown earthquake.

When a LOOP is detected, the operating compressor and dryer both trip and restart when power is received from the backup diesel generators.

9A.4.1.7 Instrumentation and Control

9A.4.1.7.1 Instrumentation

The PPS contains instrumentation to remotely monitor (indicate and alarm) and control all equipment performance, control valve status, and necessary system parameters, such as pressure, temperature, and moisture content of the compressed air in the system. Data signals from the PPS instruments and equipment are displayed on the Main Control Room (MCR) and local instrumentation panels. Local control panels for the compressors and dryers are used to operate during testing, calibration, and changing standby compressor and dryer units.

Instrumentation associated with one or more control functions are redundant. For additional information refer to Chapter 7, Subsection 7.3 "Distributed Control and Information System

Functions". This includes instruments such as vendor package (skid) instrumentation that send equipment trip signals or standby equipment start signals upon detecting specified abnormal operating conditions. Additionally, service and instrument air header pressure sensors that provide the main control parameters for the PPS are redundant, described in the Controls section below.

9A.4.1.7.2 Controls

The PPS is automatically controlled from the plant Distributed Control and Information System (DCIS). The start and stop of the PPS compressors and dryers are automatically or manually operated from the MCR. The instrumentation and control architecture in the PPS is designed to keep the instrument air available in all modes of operation. All automatic or remotely actuated valves in the PPS fail in the position most likely to keep the system operational and controllable with the exception of the PPS containment isolation valves.

The pressure in the service air header is the main control parameter in the system. The operating compressor loads/unloads based on pressure readings from the redundant pressure sensors on the service air header. Upon a loss of both compressors, or at a low pressure setpoint, the normally open service air isolation valve closes to shed the service air system load and conserve air for the instrument air subsystem to maintain control of the plant. Upon a loss of both dryers, or a low-low pressure indication from the redundant instrument air header pressure sensors, the dryer bypass line opens to allow the flow of service air to the instrument air users.

9A.4.1.8 Monitoring, Inspection, Testing, and Maintenance

All major components have manual isolation valves to permit maintenance activities without having to shut down the PPS. The maintenance boundary between the isolation valves is designed to permit the simultaneous maintenance of the maximum number of components as possible without impacting the ability of the system to remain in service.

PPS piping and valves for the containment penetrations are tested in accordance with 10CFR50, Appendix J and CNSC REGDOC-2.5.2. Test and vent connections are provided at the containment isolation valves and are used to verify that the valves meet the local leak rate limits.

The PPS outboard containment isolation valve is tested to ensure operational integrity by manual operation of the valve. The inboard containment check valve is periodically tested to ensure valve operability.

All PPS equipment is arranged to provide adequate pull space and clearance for personnel and testing equipment access to facilitate, testing, repair, and maintenance.

Breathing air is sampled in accordance with CSA Z180.1:19 to ensure the CSA breathing air quality standards are met.

Instrument air and system functionality is tested in accordance with ISA 7.0.01 (Reference 9A.4.1-2) and associated guidelines to ensure the instrument air quality standards are met.

9A.4.1.9 Radiological Aspects

Chapter 12, Subsection 12.3 provides information pertaining to design measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.4.1.10 Performance and Safety Evaluation

The SC1 Safety-Category function performed by the PPS is containment isolation. Containment isolation is provided in accordance with CNSC REGDOC-2.5.2 Section 8.6.6 (Reference 9A.4.1-

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1). The PPS CIVs are designed to maintain the leak tightness of the containment in the event of an accident and prevent radioactive releases to the environment that exceed prescribed limits. The PPS incorporates features that ensure its operation over the full range of normal plant operations.

9A.4.1.11 References

9A.4.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

9A.4.1-2 ANSI/ISA-7.0.01, "Quality Standard for Instrument Air," Instrument Society of America.

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Table 9A.4.1-1: Plant Pneumatics System Interfaces

Interfacing System	Interface Description	Interface Boundary
Nuclear Boiler System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	NBS Equipment
Safety Class 1 Instrumentation and Controls System	Safety Class 1 Instrumentation and Controls System de-energizes the Containment Isolation Valves (CIVs) solenoids which release the air inside the actuators.	PPS Equipment
Safety Class 2 and 3 Instrumentation and Controls System	Safety Class 2 and 3 Instrumentation and Controls System provide signals to SC3 instrumentation and controls in PPS.	PPS Equipment
Non- Safety Instrumentation and Control System	Receive/provide signals to non-safety category instrumentation and controls in PPS.	PPS Equipment
Process Radiation and Environmental Monitoring System	PPS provides control and instrument air to AOVs and equipment/components as necessary. Provides periodic sampling of breathing air quality and instrument air quality.	PREMS AOVs
Isolation Condenser System	PPS provides control and instrument air to AOVs and equipment/components as necessary. including ICS isolation accumulators.	ICS AOVs
Refueling and Servicing Equipment System (RES)	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At RES Equipment
Boron Injection System (BIS)	PPS provides Service Air for periodic storage tank mixing through the in-tank air sparger.	At BIS Equipment
Control Rod Drive System/High Pressure Injection	PPS provides instrument air to the HCU scram valves, and CRD flow control AOVs and Alternate Rod Insertion valves.	CRD Equipment
Isolation Condenser System Pool Cooling and Cleanup System	PPS provides instrument air for ICC I&C components, AOVs, and DMIN to facilitate demineralizer resin conditioning and exchange evolutions.	ICC AOVs and DMIN
Shut Down Cooling System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At SDC Equipment
Reactor Water Cleanup System	PPS provides control and instrument air or nitrogen to AOVs.	At CUW AOVs

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Interfacing System	Interface Description	Interface Boundary
Fuel Pools Cooling and Cleanup	PPS provides control air for mixing of demineralizer resin and instrument-quality air for controlling the FPC AOVs.	At FPC AOVs and demineralizer resin tank
Liquid Waste Management System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At LWM AOVs
Solid Waste Management System	PPS instrument air to AOVs and dewatering pumps.	At SWM AOVs and SWM dewatering pumps
Offgas System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At OGS AOVs
Condensate and Feedwater Heating System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At CFS AOVs
Condensate Filters and Demineralizers System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At CFD AOVs
Main Turbine Equipment	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At MTE AOVs
Moisture Separator Reheater System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At MSR AOVs
Main Condenser and Auxiliaries	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At MCA AOVs
Circulating Water System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At CWS AOVs
Chilled Water Equipment	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At CWE AOVs
Plant Cooling Water System	PPS provides control and instrument air to AOVs and equipment/components as necessary. PCW provides cooling water to the air compressors.	At PCW AOVs and compressor cooler isolation valves

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Interfacing System	Interface Description	Interface Boundary
Hydrogen Water Chemistry	PPS provides control and instrument air to AOVs and equipment/components as necessary. The PPS also provides air supply for air injection to the OGS.	At Hydrogen Water Chemistry AOVs
Safety Class 2 and 3 Electrical Distribution System	Safety Class 2 and 3 Electrical Distribution System provides electrical power to PPS.	At PPS equipment
Primary Containment System (CON)	PPS provides CIVs on piping penetrating primary containment, and functions as an extension of the containment boundary.	At containment penetrations
Containment Inerting System	PPS provides control and instrument air to AOVs and equipment/components as necessary. CIS provides nitrogen to components inside of containment through the PPS.	At CIS AOVs
Fire Protection System (FPS)	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At FPS AOV
Equipment and Floor Drain System	PPS provides control and instrument air to AOVs and equipment/components as necessary.	At EFS AOVs
Reactor Building Structure	PPS provides hookups for pneumatic tools throughout the RBS.	At Building Structure
Turbine Building Structure (TBS)	PPS provides hookups for pneumatic tools throughout the TBS.	At Building Structure
Radwaste Building (RWB)	PPS provides hookups for pneumatic tools throughout the RWB structure.	At Building Structure

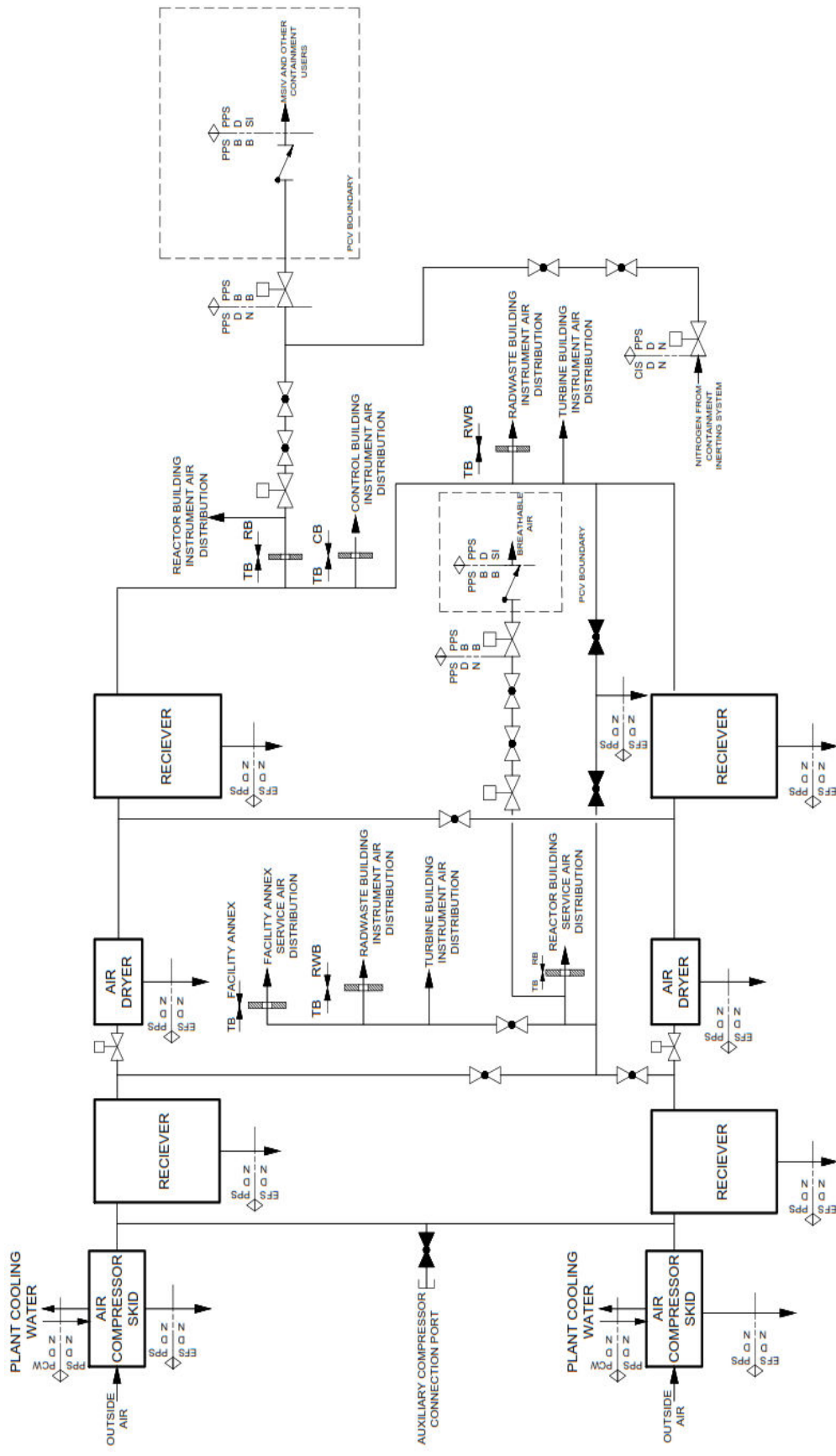


Figure 9A.4.1-1: Plant Pneumatic System

9A.4.2 Containment Inerting System

The Containment Inerting System is designed to establish and maintain an inert atmosphere (nitrogen) within the Steel-plate Composite Containment Vessel (SCCV). An inert atmosphere is maintained in all operating modes except during plant shutdown for refueling, equipment maintenance, and during limited periods of time to permit access for inspection at low reactor power. Prior to reactor shutdown the CIS can de-inert the SCCV allowing for safe personnel access without the need of a breathing apparatus. The principal objective of the CIS is to preclude the development of a combustible atmosphere by maintaining an oxygen deficient atmosphere inside SCCV.

The containment boundary is classified as Defense Line 3, Safety Class 1 (DL3/SC1) system.

The containment overpressure vent flow path is Defense Line 4b, Safety Class 3 (DL4b/SC3).

The containment supply and exhaust flow paths are (TBD).

9A.4.2.1 System and Equipment Functions

The BWRX-300 CIS provides the following functions during Normal and Off-Normal conditions.

9A.4.2.1.1 Normal Functions (Non-Safety Category)

Establish and maintain an inert atmosphere within containment for prevention of hydrogen combustion during post-accident events that may result in fuel damage. There is no significant hydrogen generation during normal operations.

Maintain a positive pressure within containment to prevent the infiltration of non-inert air during all normal plant operating modes, except plant shutdown

Establish a breathable de-inert atmosphere within containment during plant shutdown for refueling and maintenance

9A.4.2.1.2 Normal Functions (Safety Category)

The CIS penetrates containment and therefore must ensure containment boundary integrity and isolation capability.

9A.4.2.1.3 Off-Normal Functions (Non-Safety Category)

Maintain containment oxygen concentration below the maximum permissible limit during abnormal and accident conditions to ensure an inert atmosphere.

9A.4.2.1.4 Off-Normal Functions (Safety Category)

Protect against containment overpressurization, which could occur during degraded core conditions after a severe accident

The CIVs and associated piping and penetrations are designed in accordance with the rules and requirements of ASME BPVC Code, Section III, Division 1, Subsection NE-Class MC Components, and Subsection NCD, Class 2 Components, in accordance with their quality group classification. The containment pipe penetrations are Seismic Category A and the CIVs are Seismic Category B.

The design and application of the CIS system meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.6.10 (Reference 9A.4.2-1) as related to preventing deflagration or detonation that could jeopardize the integrity or leak tightness of the containment. In addition, containment isolation is provided in accordance with CNSC REGDOC-2.5.2 Section 8.6.6.

9A.4.2.2 Safety Design Bases

The Safety Design Bases of the CIS are as follows:

1. As part of Defense Line 3, the Containment Inerting System provides CIVs on piping that penetrates the containment boundary.
2. As part of Defense Line 3, the Containment Inerting System CIVs are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System.
3. To protect against containment overpressurization a containment overpressure vent flow path is provided. The containment overpressure vent flow path is Defense Line 4b, Safety Class 3 (DL4b/SC3).

9A.4.2.3 Description

Figure 9A.4.2-1 depicts the CIS. Components of CIS are located both inside and outside the reactor building. The nitrogen storage equipment, including the liquid nitrogen storage tank, nitrogen vaporizer, and heater are located in the yard.

The CIS provides the capability to supply containment with nitrogen to ensure that a noncombustible atmosphere is maintained following an accident that results in 100% reaction between the fuel cladding and water.

The CIS is capable of reducing the containment oxygen concentrations from atmospheric conditions to less than 3.5% by volume in less than 4 hours.

CIS is also capable of de-inerting the inerted containment to above 19.5% oxygen by volume within 4 hours to allow personnel safe access within containment without the use of a breathing apparatus.

The CIS consists of containment supply, containment exhaust, and containment overpressure venting flowpaths.

Containment Supply Flowpath

The containment supply flow path is used to inject nitrogen into containment for inerting (large nitrogen flow) and makeup (small nitrogen flow), inject breathable air into containment for de-inerting, and supply nitrogen to the PPS (Subsection 9A.4.1) to support actuation of air operated devices located inside containment. The containment supply flow path consists of a pressurized liquid nitrogen storage tank, vaporizer to convert the liquid nitrogen to a gaseous phase, electrical heater, piping, valves, instruments, and controls. Most of the components are located within the Reactor Building except for the storage tank, vaporizer, heater, associated valves, and piping, which are located in the yard. The containment supply flow path terminates in lower containment.

Containment Exhaust Flowpath

The containment exhaust flow path is available to be used to discharge the containment atmosphere during the inerting, makeup, and de-inerting process. The containment exhaust flow path consists of piping, valves, instruments, controls, and a tee to the containment overpressure vent flow path. All of the components are located within the Reactor Building. The exhaust flow path starts in upper containment on the opposite side of containment from the supply injection point and connects to the Heating Ventilation and Cooling System (Subsection 9A.5) before discharging to the vent stack.

Containment Overpressure Vent Flowpath

The containment overpressure vent flow path is used in case of a severe accident where containment failure by overpressure is threatened. The containment overpressure vent flow path consists of a rupture disc, a locked closed manual bypass valve, pressure indicator, isolation valve, check valve, sparger, and associated piping. The overpressure vent flow path connects to the containment exhaust flow path and terminates in the Reactor Equipment Pool. By relieving pressure into the Reactor Equipment Pool, water scrubbing is expected to occur, thereby reducing the contaminants. Additionally, the Reactor Equipment Pool is located inside the Reactor Building such that further holdup of any potential released contaminants is provided.

The rupture disk is designed in accordance with the rules and requirements of ASME BPVC, Section III, Class 2 Components. The rupture disk is designated Quality Group A. The containment pipe penetrations are Seismic Category A. The CIVs and rupture disk are Seismic Category B.

The over-pressurization vent line and associated sparger are Safety Class 3.

Environmental Qualification of Safety-Class SSC's is provided as applicable as discussed in Chapter 3, Subsection 3.9.

9A.4.2.3.1 Component Description

The following information is provided relative to the major equipment items in the ICC system.

Liquid Nitrogen Storage Tank

An insulated liquid nitrogen storage tank provides nitrogen supply to support CIS operations. The CIS liquid nitrogen storage tank is sized to provide enough nitrogen to support two consecutive nitrogen purges of containment. The tank includes level and pressure indicators, and the bottom of the tank is sloped to facilitate draining. Pressure is maintained automatically in the tank during nitrogen discharge by a circuit with an ambient heat exchanger fed by a pressure control valve. The vapor space can be manually vented to control tank internal pressure. The tank is fabricated of material suitable for cryogenic service in order to withstand low liquid nitrogen temperatures. The tank is equipped with quick disconnect/connect or equivalent type of connection to support refilling.

Vaporizer

The nitrogen vaporizers are used to convert the liquid nitrogen from the liquid nitrogen storage tank to the gaseous phase. The CIS nitrogen vaporizers are sized to provide at least 2.5 times the containment free volume of nitrogen within the 4-hour inerting period. The CIS nitrogen vaporizers are fabricated of materials suitable for cryogenic service.

Heater

Vaporization of nitrogen for inerting and makeup is provided by an ambient air heat exchanger with auxiliary electrical heating. The CIS nitrogen heaters are designed to heat the nitrogen makeup to a temperature greater than 10 °C. The CIS nitrogen heaters are fabricated of materials suitable for cryogenic service.

Rupture Disk

The rupture disk is part of the containment over-pressurization vent flowpath and provides passive overpressurization protection for containment. The CIS rupture disk burst pressure tolerance at the specified disk temperature does not exceed 5% of marked burst pressure.

9A.4.2.4 Materials

CIS Structures, Systems and Components are fabricated of materials suitable for cryogenic service as required.

9A.4.2.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.4.2-2 for CIS interfaces.

9A.4.2.6 System and Equipment Operation

Table 9A.4.2-1, "BWRX-300 Operating Modes" provides a summary of the plant operating modes relative to CIS operations.

Initial Configuration (Pre-Startup)

The initial configuration of the CIS is set to inerting Mode with the CIVs closed. The CIS is in standby readiness condition prior to reactor startup.

System Startup

During System Startup (Mode 2), the SCCV hatches are closed and the CIS is in-service. Coming out of shutdown where the SCCV was opened or any condition where the oxygen concentration is above the acceptable level, the CIS is set to inerting mode. Liquid nitrogen from the nitrogen storage tank is vaporized, heated, and subsequently injected into the lower SCCV while exhausting from the upper SCCV until the correct oxygen concentration is reached. The nitrogen is mixed with the SCCV atmosphere by the Containment Cooling System fans. Once the targeted oxygen concentration and pressure levels are reached, the CIS Containment Isolation Valves close.

Once inerting is complete, the CIS can be set to makeup mode to maintain the required oxygen concentration and a slightly positive pressure inside the SCCV to preclude oxygen infiltration from the Reactor Building. Liquid nitrogen from the nitrogen storage tank is vaporized, heated, and subsequently injected into the lower SCCV. During SCCV atmospheric heating conditions during plant startup, the exhaust line is opened to relieve excess SCCV pressure.

Normal Operations

During Power Operation (Mode 1), the containment vessel hatches are closed, and the CIS is in-service. The CIS makeup mode is activated when the SCCV pressure drops below the desired setpoint and/or the oxygen concentration rises above the desired setpoint. Positive pressure is needed in the SCCV to preclude oxygen infiltration from containment leakage. Liquid nitrogen from the nitrogen storage tank is vaporized, heated, and subsequently injected into lower SCCV. During the maximum containment atmospheric heating conditions, the containment exhaust line is opened to maintain SCCV internal pressure.

During Refueling (Mode 6), the CIS is idle. The containment vessel is open and the CIS is rendered ineffective.

Off-Normal Operations

In the event of a severe accident, resulting in SCCV flooding and elevated internal SCCV pressure, the rupture disc within the containment overpressure vent flowpath will passively open to vent the internal SCCV pressure. The rupture disc is passive requiring no operator action to open. However, an operator controlled manual bypass valve around the rupture disc can be used as an alternate method of venting.

The SCCV internal pressure is monitored in the main control room or by the local pressure indicator to support manual operation. The overall SCCV pressure decreases as venting continues. Once the SCCV internal pressure has been reduced to a safe level and normal containment heat removal capability has been regained, the containment overpressure protection air-operated isolation valve is closed to reestablish the SCCV pressure boundary. Radiological conditions associated with severe accidents are presented in Chapter 15, Subsection 15.3.

System Shutdown

During System Shutdown (Modes 3, 4, and 5), if the containment vessel hatches are required to be opened to support personnel entry, the CIS is set to the de-inerting mode. De-inerting mode is used to achieve a safe, breathable atmosphere once the reactor reaches a setpoint temperature. In de-inerting mode, the CIS injects breathable air into the lower SCCV while exhausting from the upper SCCV until the correct oxygen concentration is reached. The breathable air is mixed with the SCCV atmosphere by the Containment Cooling System fans. Once the targeted oxygen concentration is reached for a breathable atmosphere, the containment vessel hatches may be opened for personnel entry. Opening the containment vessel hatches will render the CIS ineffective, and thus it will be idle.

If the containment vessel is not required to be opened during System Shutdown (Modes 3, 4, and 5), the CIS may be set to makeup mode to maintain an inerted atmosphere in containment.

Refer to Chapter 1, Subsection 1.8 for a description of the operating modes.

9A.4.2.7 Instrumentation and Control

The following instrumentation features are included for monitoring CIS operations:

1. Flow transmitters for containment inerting flow indication
2. Temperature elements to monitor the inerting nitrogen supply temperature
3. Pressure transmitters for control of the inerting pressure control valves

Features provided for control of CIS operation include the following:

1. Containment Pressure indication provided by Process and Radiation Monitoring System (Chapter 11, Subsection 11.5)
2. Inerting nitrogen supply temperature and pressure for control of the inerting and makeup process
3. Position indication for all remotely operated valves

The following CIS displays and alarms are provided:

1. Main Control Console Indications:
 - a. Containment inlet flow rate
 - b. Position indication (full open and full closed) for all containment isolation valves
 - c. CIS control valves position indication
 - d. Containment oxygen concentration
 - e. Containment inlet temperature
2. Main Control Room Alarms
 - a. High and low nitrogen discharge temperature

- b. Low nitrogen storage tank liquid level
- c. Low and high containment inlet temperature
- d. CIS valve operation failure
- e. High oxygen levels in containment

9A.4.2.8 Monitoring, Inspection, Testing, and Maintenance

The following pertains to testing and inspection of CIS components:

1. Periodic testing is performed to demonstrate the ability of CIS to meet design requirements. Each valve is exercised both open and closed and the position indication is verified. Trip and alarm logic signals are also checked. The tests assure correct functioning of all controls, instrumentation, piping and valves. System reference characteristics, such as pressure differentials and flow rates, are documented during the pre-operational tests and are used as baseline for measurements in the subsequent operational tests. Testing is performed both during operations and shutdown.
2. During plant operation, CIS valves, instrumentation, wiring and other components outside the containment can be inspected visually at any time.
3. Containment isolation valves are tested for leakage periodically in accordance with the requirements of 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Type Test C.
4. The rupture disk is tested and replaced periodically, providing additional confidence of pressure.

Testing in support of plant pre-operational testing, startup, and commissioning is addressed in Chapter 14, Subsection 14.3.

Testing is performed to ensure required functional operability is maintained under design conditions. The Non-Safety-Category functions of the CIS are tested in accordance with ASME Code B31.1. The Safety-Category functions of the CIS are tested in accordance with ASME BPVC-III.

Maintenance practices consider industry best practices and operating experience and conform with plant safety requirements to minimize potential for personnel injury. Maintenance activities implement ALARA practices to minimize work activity dose. Maintenance activities involving plant equipment may require involvement of vendors or industry specialists.

Other maintenance provisions include the following:

1. Isolation valves are provided on both sides of the supply and exhaust control valves to isolate the control valves and allow for maintenance and repair.
2. Pull spaces are provided around valves to allow for removal and maintenance.

9A.4.2.9 Radiological Aspects

Chapter 12, Section 12.3 provides information pertaining to design measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.4.2.10 Performance and Safety Evaluation

The CIS Safety-Category functions during normal conditions include ensuring containment boundary integrity and isolation capability. During Off-Normal conditions, CIS Safety-Category functions include providing protection against containment overpressurization.

The following features are incorporated into the design of the CIS to enhance reliability of the CIS Safety-Category functions:

1. Periodic system testing, and preventive maintenance.
2. If the containment supply pressure control valve fails, a manual bypass valve can be opened to supply nitrogen to the containment. Pressure relief valves are positioned downstream of the pressure control valve to protect the downstream piping and provide for pressure control during manual bypass of the pressure control valve. The manual bypass function allows for continuous operation of CIS while the pressure control valve is out of service.
3. The CIS containment isolation valves are located outside of the SCCV which removes them from the harsh environment of the containment and protects them from the effects of flood and dynamic effects of pipe breaks. The CIVs are accessible for inspection and testing during reactor operation. Also, the CIVs fail closed on loss of air or control signal.
4. A manual bypass valve around the containment overpressure vent rupture disc can be used as an alternate method of venting. The bypass line is equipped with a local pressure indicator that can be used to support early containment venting during a beyond design basis accident.

The CIS structures, systems, and components that provide Safety-Category functions are located in the Reactor Building which provides adequate protection against natural phenomena ensuring the ability of the CIS to perform its Safety-Category functions.

9A.4.2.11 References

- 9A.4.2-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

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Table 9A.4.2-1: BWRX-300 Operational Modes

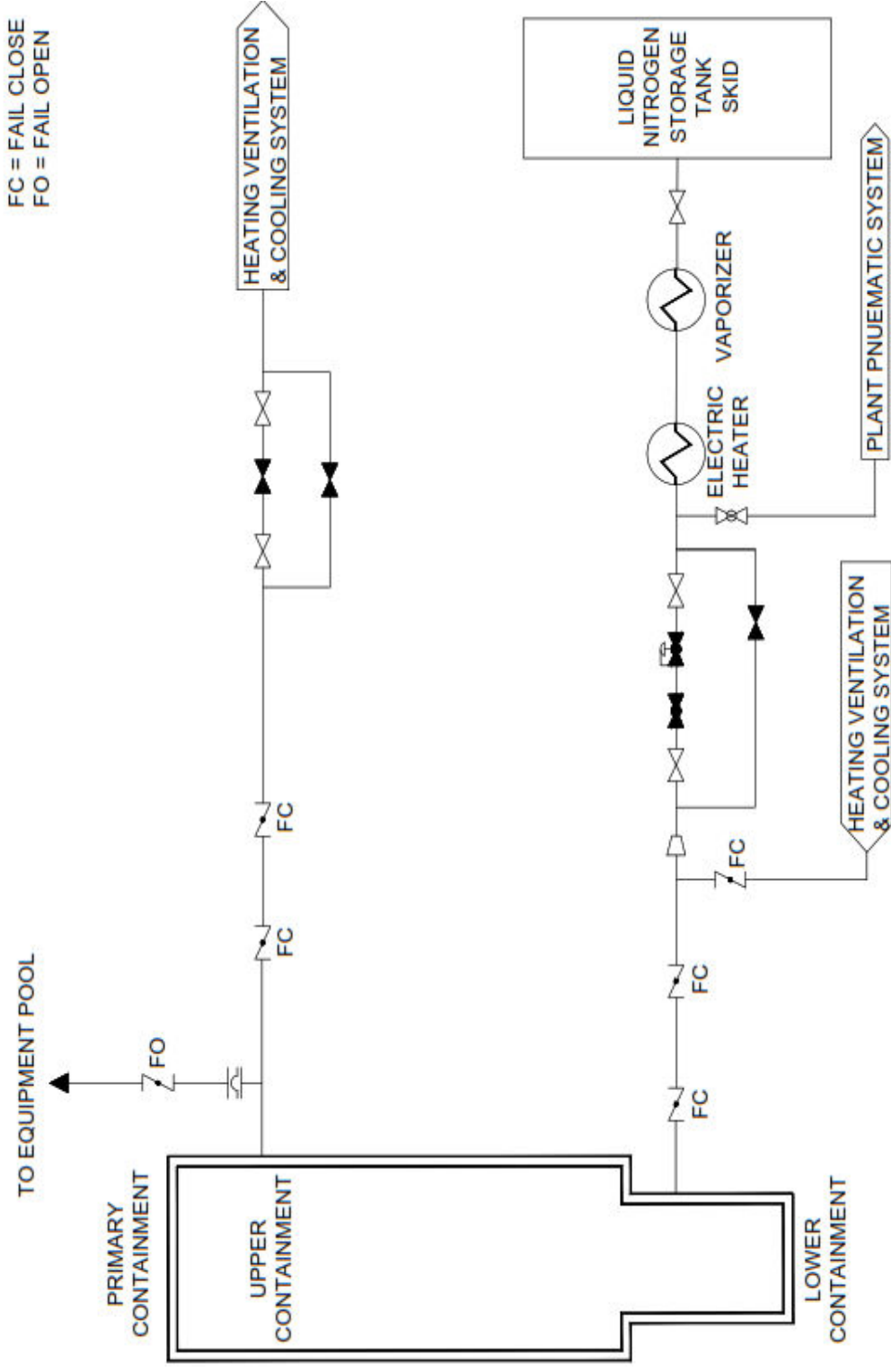
Mode	Title	Reactor Mode Switch Position	CIS Modes
1	Power Operation	Run	Inerting or Makeup
2	Startup	Refuel or Startup	Inerting or Makeup
3	Hot Shutdown	Shutdown	De-Inerting or Idle
4	Stable Shutdown	Shutdown	De-Inerting or Idle
5	Cold Shutdown	Shutdown	Idle
6	Refueling	Shutdown or Refueling	Idle

Table 9A.4.2-2: Containment Inerting System Interfaces

Interfacing System	Interface Description	Interface Boundary
Safety Class 2 and 3 Instrumentation and Control System	The CIS provides instrument data to the Safety Class 2 and 3 Instrumentation and Control System for system monitoring.	Process Parameter Inputs and component status to CIS System Logics
	The Safety Class 2 & 3 Instrumentation and Control System provides a control signal to the CIS solenoid valves.	Nitrogen Supply Isolation Valve Fresh Air Supply Isolation Valve Primary Exhaust Line Control Valve Makeup Exhaust Line Control Valve
Safety Class 1 Instrumentation and Control System	The CIS provides instrument data to the Safety Class 1 Instrumentation and Control System for system monitoring.	Process and Equipment status of safety related components in the CIS system
	The Safety Class 1 Instrumentation and Control System provides an actuation signal to the CIS containment isolation solenoid valves.	Outboard Supply Line Containment Isolation Valve Inboard Supply Line Containment Isolation Valve Inboard Exhaust Line Containment Isolation Valve Outboard Exhaust Line Containment Isolation Valve Overpressure Vent Line Isolation Valve
Reactor Building Structure	The CIS provides a containment overpressure relief path to the Equipment Pool of the Reactor Building Structure for a beyond design basis accident.	Containment Overpressure Vent Line Sparger
SCCV	The CIS penetrates the SCCV to support containment inerting, de-inerting, and makeup.	Supply Line Containment Mechanical Penetration Exhaust Line Containment Mechanical Penetration

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Interfacing System	Interface Description	Interface Boundary
Heating Ventilation and Cooling System	The CIS exhausts containment air to the Heating Ventilation and Cooling System to support containment inerting and de-inerting.	CIS Exhaust Line to HVS Building Exhaust Line downstream of Exhaust Line Control Valves
	The Heating Ventilation and Cooling System supplies breathable air to the CIS to support containment de-inerting.	Fresh Air Supply Isolation Valve
Plant Pneumatics System	The Plant Pneumatics System supplies compressed air to the CIS to support actuation of pneumatically operated devices.	Nitrogen Supply Isolation Valve Fresh Air Supply Isolation Valve Primary Exhaust Line Control Valve Makeup Exhaust Line Control Valve Outboard Supply Line Containment Isolation Valve Inboard Supply Line Containment Isolation Valve Inboard Exhaust Line Containment Isolation Valve Outboard Exhaust Line Containment Isolation Valve Overpressure Vent Line Isolation Valve
	The CIS supplies nitrogen to the Plant Pneumatics System to support actuation of pneumatically operated valves located inside of containment.	PPS Supply Line Isolation Valve



9A.5 Heating, Ventilation and Air Conditioning Systems

The Heating, Ventilation and Cooling System (HVS) serves all areas of the power block during normal operation, with the exception of the containment which is serviced by the Containment Cooling System (Subsection 9A.5.6). HVS services containment in de-inerted modes of operation cross connected with the Containment Inerting System (Subsection 9A.4.2). The HVS maintains space design temperatures, quality of air and pressurization. It provides a controlled environment for personnel safety and comfort, and for the proper operation and integrity of equipment located in the power block.

The HVS consists of subsystems for fresh air and ventilation of each building. With the exception of the Control Building subsystem (Subsection 9A.5.2), all of the subsystems exhaust to a common plant exhaust stack during all normal operation modes. The plant exhaust stack is monitored for radiation. The Control Building is outside the radiologically controlled area and therefore the exhaust does not require radioactive monitoring.

The HVS is broken down into 6 different areas:

- Control Building
- Reactor Building
- Radwaste Building
- Turbine Building
- Plant Services Area of the Turbine Building (PLSA)
- Plant Vent Stack (PVS)

9A.5.1 Reactor Building Heating, Ventilation and Air Conditioning System

9A.5.1.1 System and Equipment Functions

RB HVS, system, and equipment functions include the following:

9A.5.1.1.1 Normal Functions (Non-Safety Category)

1. Provides a controlled environment for personnel comfort and safety and equipment operation. Outside air is provided to meet the ventilation requirements for indoor air quality consistent with the requirements of ASHRAE 62.1 (Reference 9A.5.1-1).
2. Maintains designated clean areas at higher than atmospheric (positive) pressure to minimize the infiltration of outside air.
3. Maintains negative pressurization of potentially contaminated areas to control leakage of potentially radioactive effluent to the atmosphere or to other rooms.
4. The Battery Rooms are ventilated and exhausted by the RB HVS to prevent hydrogen accumulation in the rooms.
5. Reduces the potential spread of airborne contamination by maintaining airflow from areas of lower potential for contamination to areas of greater potential for contamination.
6. Detects and limits the introduction of airborne radioactivity into the SCR.
7. Provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire.

8. The HVS is designed such that failure of Non-Safety-Category function equipment does not compromise Safety Category function equipment.
9. The HVS supplies outside air to the Steel-plate Composite Containment Vessel (SCCV) via the Containment Inerting System (Subsection 9A.4.2) and exhausts the SCCV inerting gases to provide a habitable environment for maintenance personnel during outages.

9A.5.1.1.2 Off-Normal Functions (Non-Safety-Category)

RB HVS systems continue to operate during off-normal conditions until removed from service either manually or automatically.

In the event of a LOOP, the RB HVS continues to function due to its connection to the Diesel Generators which activate and provide backup power to their respective train-supported equipment. This includes the normal AHUs, their associated electric heaters, and Fan Coil Units for the Defense Line 2 (DL2) and DL4a DCIS rooms.

9A.5.1.1.3 Normal Function (Safety-Category)

The RB HVS provides required cooling to Safety Class rooms with two redundant SC3 FCUs in each room (Figure 9A.5.2-1).

9A.5.1.1.4 Off-Normal Functions (Safety-Category)

The RB HVS performs Safety-Category (TBD) functions. The RB HVS has sufficient capacity to fulfill its safety function and there are no credible single failures or operator errors that could defeat the performance of the safety function for which the system was designed.

The RB HVS includes two Safety-Category (TBD), 100% capacity pressurization fans with emergency filtration units which supply filtered outside air to the SCR which is used to pressurize the space relative to adjacent spaces, protecting the operators from hazards such as high radiation levels resulting from DBAs, release of radioactive material. These pressurization fans are provided with backup power making them available for use during a LOOP or Station Blackout (SBO).

The SCR and supporting Division 1, 2, and 3 Electrical and Switchgear rooms are provided with passive cooling. Emergency filtration and pressurization systems for the SCR are powered from the SC1 Electrical Distribution System batteries for the required 72 hours.

Tornado dampers are provided at each RB HVS penetration to protect the RB structure.

The RB HVS includes Safety-Category (TBD) supply and exhaust isolation dampers for the RB, which close automatically in the event of a fuel handling accident, isolating the RB. These dampers located at the RB boundary and the associated controls that provide the isolation signal are seismic..

The design of the HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2 (Reference 9A.5.1-2) Section 7.10 as related to providing an HVAC system capable of maintaining the required environmental conditions for systems and components important to safety in all states.

Refer to Chapter 11, Subsection 11.3 for information pertaining to Gaseous Waste Management System.

The design of the RB HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.11.2 as related to providing a ventilation system with a filtration system capable of the following:

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1. Preventing unacceptable dispersion of airborne contaminants within the plant. Airborne contaminants are contained through the use of two 100% capacity RB Exhaust AHUs with HEPA filtration which take suction on the RB keeping it at negative pressure, ensuring that air flows into the RB from adjacent clean air spaces. The potential spread of airborne contamination is minimized by maintaining airflow from areas of lower potential for contamination to areas of greater potential for contamination.
2. Reducing the concentration of airborne radioactive substances to levels compatible with the need for access to each particular area is achieved by using shielding, ventilation, monitoring instrumentation and ALARA design concepts as discussed in Chapter 12 to ensure the overall design minimizes radiation exposure to workers and to the public.
3. Keeping the level of airborne radioactive substances in the plant below prescribed limits. The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance is below 0.1 the derived air concentration as specified in 10 CFR 20 Appendix B (Reference 9A.5.1-3) during normal power operation.
4. Application of ALARA design principles in normal operation as discussed in Chapter 12, Subsection 12.1.5.4.
5. Ventilating rooms containing inert or noxious gases without impairing the capability to control radioactive releases by using charcoal filters.

9A.5.1.2 Safety Design Bases

1. As part of Safety-Category (TBD) requirements, the RB HVS ensures that a habitable environment is maintained in the SCR for a minimum of 72 hours following an initiating event.
2. As part of Safety-Category (TBD) requirements, the RB HVS ensures that the SCR temperature does not exceed the limiting temperature during a postulated design basis accident event.
3. As part of Safety-Category (TBD) requirements, the RB HVS is designed to ensure that the SCR operators do not receive radiation exposure in excess of 0.05 Sv (5 rem) total effective dose equivalent in a single year.
4. As part of Defense Line 2, the RB HVS provides the capability to isolate ventilation to the SCR when signaled to do so from Safety-Category (TBD) Instrumentation and Control System.
5. Tornado dampers are provided for the RB HVS in support of Safety-Category (TBD) functions.
6. As part of Defense Line 2, the RB HVS provides the capability to isolate ventilation to the refuel floor when signaled to do so from Safety-Category (TBD) Instrumentation and Control System.
7. RB HVS interfaces with the Safety-Category (TBD) Instrumentation and Control System monitor and diagnostics function to monitor Safety-Category (TBD) equipment room temperature information.

8. The RB HVS battery room exhaust fans are powered from the same bus as Safety-Category (TBD) Electrical Distribution System battery chargers associated with the respective divisional room.

9A.5.1.3 Description

Figure 9A.5.1-1 depicts the Reactor Building HVAC Process Flow Diagram.

The RB, for HVAC purposes, is divided up into two main areas which includes the lower levels and the upper areas.

The RB lower levels are supplied with conditioned air provided by one of two AHUs, located on the Control Building roof. These AHUs supply air through vertical duct chases down to the lowest elevation of the RB. The RB lower-level supply AHUs also supply outside air to the Containment Inerting System (Subsection 9A.4.2) when required for containment de-inerting and to provide a habitable environment for maintenance personnel during refueling and maintenance operations.

Two 100% capacity RB exhaust AHUs with HEPA filtration take suction on the RB keeping it at negative pressure, ensuring that air flows into the RB from adjacent clean air spaces. The operating exhaust AHU discharges through ductwork to the Continuous Exhaust Air Plenum (CEAP) and to the PVS. These AHUs are provided with adjustable speed drives to be able to reduce flow comparable to reduced supply air flow to the RB. The normal lower-level exhaust ductwork also receives air and inerting gases from the CIS when required, which is filtered and exhausted to the Continuous Exhaust Air Plenum and to the PVS.

Each of the three Division Battery Rooms have two 100% capacity exhaust fans which are provided with backup emergency power, exhausting to atmosphere in the event of a Loss-of-Offsite Power through ductwork and a louver mounted in a room up above. The Battery Rooms normally exhaust to the operating RB Exhaust AHU and the PVS. The RB lower-level AHU's also supply conditioned outside air to the Containment Inerting System when needed.

The supply ductwork to each of the three Battery Rooms, (Division 1, 2 and 3 Battery Room) and the Secondary Control Room are each provided with electric duct heaters.

Supplemental cooling and air circulation is provided by fan coil units.

Heating to the RB Stairwells is supplied by electric cabinet heaters while heating to the Entry/Truck Bay is provided by an electric unit heater.

When needed, filtered outside air is supplied to the Secondary Control Room by one of two capacity Pressurization Fans and Emergency Filtration Unit. The two pressurization fans draw outside air through dedicated SC1, Seismic Category B safety-related ductwork with blast resistant openings located in the exterior walls. The two makeup AHUs draw outside air through dedicated Safety-Category 1 ductwork with blast resistant openings.

The operating SCR makeup AHU pressurization fan maintains a minimum positive pressure in the SCR with filtered makeup air, protecting the operators from potential radioactive gases. The SCR is normally provided filtered, conditioned air from the operating RB Lower-Level AHU.

The RB upper levels are supplied with conditioned air provided by AHUs located on the Plant Service Area roof. These AHUs are provided with filters, electric heating coils, and chilled water supplied cooling coils, supplying conditioned outside air to the RB upper levels. These AHUs employ adjustable speed drives to vary flow to the fuel pool as needed during refueling, and to reduce flow during reduced temperature ambient conditions.

Exhaust air from the RB lower levels is combined with exhaust air from the upper levels. Exhaust air is discharged out of the RB building by AHUs. These AHUs are provided with High Efficiency

Particulate Air (HEPA) filters, prefilters and exhaust fans. The discharge is sent to the Continuous Exhaust Air Plenum and PVS.

9A.5.1.4 Component Description

Design information related to the major components of the Reactor Building HVS is provided below.

9A.5.1.4.1 Filters

The various building filtration levels are specified within each specific Subsection. Filters meet the applicable efficiency rating as stated below. American Society of Heating, Refrigerating and Air Conditioning Engineers (ASHRAE) Standard 52.2 (Reference 9A.5.3-4) establishes a filter's Minimum Efficiency Reporting Value (MERV) and establishes a filter's Average Atmospheric Dust Spot Efficiency. The following ASHRAE filter classifications are MERV specified (with Dust Spot Efficiency in parenthesis) below:

- Low Efficiency: MERV 1-4 (Less than 20%)
- Medium Efficiency: MERV 5-12 (As least 20% but less than 80%)
- High Efficiency: MERV 13-16 (Greater than or equal to 80%)

HEPA filters are specified for various building filtration systems. Filters meet the applicable efficiency rating as stated below.

Filters with efficiency greater than MERV 16 by ASHRAE Standard 52.2 (MERV 17-20) (Reference 9A.5.1-4) are usually rated by the dioctylphthalate) test method. This test is based on the ability of a filter to remove an aerosol consisting of 0.3 micrometer (micron) particles of a test challenge. HEPA filters are extended-medium dry-type filters in a rigid frame, having minimum particle-collection efficiency of 99.97% on 0.3-micron particles which meets ASME AG-1, Code on Nuclear Air and Gas Treatment, Section FC, HEPA Filters (Reference 9A.5.1-5).

HEPA filters are constructed, qualified, and tested per Underwriters Laboratory-586, High Efficiency, Particulate, Air Filter Units (Reference 9A.5.1-6).

Each of the two SCR Emergency Makeup Filtration Units contains a charcoal filter, located between upstream and downstream banks of HEPA filters. The upstream bank protects the charcoal from clogging. The downstream bank prevents discharging charcoal fines in the SCR. The purpose of the charcoal bed is to remove potentially radioactive gases from the air stream. The charcoal bed also has properties associated with toxic gas adsorption.

9A.5.1.4.2 Air Handler Units

Each AHU consists of an inlet area, filters (as specified by the system), electric heating coils, (as required) cooling coils (as required) and the respective fans (supply or exhaust). Bag In/Bag Out AHUs and Emergency Filter Units (EFUs) are intended for use in units that contain potentially contaminated filters to facilitate filter changeout without the worker or room being exposed to the potentially contaminated filter.

9A.5.1.4.3 Supply and Exhaust Fans

The various building ventilation systems are provided with supply and exhaust fans, sometimes incorporated into AHUs. These fans are either centrifugal or axial fans depending on the suitability to the specific system. The fans are designed, manufactured, and supplied in accordance with the standards of the Air Movement and Control Association International. Fans in various areas are equipped with Adjustable Speed Drive mechanisms to control airflows for the specific system application.

9A.5.1.4.4 Heating Coils/Elements

Various AHUs are equipped with electrical heating coils/elements. Electric coils are designed and supplied to the requirements of Underwriters Laboratory, "Heating and Cooling Equipment" (Reference 9A.5.1-7).

9A.5.1.4.5 Cooling Coils

The Tubular fin type cooling coils are designed, constructed, and installed in accordance with ASHRAE 33, Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils (Reference 9A.5.2-8) and ANSI/ARI 410 (Reference 9A.5.2-9) and Underwriters Laboratory (Reference 9A.5.2-7).

Cooling coil condensate is collected in drain pans within the air handler units with the drain pan discharge (condensate) routed to the Equipment and Floor Drain System (Subsection 9A.9.3).

9A.5.1.5 Materials

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing irradiation of the plant components, corrodents and mitigating the degradation of materials through material chemistry, heat treatment, contamination, and material processes controls identified in the equipment and purchase specifications.

9A.5.1.6 Interfaces with Other Equipment or Systems

Refer to Table 9A.5.1-1 for RB HVS interfaces with other equipment or systems.

9A.5.1.7 System and Equipment and Operation

9A.5.1.7.1 Normal Operation

The RB lower elevations will normally be provided heated, conditioned, and filtered once-through supply air from one of two operating AHUs located on the Control Building roof through supply ductwork. AHU cooling is supplemented by cooling provided by chilled water supplied FCUs located in rooms that require extra cooling, with FCUs being controlled by room thermostats. An electric duct heater in the supply duct to each Battery Room and to the SCR is expected to normally be operating as required to maintain SCR habitability and Battery Room temperatures as recommended by the battery manufacturer.

The RB upper elevations, consisting mainly of the fuel handling area operating deck, will normally be provided heated, conditioned, and filtered once-through supply air from one of two operating AHUs located on the Plant Service Area roof through supply ductwork. For each of the operating RB upper-level supply AHUs, the second supply AHU will normally be in standby, ready for auto-start should the operating AHU fail.

The Truck Bay, in addition to being provided conditioned air from the operating upper-level supply AHU, is heated by an electric unit heater, controlled by local thermostat.

Air will normally be exhausted from the RB upper and lower levels, including the Battery Rooms, from one of two operating exhaust AHUs discharging ultimately to atmosphere from the PVS.

For each of the RB supply and exhaust AHUs, the second AHU will normally be operational, in standby, ready for auto-start should the operating AHU fail.

9A.5.1.7.2 Off-Normal Operation

In the event of a LOOP, power is maintained to both RB exhaust AHUs, and all four RB supply AHUs and associated heating coils for the 4 supply AHUs.

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In the event of a SBO, power is lost to the RB exhaust AHUs as well as the upper and lower-level supply AHUs which normally provide heating and air conditioning for the RB, as well as electric duct heaters for the Battery Rooms.

Each Battery Room, normally exhausted via the RB exhaust AHU, continues to be exhausted via one of two 100 percent capacity exhaust fans, powered from its backup power source, for the purpose of being placed into service during loss of power events. These fans are controlled by ON/OFF/AUTO-switches located in the SCR, with fan auto-starts initiated by undervoltage on the RB exhaust AHUs' power supply.

In the event of a loss of the MCR in the CB, the SCR in the RB is activated, and its dedicated HVAC system is placed into service. The emergency pressurization fans with charcoal radioactive iodine gas filtration, and backed up by battery power, will also be placed into service automatically upon a loss of power to the RB lower-level normal supply AHUs or detection of radiation, at the RB lower-level normal supply AHU intakes.

The Secondary Control Room pressurization EFUs and emergency pressurization fans, including the intake, ductwork, and power supplies are designed to operate for up to seventy-two hours without normal AC power.

In the event of a fuel handling accident on the refueling floor in the Fuel Handling Machine area, a radiation monitor alarm initiates closure of the RB isolation dampers and securing of the RB upper level supply AHUs.

9A.5.1.8 Instrumentation and Control

The following signals are provided as inputs to the control logic:

1. Temperature elements to monitor space or duct air temperatures, and to control cooling water control valves, electric heating elements, or control fan speeds.
2. Moisture element in the SCR room to engage RB lower-level supply AHU dehumidification control.
3. Differential pressure transmitters and air flow instruments, as required, to monitor fan operation.
4. Differential pressure transmitters to monitor pressure drop across AHU filters.
5. Intake, exhaust, and return air damper position monitoring
6. Duct mounted smoke detectors to shut down fans as required

Manual initiation and shutdown of the RB HVS are provided at the Temperature Control Panel.

The following control features are implemented in the RB HVS:

1. The RB AHUs, FCUs, and exhaust fans are operated from the RB temperature control panels. An AHU start request automatically opens all associated shutoff dampers. A damper operating failure (any associated damper not 100% open while fan is running) automatically trips that particular AHU.
2. Any AHU trip automatically closes all its associated shutoff dampers.
3. Any AHU started in standby because of a low air flow condition automatically trips the AHU that generated the low flow condition after proof that the standby AHU is running. To avoid spurious standby starts and AHU trips, all air flow logic incorporates time delays to allow fans to generate steady-state flow.

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4. Duct mounted temperature elements monitor the operation of the RB AHUs.
5. Local wall mounted temperature elements in the SCR and Battery Rooms control the electric duct heaters in the supply air ducts to those rooms.
6. Control valves in RB AHU cooling coil supply return waterlines are modulated to maintain a constant coil leaving air temperature. If the outside air temperature is less than a predetermined temperature, the control valve supplying the RB supply AHU cooling coils is 100% closed; otherwise, the valve is modulated.
7. The FPS provides a signal to shut down the RB AHUs in the event of a fire. The Battery Room exhaust fans continue to operate, preventing a potentially explosive hydrogen concentration from occurring. Smoke removal is also performed in conjunction with the FPS. Duct mounted smoke detectors provide a smoke detected signal to the FPS local fire protection panels.
8. The Battery Room exhaust fans operate off timers. Failure of any Battery Room exhaust fan is annunciated via a common HVS alarm in the MCR.
9. The lower RB supply AHU intakes are monitored for toxic gas and radiation. If toxic gas or radiation are detected, the lower RB supply AHU turns off and the isolation dampers shut. Additionally, the SCR normal supply and exhaust isolation dampers shut, the SCR pressurization fans energize, supplying the SCR with filtered outside air.
10. Upon high radiation detection in the Operating Deck area of the RB, the RB upper level supply and exhaust isolation dampers automatically close, and associated supply AHUs are secured, isolating supply and exhaust air to/from that area. These dampers along with associated actuators are SC3, seismically qualified, and fail shut on a loss of power or signal.

Instrumentation is designed to support human performance. Alarms and notifications are clearly labeled and identifiable.

The RB HVS system is installed and located in physically accessible locations whenever possible. Otherwise, the means by which operators and personnel can access equipment is provided.

The MCR is provided with a Master Alarm indicator from each temperature control panel. The Master Alarm indicator prompts operators to conduct further review at the local temperature control panel. The local temperature control panel provides additional information and the means of resetting alarm conditions on a return-to-normal transition.

Any Safety-Category functions or other required functions happen automatically with notification provided to operators.

9A.5.1.9 Monitoring, Inspection, Testing, and Maintenance

9A.5.1.9.1 System Level Requirements

A positive pressure test is performed on RB HVS serving contaminated areas in the RB. This test is performed as required.

Plant-level HVAC inspection and maintenance is conducted in accordance with ASME N511 (Reference 9A.5.1-10).

9A.5.1.9.2 Component Level Requirements

Air handlers are field tested as required per ASME N511, (Reference 9A.5.1-10).

Charcoal beds used for filtration of radiological effluents support meeting ALARA requirements as discussed in Chapter 12, Section 12.1.

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Personnel and lay-down access are provided around instruments to allow adequate space for maintenance purposes.

HEPA filters conform to the requirements of ASME of AG-1 (Reference 9A.5.1-4).

9A.5.1.10 Radiological Aspects

Chapter 12, Section 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.5.1.11 Performance and Safety Evaluation

The RB HVS has sufficient capacity to fulfill its safety function and there are no credible single failures or operator errors that could defeat the performance of the safety function for which the system is designed. The RB HVS is designed with redundancy to assure that subsystems are normally available during all modes of plant operation, including startup and shutdown.

The RB HVS ensures that a habitable environment is maintained in the SCR for a minimum of 72 hours following an initiating event. The SCR and supporting Division 1, 2, and 3 Electrical and Switchgear rooms are provided with passive cooling systems powered from the Safety-Category (TBD) Electrical Distribution System batteries for the required 72 hours. The supplemental passive cooling system also ensures that the SCR temperature does not exceed limiting temperature limits during a postulated design basis accident.

The RB HVS is designed to preclude SCR operators from receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent in a single year by utilizing two Safety-Category (TBD), 100% capacity pressurization fans which supply filtered outside air to the SCR which pressurizes the space relative to adjacent spaces. This design feature protects operators from radiation associated with a Design Basis Accident.

The RB HVS includes supply and exhaust isolation dampers which close automatically in the event of a fuel handling accident, isolating the RB upper levels. These dampers are located at the RB boundary. In addition, the RB HVS provides the capability to isolate ventilation to the SCR. In both cases, the associated controls that provide the isolation signal is supplied from the Safety-Category (TBD) Instrumentation and Control System.

Tornado dampers are provided for the RB HVS supporting Safety-Category (TBD) functions.

The Safety-Category (TBD) Instrumentation and Control System monitors and provides diagnostic functions relative to RB HVS Safety-Category (TBD) equipment room temperature.

Battery room exhaust fans are powered from the same bus as the Safety Class 1 Electrical Distribution System battery chargers associated with the respective divisional room. This requirement prevents hydrogen production from battery charging operations if the exhaust fans stop working due to a loss in the bus power.

9A.5.1.12 References

9A.5.1-1 ANSI/ASHRAE 62.1, "Ventilation for Acceptable Indoor Air Quality."

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- 9A.5.1-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.1-3 10 CFR 20 Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentration for Release to Sewerage."
- 9A.5.1-4 ANSI/ASHRAE 52.2, "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size."
- 9A.5.1-5 ASME AG-1, "Code on Nuclear Air and Gas Treatment," Section FC, "HEPA Filters," American Society of Mechanical Engineers.
- 9A.5.1-6 UL-586, "High Efficiency, Particulate, Air Filter Units," Underwriters Laboratory.
- 9A.5.1-7 UL 1995, "Heating and Cooling Equipment," Underwriters Laboratory.
- 9A.5.1-8 ANSI/ASHRAE 33, "Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils."
- 9A.5.1-9 AHRI Standard 410, "Forced Circulation Air-Cooling and Air-Heating Coils," Air Conditioning, Heating, and Refrigeration Institute.
- 9A.5.1-10 ASME N511, "In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems," American Society of Mechanical Engineers.

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Table 9A.5.1-1: Reactor Building Heating, Ventilation and Air Conditioning System Interfaces

Interfacing System	Interface Description	Interface Boundary
Non-Safety Instrumentation and Controls	The Non-Safety Instrumentation and Controls provides distributed control and instrumentation data communication networks	At the HVS equipment
Process Radiation and Environmental Monitoring System	The PREMS provides continuous radiation monitoring of the HVS	At the HVS equipment
Chilled Water Equipment	The CWE provides chilled water for cooling of the HVAC FCUs and AHUs	At the HVS equipment
Safety-Category (TBD) Electrical Distribution System	The Safety-Category (TBD) Electrical Distribution System provides electrical power to SCR emergency fans and electric duct heaters	At the HVS equipment and heaters
Non-Safety Electrical Distribution System	Non-Safety Electrical Distribution System provides Non-Safety Category electrical power to the HVS Non-Safety Category electrical loads	At the HVS equipment
Equipment and Floor Drain	Provides drains for condensation off AHU/Air Conditioning Unit cooling coils.	At the air handler or fan coil unit flange
Safety-Category (TBD) Instrumentation and Control System	Safety-Category (TBD) Instrumentation and Control System provides Safety-Category (TBD) control to the SCR EFUs and associated dampers.	At the RB HVS equipment
Safety-Category (TBD) Instrumentation and Control System	Safety-Category (TBD) Instrumentation and Control System provides Safety-Category (TBD) control to RB supply and exhaust isolation dampers, TB supply and exhaust dampers, and CRE EFUs.	At the RB HVS equipment
Safety-Category (TBD) Electrical Distribution System	Safety-Category (TBD) Electrical Distribution System provides electrical power to SDG Rooms HVS equipment, RB supply and exhaust isolation dampers, TB Supply and exhaust isolation dampers, CRE EFUs	At the RB HVS equipment
Fire Protection System	Fire Protection System provides start/stop signals to RB HVS to shut down fans.	Fire Protection System contacts to be located within 1 m of the associated temperature control panel

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Interfacing System	Interface Description	Interface Boundary
Containment Inerting System	Containment Inerting System discharges containment inerting gases to the Continuous Exhaust Air Plenum /Plant Vent Stack during purging. RB HVS provides outside air supply to CIS for containment de-inerting	At the Containment Inerting System/RB HVS piping interface
Main Control Room Panels	Main Control Room Panels provide the status, alarms, and indications in the control room that are required for system monitoring	At the RB HVS equipment

The diagram is a complex architectural floor plan of the Reactor Building Secondary Control Room, showing various functional areas, equipment, and structural elements across multiple levels. The plan is oriented with the turbine building roof at the top and the reactor building secondary control room at the bottom.

Key Areas and Rooms:

- Operating Deck:** Located at the top left, featuring an electric unit heater (EUT 8) and a fuel pool.
- Reactor Cavity Pool:** A large rectangular pool area in the upper center.
- ICCS Pool A, B & C Equipment Pool:** Located at the bottom left, containing various equipment and piping.
- Service Areas:** Multiple service rooms (e.g., SERVICE -3.0, SERVICE -3.1, SERVICE -3.2) and a service hatch (-3.5) are distributed throughout the plan.
- Control Rooms:** The Secondary Control Room is a large central area, with a Secondary Control Room (ELEV 0) and a Secondary Control Room (ELEV 0.5) also indicated.
- Electrical Distribution:** Several electrical distribution rooms (e.g., ELECTRICAL DISTRIBUTION, ELECTRICAL DISTRIBUTION) are shown, along with a large electrical unit heater (EUT 1).
- Stairways:** Stair A and Stair B are located in the center of the plan.
- Equipment Rooms:** Rooms for equipment, including a large equipment room (EQUIPMENT) and a room for equipment (EQUIPMENT).
- Service Piping:** Extensive service piping is shown throughout the plan, connecting various rooms and equipment.
- Structural Elements:** The plan includes numerous structural details, such as walls, doors, and equipment racks, as well as a large area labeled "PRIMARY CONTAINMENT" at the bottom.

Levels and Elevation:

- ELEV 0.0:** The main level of the Reactor Building Secondary Control Room.
- ELEV 0.5:** A secondary level, likely for the turbine building roof.
- ELEV 1.0:** A level for the turbine building roof.
- ELEV 1.5:** A level for the turbine building roof.
- ELEV 2.0:** A level for the turbine building roof.
- ELEV 2.5:** A level for the turbine building roof.
- ELEV 3.0:** A level for the turbine building roof.
- ELEV 3.5:** A level for the turbine building roof.
- ELEV 4.0:** A level for the turbine building roof.
- ELEV 4.5:** A level for the turbine building roof.
- ELEV 5.0:** A level for the turbine building roof.
- ELEV 5.5:** A level for the turbine building roof.
- ELEV 6.0:** A level for the turbine building roof.
- ELEV 6.5:** A level for the turbine building roof.
- ELEV 7.0:** A level for the turbine building roof.
- ELEV 7.5:** A level for the turbine building roof.
- ELEV 8.0:** A level for the turbine building roof.
- ELEV 8.5:** A level for the turbine building roof.
- ELEV 9.0:** A level for the turbine building roof.
- ELEV 9.5:** A level for the turbine building roof.
- ELEV 10.0:** A level for the turbine building roof.
- ELEV 10.5:** A level for the turbine building roof.
- ELEV 11.0:** A level for the turbine building roof.
- ELEV 11.5:** A level for the turbine building roof.
- ELEV 12.0:** A level for the turbine building roof.
- ELEV 12.5:** A level for the turbine building roof.
- ELEV 13.0:** A level for the turbine building roof.
- ELEV 13.5:** A level for the turbine building roof.
- ELEV 14.0:** A level for the turbine building roof.
- ELEV 14.5:** A level for the turbine building roof.
- ELEV 15.0:** A level for the turbine building roof.
- ELEV 15.5:** A level for the turbine building roof.
- ELEV 16.0:** A level for the turbine building roof.
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- ELEV 17.0:** A level for the turbine building roof.
- ELEV 17.5:** A level for the turbine building roof.
- ELEV 18.0:** A level for the turbine building roof.
- ELEV 18.5:** A level for the turbine building roof.
- ELEV 19.0:** A level for the turbine building roof.
- ELEV 19.5:** A level for the turbine building roof.
- ELEV 20.0:** A level for the turbine building roof.
- ELEV 20.5:** A level for the turbine building roof.
- ELEV 21.0:** A level for the turbine building roof.
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- ELEV 22.0:** A level for the turbine building roof.
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- ELEV 26.0:** A level for the turbine building roof.
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- ELEV 70.0:** A level for the turbine building roof.
- ELEV 70.5:** A level for the turbine building roof.
- ELEV 71.0:** A level for the turbine building roof.
- ELEV 71.5:** A level for the turbine building roof.
- ELEV 72.0:** A level for the turbine building roof.
- ELEV 72.5:** A level for the turbine building roof.
- ELEV 73.0:** A level for the turbine building roof.
- ELEV 73.5:** A level for the turbine building roof.
- ELEV 74.0:** A level for the turbine building roof.
- ELEV 74.5:** A level for the turbine building roof.

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9A.5.2 Control Building Heating, Ventilation and Air Conditioning System

9A.5.2.1 System and Equipment Functions

9A.5.2.1.1 Normal Functions (Non-Safety Category)

1. Provides a controlled environment for personnel comfort and safety and equipment operation. Sufficient outside air is provided to meet the ventilation requirements for acceptable indoor air quality consistent with the applicable requirements of ASHRAE 62.1 (Reference 9A.5.2-1).
2. Provides a controlled environment for operation of equipment in the Power Block during normal, startup and shutdown operations.
3. Provides isolation features to support testing and maintenance.
4. The CB HVS supply ductwork to all the Battery Rooms, are provided electric duct heaters for more precise temperature control in those rooms.
5. The Battery Rooms are ventilated and exhausted to maintain hydrogen levels below required limits in the rooms.
6. The CRE in the Control Building is maintained at a higher pressure than surrounding areas with the CRE EFUs in service.
7. Detects and limits the introduction of airborne hazardous materials (toxic gas, radioactivity, or smoke) into the Control Room in the Control Building.
8. The HVS is designed such that failure of Non-Safety-Category function equipment does not compromise Safety-Category function equipment.
9. Provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the power block to the outside atmosphere in the event of a fire.

9A.5.2.1.2 Off-Normal Functions (Non-Safety Category)

The Control Building HVS performs several Non-Safety-Category functions during off-normal conditions. The CB HVS continues to operate during off-normal conditions until removed from service either manually or automatically.

CB HVS equipment that is normally in standby that is required to function during off-normal events include the following:

1. In the event of a fire outside the CB, the outside air through the normal operating AHU is secured and the operating AHU continues to operate in recirculation mode only.
2. In the event of a fire inside the Control Building but outside of the Control Room Envelope (CRE), a Control Room Envelope Emergency Filtration Unit can be placed in-service, thereby supplying the CRE in pressurization mode with the CRE isolated.
3. The CB is provided ventilation air during toxic gas events as follows. The CB toxic gas filtration units operate automatically when toxic gas is detected at the CB AHU outside air intakes. Normal outside air supply to the operating CB AHU isolates and the toxic gas filtration units discharge damper opens, allowing the associated toxic gas filtration unit to supply filtered outside pressurized air to the entire Control Building through the normal CB supply AHU which continues to operate during a toxic gas event. The toxic gas filtration units are equipped with HEPA and charcoal filters to remove toxic and radioactive gases and material from the outside air supply to the normal CB AHUs.

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4. In the event of high radiation detected at the CB normal air intake, the CB normal supply AHU isolates and a CRE-EFU unit auto starts to provide pressurization air to the CRE. CRE-EFU contain charcoal filtration permitting removal of radioactive iodine.
5. The CB HVS provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the Control Building.
6. Battery room exhaust fans continue to operate regardless of the operational mode of the CB AHUs, or CRE EFUs. Battery room exhaust fans operate on a timer to only energize after a set period of time (determined by the offgassing rate of the batteries) and only to be energized for the minimum time needed to reduce H₂ concentration to below 1%.
7. In the event of a buildup of smoke inside the CB requiring a purge, two exhaust fans are manually operated to remove smoke. During this time, the AHUs providing supply operate as 100% outside air with no recirculation until such time as the smoke has been sufficiently removed from the building.
8. In the event of a LOOP, the Diesel Generators activate and provide backup power to their respective train-supported equipment. This includes the normal AHUs, Control Room Envelope Emergency Filter Units including their associated electric heaters, and FCUs for the DL2 and DL4A rooms.

9A.5.2.1.3 Normal Functions (Safety Category)

1. The CB HVS provides required cooling to the DL4a Room in the CB with two redundant Safety-Category (TBD) FCUs.
2. The CB HVS provides required cooling to the Safety-Category (TBD) Instrumentation and Control System DL2 Room A and Safety-Category (TBD) Instrumentation and Control System DL2 Room B with redundant Safety-Category (TBD) FCUs in each room.

9A.5.2.1.4 Off-Normal Functions (Safety Category)

SC3 Functions:

1. The normal CB outside air intakes are monitored for airborne radioactivity. Upon detection of a high radiation condition, the operating normal supply AHU will de-energize, auto-start of the standby unit is defeated, and the CRE goes into isolation mode with normal CB supply and return isolation dampers closing. In addition CRE EFUs automatically start, pressurizing the CRE with filtered supply air, utilizing a mixture of approximately 90% return air from the CRE. The 10% outside air passes through the CRE-EFU with potential radioactive iodine being removed via charcoal filtration.
2. Tornado dampers are provided at each CB HVS outside air interface to protect the CB structure.

The design of the CB HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2, Section 8.11.2 (Reference 9A.5.2-2) as related to providing a ventilation system with a filtration system capable of the following:

1. Preventing unacceptable dispersion of airborne contaminants within the plant by monitoring the normal CB outside air intakes for radioactivity. Upon detection of radioactivity the CRE goes into isolation mode with normal CB supply and return isolation dampers going closed. CRE-EFU automatically starts, pressurizing the CRE

with filtered supply air, utilizing a mixture of approximately 90% return air from the CRE. The 10% outside air passes through the CRE-EFU with potential radioactive iodine being removed via charcoal filtration.

2. Reducing the concentration of airborne radioactive substances to levels compatible with the need for access to each particular area is achieved by using shielding, ventilation, monitoring instrumentation and ALARA design concepts as discussed in Chapter 12 to ensure the overall design minimizes radiation exposure to workers and to the public.
3. Keeping the level of airborne radioactive substances in the plant below prescribed limits. The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance is below 0.1 the derived air concentration as specified in 10 CFR 20 Appendix B (Reference 9A.5.2-3) during normal power operation.
4. Application of ALARA design principles in normal operation as discussed in Chapter 12, Subsection 12.1.5.4.
5. Ventilating rooms containing inert or noxious gases without impairing the capability to control radioactive releases by using charcoal filters that have properties for toxic gas adsorption.

9A.5.2.2 Safety Design Bases

The safety design bases associated with the Control Building HVS include the following:

1. The Safety-Category (TBD) CB HVS ensures that a habitable environment is maintained in the MCR.
2. The CB HVS maintains a slight positive pressure in the CRE during emergency events.

9A.5.2.3 Description

Figure 9A.5.2-1 depicts the Control Building HVAC Process Flow Diagram.

The Control Building HVS subsystems are comprised of redundant trains of equipment that support all CB HVS functions during normal operating modes.

The Battery Room exhaust fans in the Control Building are backed up by batter power.

The CRE in the Control Building is maintained at a higher pressure than surrounding areas with the CRE-EFU in service.

The Control Room HVS detects and limits the introduction of airborne hazardous materials (radioactivity, or smoke) into the Control Room in the Control Building. This system is designed with the capability to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire. Smoke control and removal functions are in accordance with National Fire Protection Association (NFPA); NFPA 92A and CSA N293/293S1 and National Building Code of Canada (NBCC) and National Fire Code of Canada as described in Subsection 9A.6).

The MCR is provided filtered, conditioned air from two 100% capacity AHUs via supply and return ductwork. Refer to Chapter 6, Subsection 6.4 for information pertaining to Control Room Habitability.

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The battery rooms are each provided with two 50% capacity exhaust fans which operate based off timers. Other exhaust fans are provided for the restroom/shower areas and the control room restroom/janitor's closet.

The CB is a single-story building. Some rooms contain significant quantities of electrical equipment and associated heat dissipation loads. Other rooms, such as the MCR, Radiation Protection offices, and Conference Room, contain areas of less heat density. The Battery Rooms, Restrooms, Break Room, and Janitor's Closet are also low heat density rooms and will be served by exhaust fans exhausting to the outdoors.

The CB is normally air conditioned and heated by 2 x 100% capacity chilled water supplied AHUs located on the CB roof, discharging through supply ductwork and a return plenum. Approximately 10% fresh makeup air is brought into the operating AHU. In addition to the chilled water coils, the AHUs contain electric heating coils, humidification/dehumidification features, filtration, and isolation dampers. Supply ductwork upstream of the MCR and offices is provided with Volume Control Units to vary supply air flow based on thermostatic control. AHU cooling in the high heat load rooms is supplemented by chilled water supplied FCUs with a Train A and a Train B FCU located in each of these rooms.

The Battery Rooms are maintained at the battery manufacturer recommended room temperature. Battery Room temperatures are maintained via Battery Room supply duct electric duct heaters and thermostats located in the rooms.

The outside air intake is instrumented to analyze for toxic gases and for airborne radioactivity as applicable. To accommodate MCR habitability contingencies the following additional AHUs are provided for the CB:

1. CRE-EFU that operate automatically upon detection of high radiation level at the operating CB supply AHU outside air intake. The CRE goes into isolation mode with normal CB supply and return air dampers going closed. A mixture of approximately 90% return air from the CRE and 10% outside air will pass through the EFU with potential radioactive iodine being removed via charcoal filtration. EFU operation maintains the CRE slightly pressurized. The operating CB normal supply AHU de-energizes upon high radiation detection at the intake and auto-start of the standby unit is defeated. Battery Room exhaust fans continue to operate based off timers.
2. CB toxic gas filtration units that operate automatically when toxic gas has been detected at the CB AHU outside air intakes. In this event, normal outside air supply to the operating CB AHU isolates and the toxic gas filtration unit discharge damper opens, allowing the associated toxic gas filtration unit to supply pressurization air to the CB through the normal CB supply AHU, which continues to operate during a toxic gas event.

The CB normal supply AHUs are provided ASD to be able to reduce flow as conditions change.

CRE isolation and CRE-EFU operation provide a pressurized envelope with respect to adjacent spaces, maintaining CRE habitability during an airborne radiation event. Toxic gas filtration unit operation maintains the entire CB pressurized, maintaining CB habitability during a toxic gas event.

Two smoke exhaust fans are provided for the CB. During recovery from a fire, smoke is exhausted from the CB by operating a CB normal supply AHU in 100% outside air mode in conjunction with operation of two CB smoke exhaust fans.

The CB HVS is provided tornado rated dampers as necessary to withstand high wind events.

9A.5.2.3.1 Component Description

Filters

The various building filtration levels are specified within each specific Subsection. Filters meet the applicable efficiency rating as stated below. American Society of Heating, Refrigerating and Air Conditioning Engineers Standard 52.2 (Reference 9A.5.2-4) establishes a filter's MERV and establishes a filter's Average Atmospheric Dust Spot Efficiency. The following ASHRAE filter classifications are MERV specified (with Dust Spot Efficiency in parenthesis) below:

- Low Efficiency: MERV (Less than 20%)
- Medium Efficiency: MERV (As least 20% but less than 80%)
- High Efficiency: MERV (Greater than or equal to 80%)

HEPA Filters are specified for various building filtration systems. Filters meet the applicable efficiency rating as stated below.

Filters with efficiency greater than MERV 16 by ASHRAE Standard 52.2 (MERV 17-20) (Reference 9A.5.2-4) are usually rated by the dioctylphthalate test method. This test is based on the ability of a filter to remove an aerosol consisting of 0.3 micrometer (micron) particles of a test challenge. HEPA filters are extended-medium dry-type filters in a rigid frame, having minimum particle-collection efficiency of 99.97% on 0.3-micron particles which meets ASME AG-1, Code on Nuclear Air and Gas Treatment, Section FC, HEPA Filters (Reference 9A.5.2-5).

HEPA filters are constructed, qualified, and tested per Underwriters Laboratory-586, High Efficiency, Particulate, Air Filter Units (Reference 9A.5.2-6).

Air Handler Units

Each AHU consists of an inlet area, filters (as specified by the system), heating elements (coils), cooling coils (as required) and the respective fans (supply or exhaust). The Air-Cleaning Units and Components are designed in accordance with ASME/ANSI AG-1-2019 Code on Nuclear Air and Gas Treatment (Reference 9A.5.2-7).

Supply and Exhaust Fans

The various building ventilation systems are provided with supply and exhaust fans, which are sometimes incorporated into AHUs. These fans are either centrifugal or axial fans depending on the suitability to the specific system. The fans are designed, manufactured, and supplied in accordance with the standards of AMCA. Fans in various areas are equipped with Adjustable Speed Drive mechanisms to control airflows for the specific system application.

Heating Coils/Elements

Various AHUs are equipped with electrical heating coils/elements. Electric coils are designed and supplied to the requirements of Underwriters Laboratory, Heating and Cooling Equipment (Reference 9A.5.2-8).

Cooling Coils

The cooling coils are designed, constructed, and installed in accordance with ASHRAE 33 (Reference 9A.5.2-9), Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils, and ANSI/ARI 410 (Reference 9A.5.2-10) and Underwriters Laboratory (Reference 9A.5.2-8).

Cooling coil condensate is collected in drain pans within the air handler units with the drain pan discharge (condensate) routed to a floor drain located within the room. These floor drains connect to the EFS (Chapter 9, Subsection 9A.9.3).

9A.5.2.4 Materials

Refer to Subsection 9A.5.1 for information pertaining to HVS materials.

9A.5.2.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.5.2-1 for Control Building HVS interfaces with other equipment or systems.

9A.5.2.6 System and Equipment Operation

9A.5.2.6.1 CB HVS Normal Operations

The CB HVS is provided heated, conditioned, and filtered supply air from one of two operating AHUs through supply ductwork, with approximately 90% of the air recirculated back to the units. The second AHU is in standby, ready for auto-start should the operating unit fail. The remaining 10% of the supply air is made up by outside air brought into the operating AHU located on the CB roof. The two exhaust fans provided for each Battery Room are operating based off timers, exhausting to the outdoors. Exhaust fans for the restrooms, break room, and janitors' closet are all expected to be operating continuously, exhausting to the outdoors. An electric duct heater in the supply duct to each Battery Room is normally operating as required to maintain Battery Room temperatures near the optimal battery temperature as recommended by the battery manufacturer.

9A.5.2.6.2 CB HVS Off-Normal Operations

In the event of an SBO, power is lost to the CB AHUs which normally provide heating and air conditioning for the CB, as well as electric duct heaters and cabinet heaters. The restroom, break room, and janitors closet exhaust fans become inoperable. The Control Room operators relocate to the SCR in the RB if the MCR becomes uninhabitable.

In the event of a CB normal outside air intake radiation monitor alarm, the operating normal supply AHU de-energizes and any associated toxic gas filtration units also de-energize, and the same train CRE-EFU automatically starts. The CRE isolation dampers close in conjunction with the start of the CRE-EFU. Diesel generator backup power is provided to the CRE-EFU supply fans and associated heating coils. Restroom, break room, and janitors closet exhaust fans become inoperable. The Battery Room exhaust fans continue to operate off timers using backup power, preventing possible accumulations of off-gassed hydrogen to the rooms; the timers ensure the exhaust fans only run as needed to eliminate hydrogen buildup. Electric duct and cabinet heaters are de-energized.

In the event of a CB normal outside air intake toxic gas detection, the operating normal AHU normal outside air intake closes, and the same train toxic gas filtration unit automatically starts and its associated motor control dampers automatically open, providing an alternate source of filtered outside air for CB pressurization. Diesel Generator backup power is provided to the CB normal supply AHU fans and associated heating coils, as well as the associated Toxic Gas Filtration Unit fan. Restroom, break room, and janitors closet exhaust fans become inoperable. The Battery Room exhaust fans continue to operate off timers using backup power, preventing possible accumulations of off-gassed hydrogen to the rooms; the timers ensure the exhaust fans only run as needed to eliminate hydrogen buildup. Electric duct and cabinet heaters are de-energized. Positive pressure is maintained in the CB by way of providing outside air through the normal supply AHU, the toxic gas filtration unit, or the CRE-EFU.

9A.5.2.7 Instrumentation and Control

The following signals are provided as inputs to the control logic:

1. Temperature elements to monitor space or duct air temperatures, and to control cooling water control valves, electric heating elements, or control fan speeds."

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2. Moisture element in the MCR to engage the CB Normal Supply AHU humidification and dehumidification control.
3. Differential pressure transmitters and air flow instruments, as required, to monitor fan operation.
4. Differential pressure transmitters to monitor pressure drop across AHU filters.
5. Intake, exhaust, and return air damper position monitoring.
6. Duct mounted smoke detectors to shut down fans as required.

The HVS subsystems have dedicated control panels that regulate the temperature of the spaces to within specified ranges.

Manual initiation and shutdown of the CB HVS is provided from the CB Temperature Control Panel located in the Technical Support Center.

The following control features are implemented in the CB HVS:

1. The CB AHUs, supply fans, and exhaust fans have the capability of being remotely started from the Emergency Operations Center temperature control panel. Bathroom exhaust fans are wired directly from lighting panel, locally controlled with no DCIS. To avoid air infiltration from the surrounding areas, all rooms in the CB with the exception of the janitor's closet, restrooms, breakroom, Battery Rooms and stairwell are positively pressurized by the introduction of filtered outside air and by maintaining the pressure boundary integrity.
2. Battery Room exhaust fans operate off timers. Failure of any Battery Room exhaust fan is annunciated via a common CB HVS alarm in the MCR.
3. Any fan started in standby because of a low airflow condition automatically trips the fan that generated the low flow condition after confirmation that the standby fan is running. To avoid spurious standby starts and fan trips, all airflow logic incorporates time delays to allow fans to generate steady-state flow.
4. The CB differential pressure transmitters are used to monitor the pressure in the MCR and alarm low differential pressure.
5. Control valves located at the cooling coils modulate the flow of chilled water in AHUs in response to air temperature signals from the CB HVS subsystems .
6. Local temperature switches cycle ON/OFF electric cabinet heaters cycle "ON/OFF."
7. The humidifiers in the CB AHU are controlled by the moisture elements located in the MCR.
8. The outside air normal intake dampers to the CB AHUs cannot be opened if smoke, toxic gas, or radiation is detected.
9. The CB recirculation air damper cannot be closed if radiation or smoke is detected in the outside air intake.
10. Upon detection of toxic gas at an outside air normal intake, the normal outside air intake damper closes and the associated toxic gas filtration unit and its associated suction and discharge dampers open, providing an alternate, filtered source of outside makeup air to the operating normal supply AHUs.

11. Upon receipt of an outside air normal intake radiation monitor alarm, the operating normal supply automatically stop, and the associated same train automatically starts, pressurizing the CRE and protecting CRE staff from possible radioactive iodine.
12. The Fire Protection System provides a signal to shut down the CB normal and most HVAC fans in the event of a fire. The Battery Room exhaust fans continue to operate, preventing a potentially explosive hydrogen concentration from occurring. Smoke removal is also performed in conjunction with the FPS. Duct mounted smoke detectors provide a smoke detected signal to the FPS local fire protection panels.

9A.5.2.8 Monitoring, Inspection, Testing, and Maintenance

Air handlers are field tested per ASME N511, In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems (Reference 9A.5.2-11).

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Personnel and lay-down access are provided around instruments to allow adequate space for maintenance purposes.

Component Level Requirements

Charcoal beds used for filtration of radiological effluents meet ALARA requirements as discussed in Chapter 12, Section 12.3.

9A.5.2.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.5.2.10 Performance and Safety Evaluation

In the event of a LOOP, diesel generator backup power is provided to the CB AHU supply fans. In the event of an SBO, power is lost to the CB AHUs. However, for both the LOOP and SBO cases the Battery Room exhaust fans continue to operate off timers using backup battery power in order to prevent potential accumulations of off-gassed hydrogen to the Battery Rooms. The timers ensure the exhaust fans only run as needed to eliminate hydrogen buildup. In the event the MCR becomes uninhabitable the Control Room operators relocate to the SCR in the RB.

CB HVS design includes:

1. Redundant Safety-Category (TBD) FCUs which provides cooling to the DL4a room.
2. Redundant Safety-Category (TBD) FCUs which provides required cooling to the Safety-Category (TBD) Instrumentation and Control System DL2 Room A and Safety-Category (TBD) Instrumentation and Control System DL2 Room B.

The RB HVS includes Safety-Category (TBD), 100% capacity pressurization fans with emergency filtration units that supply filtered outside air to the SCR needed to pressurize the space relative to adjacent spaces, protecting the operators from hazards such as release of radioactive materials, fire, or smoke in the RB. The pressurization fans are provided with backup power making them available for use during a LOOP or SBO.

9A.5.2.11 References

9A.5.2-1 ASHRAE 62.1, "Ventilation for Acceptable Indoor Air Quality."

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- 9A.5.2-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.2-3 10 CFR 20 Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentration for Release to Sewerage."
- 9A.5.2-4 ASHRAE 52.2, "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size."
- 9A.5.2-5 ASME AG-1, "Code on Nuclear Air and Gas Treatment," Section FC, "HEPA Filters," American Society of Mechanical Engineers.
- 9A.5.2-6 UL-586, "High Efficiency, Particulate, Air Filter Units," Underwriters Laboratory.
- 9A.5.2-7 ASME AG-1, "Code on Nuclear Air and Gas Treatment."
- 9A.5.2-8 UL 1995, "Heating and Cooling Equipment," Underwriters Laboratory.
- 9A.5.2-9 ASHRAE 33, "Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils."
- 9A.5.2-10 AHRI Standard 410, "Forced Circulation Air-Cooling and Air-Heating Coils," Air Conditioning, Heating, and Refrigeration Institute.
- 9A.5.2-11 ASME N511, "In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems," American Society of Mechanical Engineers.

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Table 9A.5.2-1: Control Building Heating, Ventilation and Air Conditioning System Interfaces

Interfacing System	Interface Description	Interface Boundary
Non-Safety Instrumentation and Controls	The Non-Safety Instrumentation and Controls provides distributed control and instrumentation data communication networks.	At the HVS equipment
Process Radiation and Environmental Monitoring	The PREMS provides continuous radiation and toxic gas monitoring of the HVS.	At the HVS equipment
Chilled Water Equipment	The CWE provides chilled water inside coils for cooling of the HVAC FCUs and AHUs.	At the HVS equipment
Safety-Category (TBD) Electrical Distribution System	The Safety-Category (TBD) Electrical Distribution System provides electrical power to SCR emergency pressurization fans and electric duct heaters.	At the HVS equipment
Safety-Category (TBD) Electrical Distribution System	The Safety-Category (TBD) Electrical Distribution System provides electrical power to SDG Rooms HVS equipment, RB supply and exhaust isolation dampers, TB Supply and exhaust isolation dampers, CRE EFUs.	At the HVS equipment
Non-Safety Electrical Distribution System	The Non-Safety Electrical Distribution System provides Non-Safety Category electrical power to the HVS Non-Safety Category electrical loads.	At the HVS equipment
Water Gas and Chemical Pads	Provides clean water supply used for space humidification if needed. CB AHU & FCU condensate will be collected in drains.	At the HVS equipment
Main Control Room Panels	Main Control Room Panels provide the status, alarms, and indications in the control room that are required for system monitoring.	At the HVS equipment

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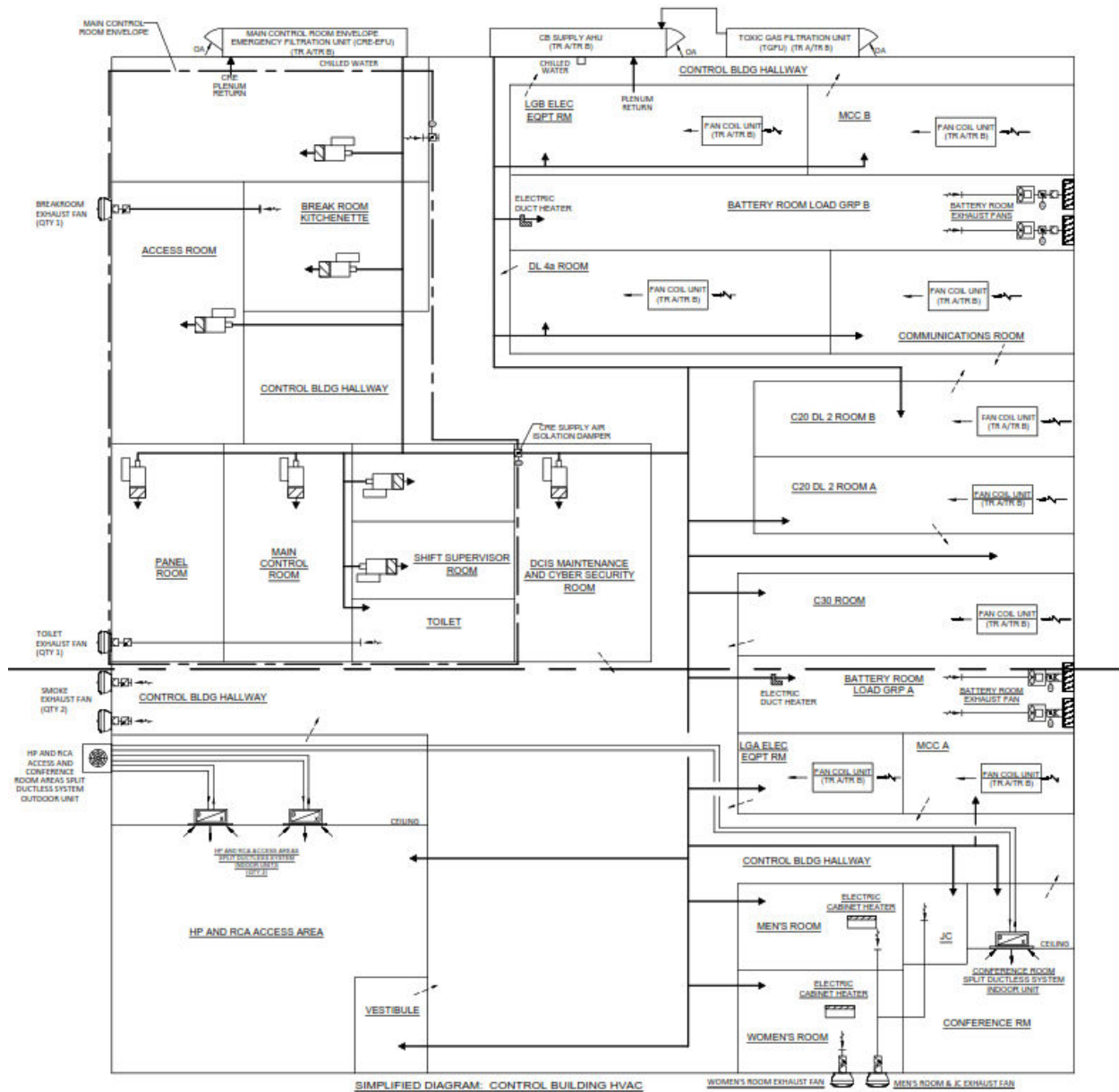


Figure 9A.5.2-1: Control Building HVAC

9A.5.3 Radwaste Building Heating, Ventilation and Air Conditioning Systems

The Radwaste Building contains room-temperature tanks and small to medium-sized pumps. The RWB is not normally occupied. However, plant personnel access and spend time in the Chem Lab and dress out areas within the RWB. As such, most of the building is ventilated to five air changes/hour.

The RWB Chem Lab is provided an electrical cabinet heater and a Fan Coil Unit to supplement AHU HVAC needs. The Chem Lab fume hood exhaust to the RWB exhaust AHU suction, with final discharge to the PVS.

9A.5.3.1 System and Equipment Functions

9A.5.3.1.1 Normal Functions (Non-Safety Category)

1. The RWB HVS provides a controlled environment for personnel comfort and safety and equipment operation. Sufficient outside air is provided to meet the ventilation requirements for acceptable indoor air quality consistent with the applicable requirements of ASHRAE 62.1 (Reference 9A.5.3-1).
2. The RWB HVS provides a controlled environment for the proper operation and integrity of equipment in the RWB during normal, startup and shutdown operations.
3. The RWB HVS provides necessary isolation features to support testing and maintenance.
4. The RWB HVS minimizes exposure to personnel during inspection and maintenance by locating equipment and instrumentation as far as practical from potential sources of high radiation.
5. The RWB HVS maintains designated clean areas higher than atmospheric (positive) pressure to minimize the infiltration of air.
6. The RWB HVS maintains negative pressurization of potentially contaminated areas to control leakage of potentially radioactive effluent to the atmosphere, or other rooms needing to be maintained at negative pressure including the Charcoal Adsorber Vault.
7. The RWB HVS provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire.

9A.5.3.1.2 Off-Normal Functions (Safety-Category)

The RWB HVS does not perform any Safety-Category functions.

The Radwaste Building HVS is not required to operate during or after a design basis event. The design of the RWB HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.11.2 (Reference 9A.5.3-2) as related to providing a ventilation system with a filtration system capable of the following:

1. Preventing unacceptable dispersion of airborne contaminants within the plant through the use of the exhaust AHUs to maintain a negative pressure in clean areas, resulting in air transferring from clean to "dirty" areas."
2. Reducing the concentration of airborne radioactive substances to levels compatible with the need for access to each particular area is achieved by using shielding, ventilation, monitoring instrumentation and ALARA design concepts as discussed in

Chapter 12 to ensure the overall design minimizes radiation exposure to workers and to the public.

3. Keeping the level of airborne radioactive substances in the plant below prescribed limits. The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance is below 0.1 the derived air concentration as specified in 10 CFR 20 Appendix B (Reference 9A.5.2-3) during normal power operation.
4. Application of ALARA design principles in normal operation as discussed in Subsection 12.1.5.4.
5. Ventilating rooms containing inert or noxious gases without impairing the capability to control radioactive releases using HEPA filters. The HEPA filters associated with the RWB HVS exhaust AHUs assist in ensuring radioactive material entrained in gaseous effluent does not exceed the limits specified in SOR 2000-203 (Reference 9A.5.3-4), for normal operations and anticipated operational occurrences.

9A.5.3.1.3 Normal Function Safety-Category

The RWB HVS does not perform any Safety-Category functions.

9A.5.3.1.4 Off-Normal Function Non-Safety-Category

The Radwaste Building HVS no longer receives chilled water support during off-normal conditions and is removed from service either manually or automatically.

9A.5.3.2 Safety Design Bases

The Radwaste Building HVS does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

9A.5.3.3 Description

Figure 9A.5.3-1 depicts the Radwaste Building HVAC Process Flow Diagram.

RWB HVS is comprised of two 100% capacity supply AHUs and two 100% capacity exhaust AHUs with HEPA filtration, functioning in a push-pull manner. In general, clean filtered outside air, heated when needed, is supplied through ductwork to the lobby and Chem Lab, pressurizing those areas. The exhaust AHUs take suction on the potentially contaminated tank and pump rooms, creating negative pressures in those spaces, resulting in air transferring from clean to "dirty" areas." ASDs are provided for each of the four AHUs, permitting air flow to be ramped down in the wintertime, saving on heating load. Cooling in the lobby area and the Chem Lab is supplemented by chilled water supplied FCUs. The Chem Lab is also provided a small cabinet heater, and the lobby area is provided with electric unit heaters for exterior wall heating.

During recovery from a fire, smoke is exhausted from the RWB by operating the normal supply AHUs in 100% outside air mode in conjunction with the RWB exhaust AHUs.

9A.5.3.3.1 Component Description

Filters

RWB HVS filters meet the applicable efficiency rating as stated below. ASHRAE Standard 52.2 (Reference 9A.5.3-5) establishes a filter's MERV and establishes a filter's Average Atmospheric Dust Spot Efficiency. The following ASHRAE filter classifications are MERV specified (with Dust Spot Efficiency in parenthesis) below:

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Low Efficiency:	MERV 1-4 (Less than 20%)
Medium Efficiency:	MERV 5-12 (As least 20% but less than 80%)
High Efficiency:	MERV 13-16 (Greater than or equal to 80%)

HEPA filters are specified for various building filtration systems. Filters meet the applicable efficiency rating as stated below.

Filters with efficiency greater than MERV 16 by ASHRAE Standard 52.2 (MERV 17-20) (Reference 9A.5.3-5) are usually rated by the dioctylphthalate test method. This test is based on the ability of a filter to remove an aerosol consisting of 0.3 micrometer (micron) particles of a test challenge. HEPA filters are extended-medium dry-type filters in a rigid frame, having minimum particle- collection efficiency of 99.97% on 0.3-micron particles which meets ASME AG-1, Code on Nuclear Air and Gas Treatment, Section FC, HEPA Filters (Reference 9A.5.3-6).

HEPA filters are constructed, qualified, and tested per Underwriters Laboratory-586, High Efficiency, Particulate, Air Filter Units (Reference 9A.5.3-7).

Air Handler Units

Each AHU consists of an inlet area, filters (as specified by the system), heating elements (coils), cooling coils (as required) and the respective fans (supply or exhaust). The Air-Cleaning Units and Components are designed in accordance with ASME/ANSI AG-1 Code on Nuclear Air and Gas Treatment (Reference 9A.5.3-8).

Supply and Exhaust Fans

The various building ventilation systems are provided with supply and exhaust fans, which are sometimes incorporated into AHUs. These fans are either centrifugal or axial fans depending on the suitability to the specific system. The fans are designed, manufactured, and supplied in accordance with the applicable standards of AMCA. Fans are equipped with Adjustable Speed Drive mechanisms to control airflows for the specific system application.

Heating Coils/Elements

Various AHUs are equipped with electrical heating coils/elements. Electric coils are designed and supplied to the requirements of Underwriters Laboratory, Heating and Cooling Equipment (9A.5.3-6).

Cooling Coils

The cooling coils are designed, constructed, and installed in accordance with ASHRAE 33, Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils (Reference 9A.5.3-9), and ANSI/ARI 410 (Reference 9A.5.3-10) and Underwriters Laboratory, (Reference 9A.5.3-11).

Cooling coil condensate is collected in drain pans within the air handler units with the drain pan discharge (condensate) routed to a floor drain located within the room. These floor drains connect to the EFS (Subsection 9A.9.3) subsystem.

9A.5.3.4 Materials

Refer to Subsection 9A.5.1 for information pertaining to HVS materials.

9A.5.3.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.5.3-1 for Radwaste Building HVS interfaces with other equipment or systems.

9A.5.3.6 System and Equipment Operation

9A.5.3.6.1 Normal Operations

The RWB will normally be provided heated, filtered once-through supply air from two 50% capacity operating AHUs located on the second elevation of the RWB through supply ductwork. Outside air will be drawn in the RWB supply AHUs through suction ductwork and intake louvers located in the nearby exterior wall.

Air is exhausted from the RWB through two 50% capacity operating exhaust AHUs located on the RWB roof, which discharge to atmosphere from the PVS. AHU cooling provided by ventilation air is supplemented by cooling provided by FCUs. FCUs are controlled by room thermostats. AHU heating is supplemented by electric unit heater.

The RWB Chem Lab is provided an electrical cabinet heater and an FCU to supplement AHU HVAC needs, both are controlled by local thermostats. The Chem Lab fume hood exhausts to RWB exhaust AHU suction, with final discharge to the PVS.

9A.5.3.6.2 Off-Normal Operations

In the event of a LOOP or SBO, power is lost to all the RWB HVS equipment including the supply and exhaust AHUs, FCUs, electric unit heaters, and the Chem Lab cabinet heater and fume hood.

9A.5.3.7 Instrumentation and Control

The following signals are provided as inputs to the control logic:

1. Temperature elements to monitor space or duct air temperatures, and to control electric heating elements, or control fan speeds.
2. Differential pressure transmitters and air flow instruments, as required, to monitor fan operation.
3. Differential pressure transmitters to monitor pressure drop across AHU filters.
4. Intake, exhaust, and return air damper position monitoring.

Duct mounted smoke detectors to shut down fans as required.

Manual remote initiation and shutdown of the RWB HVS is provided from the RWB temperature control panel located in the Chem Lab.

The following control features are implemented in Radwaste Building HVS:

Controls for RWB HVS supply and exhaust AHUs are located in the RWB Chem Lab. All RWB AHUs have the capability of being manually started or stopped from the RWB temperature control panel.

RWB HVS AHU Operation and Interlocks:

1. Failure to start an AHU automatically starts the other AHU
2. An AHU cannot start until its shutoff dampers are confirmed 100% open
3. An AHU start request automatically opens as associated shutoff damper
4. Low flow at the inlet of the AHU trips the AHU. AHU trip initiates startup of the standby AHU
5. Redundant AHU has a "standby" or "auto" mode of operation
6. An AHU trip automatically closes its associated dampers

7. The Fire Protection System (Section 9A.6) provides a signal to shut down the RWB AHUs and most HVAC fans in the event of a fire. Smoke removal also is performed in conjunction with the Fire Protection System. Duct mounted smoke detectors provide a smoke detected signal to the Fire Protection System local fire protection panels.

Radwaste Building HVS unit heater operation and interlocks:

- A unit heater is started automatically by a local thermostat when heating is needed.

9A.5.3.8 Monitoring, Inspection, Testing, and Maintenance

Air handlers are field tested per ASME N511, In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems (Reference 9A.5.3-12).

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Personnel and lay-down access are provided around instruments to allow adequate space for maintenance purposes.

9A.5.3.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.5.3.10 Performance and Safety Evaluation

The radwaste building HVAC system does not perform any Safety-Category functions and therefore requires no nuclear safety evaluation.

Operational failure of any single unit of the RWB HVS does not prevent Safety Class equipment from performing any Safety-Category function.

9A.5.3.11 References

- 9A.5.3-1 ANSI/ASHRAE 62.1, "Ventilation for Acceptable Indoor Air Quality."
- 9A.5.3-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.3-3 10 CFR 20 Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentration for Release to Sewerage."
- 9A.5.3-4 Government of Canada SOR/2020-202, "General Nuclear Safety and Control Regulations."
- 9A.5.3-5 ANSI/ASHRAE 52.2, "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size."
- 9A.5.3-6 ASME AG-1, "Code on Nuclear Air and Gas Treatment," Section FC, "HEPA Filters," American Society of Mechanical Engineers.
- 9A.5.3-7 UL-586, "High Efficiency, Particulate, Air Filter Units," Underwriters Laboratory.
- 9A.5.3-8 ASME AG-1, "Code on Nuclear Air and Gas Treatment," American Society of Mechanical Engineers.

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- 9A.5.3-9 ANSI/ASHRAE 33, "Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils."
- 9A.5.3-10 AHRI Standard 410, "Forced Circulation Air-Cooling and Air-Heating Coils."
- 9A.5.3-11 UL 1995, "Heating and Cooling Equipment," Underwriters Laboratory.
- 9A.5.3-12 ASME N511, "In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems," American Society of Mechanical Engineers.

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Table 9A.5.3-1: Radwaste Building Heating, Ventilation and Air Conditioning System Interfaces

Interfacing System	Interface Description	Interface Boundary
Non-Safety Instrumentation and Controls	The Non-Safety Instrumentation and Controls provides distributed control and instrumentation data communication networks.	At the HVS equipment
Process Radiation and Environmental Monitoring	The PREMS provides continuous radiation monitoring of the RWB HVS.	At the HVS equipment
Chilled Water Equipment	The CWE provides chilled water inside coils for cooling of the HVAC FCUs and AHUs.	At the HVS equipment
Non-Safety Electrical Distribution System	The Non-Safety Electrical Distribution System provides Non-Safety Category electrical power to the RWB HVS Non-Safety Category electrical loads.	At the HVS equipment
Equipment and Floor Drain	Provides drains for condensation off AHU/Air Conditioning cooling coils.	At the air handler or fan coil unit flange
Fire Protection System	Fire Protection System provides start/stop signals to RWB HVS to shut down fans.	Fire Protection System contacts to be located within 1 m of the associated temperature control panel
Main Control Room Panels	Main Control Room Panels provide the status, alarms, and indications in the control room that are required for system monitoring.	At the RWB HVS equipment

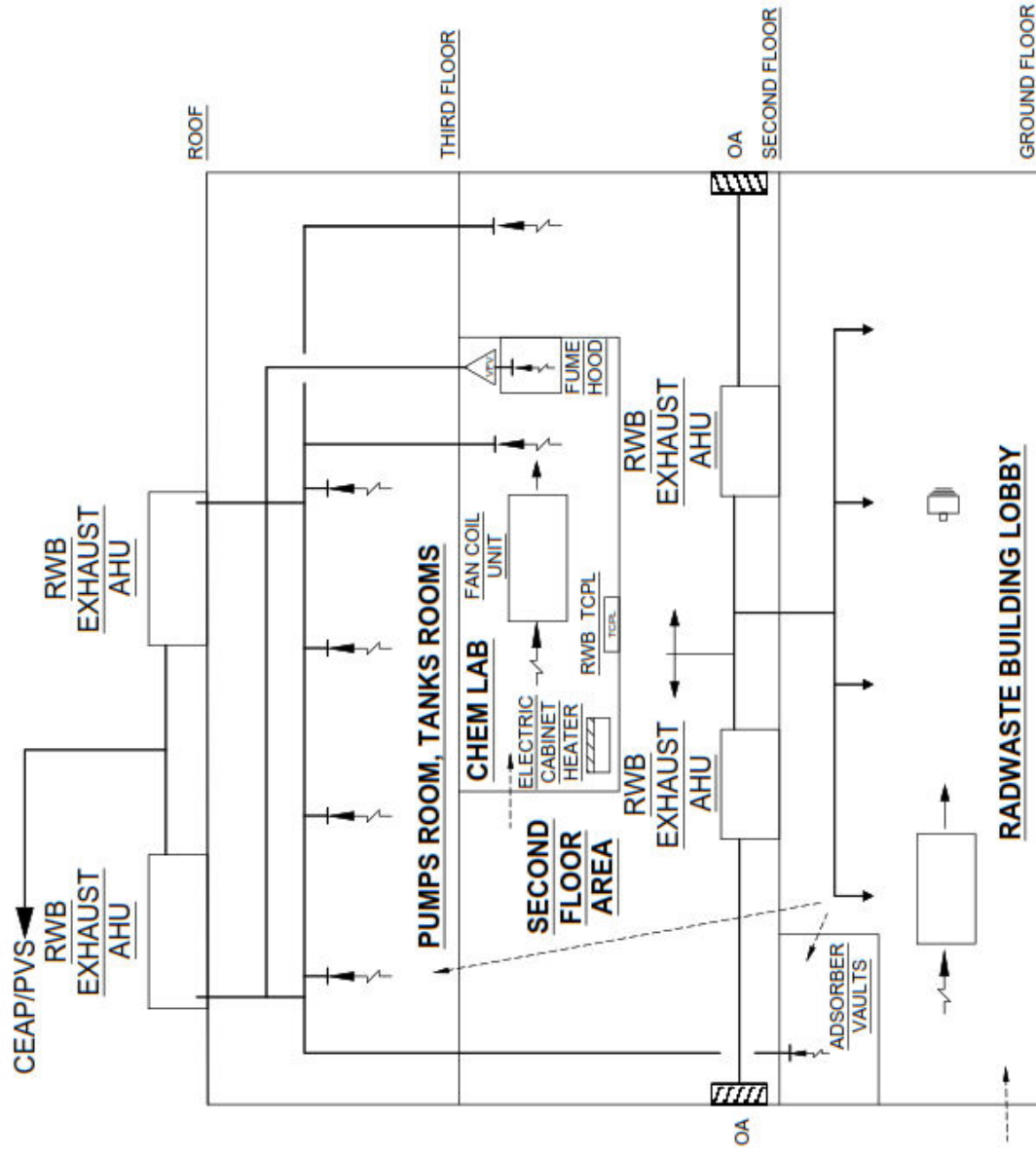


Figure 9A.5.3-1: Radwaste Building HVAC

9A.5.4 Turbine Building, Heating, Ventilation and Air-Condition System

System and Equipment Functions

9A.5.4.1 System and Equipment Functions

9A.5.4.1.1 Normal Functions (Non-Safety Category)

1. The TB HVS provides a controlled environment for personnel comfort and safety and equipment operation. Sufficient outside air is provided to meet the ventilation requirements for acceptable indoor air quality consistent with the applicable requirements of ASHRAE 62.1 (Reference 9A.5.4-1).
2. The TB HVS provides a controlled environment for the proper operation and integrity of equipment in the TB during normal, startup and shutdown operations.
3. The TB HVS provides redundant active components to increase reliability, availability, and maintainability of the ventilation system.
4. The TB HVS provides isolation features to support testing and maintenance.
5. The TB HVS minimizes exposure to personnel during inspection and maintenance by locating equipment and instrumentation as far as practical from potential sources of high radiation.
6. The TB HVS maintains designated clean areas higher than atmospheric (positive) pressure to minimize the infiltration of air.
7. The TB HVS maintains negative pressurization of potentially contaminated areas to control leakage of potentially radioactive effluent to the atmosphere, or other rooms needing to be maintained at negative pressure
8. The TB HVS provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire.
9. The TB Battery Room is ventilated and exhausted to maintain hydrogen levels below required limits in the room.
10. The TB Continuous Exhaust Air Plenum collects and mixes potentially radioactive discharge air from various buildings and releases the mixed air to atmosphere via the PVS.

9A.5.4.1.2 Normal Functions (Safety-Category)

The TB HVS provides a controlled environment for the proper operation and integrity of Safety Class instrumentation located in the TB during normal, startup and shutdown operations.

9A.5.4.1.3 Off-Normal Functions (Non-Safety-Category)

TB HVS systems continue to operate during off-normal conditions until removed from service either manually or automatically.

9A.5.4.1.4 Off-Normal Functions (Safety-Category)

1. The TB HVS includes supply and exhaust isolation dampers for the TB, which close automatically in the event of a TB high radiation input from the PREMS (Subsection 9A.3.1), isolating the TB.
2. Tornado dampers are provided at the TB Diesel Generator room for the outside air intake and the DG heat exhaust louver to protect the DG structure.

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The TB HVS is not required to operate during or after a design basis event. The design of the TB HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.11.2 (Reference 9A.5.4-2) as related to providing a ventilation system with a filtration system capable of the following:

1. Preventing unacceptable dispersion of all airborne contaminants within the plant using air flow to effect transfer of air from clean areas to potentially contaminated areas.
2. Reducing the concentration of airborne radioactive substances to levels compatible with the need for access to each particular area is achieved by using shielding, ventilation, monitoring instrumentation and ALARA design concepts as discussed in Chapter 12 to ensure the overall design minimizes radiation exposure to workers and to the public.
3. Keeping the level of airborne radioactive substances in the plant below prescribed limits. The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance is below 0.1 the derived air concentration as specified in 10 CFR 20 Appendix B (Reference 9A.5.2-3) during normal power operation.
4. Application of ALARA design principles in normal operation as discussed in Subsection 12.1.5.4.
5. Ventilating rooms containing inert or noxious gases without impairing the capability to control radioactive releases using HEPA filters. The HEPA filters associated with the TB HVS exhaust AHUs assist in ensuring radioactive material entrained in gaseous effluent does not exceed the limits specified in SOR 2000-203 (Reference 9A.5.3-4), for normal operations and anticipated operational occurrences.

9A.5.4.2 Safety Design Bases

The TB HVS is not required to operate during or following a Design Basis Accident. The safety design bases associated with the TB HVS is provided below:

1. Safety-Category (TBD) Tornado dampers are provided at the TB DG room for the Outside Air intake and the DG heat exhaust louver to protect the DG structure.
2. As part of Defense Line 2, the TB HVS provide the capability to isolate ventilation to the TB when signaled to do so from the Safety-Category (TBD) Instrumentation and Control System.
3. TB HVS maintains temperatures in the TB required to support operation of Safety Class instrumentation.

9A.5.4.3 Description

Figure 9A.5.4-1 depicts the Turbine Building HVAC Process Flow Diagram.

The TB is a large building with significant cooling loads. It is divided into two main areas: The potentially contaminated areas inside the shield area, which houses the Main Turbine, Main Condenser, Moisture Separator Reheater, Feedwater Heaters and other significant heat loads. Outside the shield area includes the Main Generator, Air Compressor area, assorted electrical pumps and heat exchangers, and assorted electrical equipment. There are also two isolated Diesel Generator Rooms located on the Ground Floor, plus four enclosed stairwells. There is also a Battery Room on the Ground Floor. There are also two Switchgear Rooms located on the mezzanine level. Each is provided with chilled water supplied Fan Coil Units, and a small amount of supply air from two of the six TB AHUs for pressurization.

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The main generator is air-cooled and does not use hydrogen as the coolant. Therefore, there are no special requirements for hydrogen controls in the area.

The TB is supplied with outside air using six supply AHUs. These AHUs are provided with electric heating coils to provide a reasonable supply air temperature, and upstream prefilters to keep dirt off the coils, and for building cleanliness. The supply air is distributed through ductwork to areas both inside and outside the shield area. Most floors in the TB consists of metal grating, allowing supply air provided to lower elevations to increase in temperature and elevate naturally. Air is exhausted from the building via four AHUs with HEPA filtration located on the TB roof, taking suction from roof penetrations or suction ductwork. Air flow rates to areas have been chosen to effect transfer air from clean areas outside the shield area to potentially contaminated areas inside the shield area. TB supply and exhaust AHUs are provided AFDs to be able to reduce flow as permitted by plant operation and reduced ambient temperature conditions.

The PVS is located towards the north end of the TB roof. It is provided exhaust air from PVS exhaust fans venting the potentially radioactive air up and away from the TB. Each of the PVS exhaust fans takes suction on the Continuous Exhaust Air Plenum, which receives and mixes discharge air from the various building exhaust AHUs. The PVS radiation monitor is addressed in Chapter 11, Subsection 11.5.

The Diesel Generator Rooms are located in the TB. The Diesel Generator rooms are heated using electric unit heaters. The ventilation design of the Diesel Generator Rooms is for air to enter each room through two inlet air dampers connected to upstream ductwork and a common intake louver, providing outside air for both heat removal and for the combustion of diesel fuel. A small part of the exhaust air from the room exits from a wall mounted exhaust fan, located high on the wall. Most of the exhaust air from the rooms exits from the diesel generator as either radiator exhaust, ducted out the room through a wall-mounted louver/damper, or hard piped out the room through combustion piping/silencer.

The TB Battery Room is ventilated and conditioned by two mini-split HVAC systems, one for Train A and one for Train B, each system with an outdoor wall-mounted condenser unit and an indoor wall-mounted AHU, connected by refrigerant piping. Each AHU brings a small amount of filtered, conditioned outside air into the room, maintaining a slight positive pressure in the space. The room is heated by electric unit heaters.

The walls of the TB Battery Room and each of the two DG Rooms are constructed leak-tight relative to the TB to maintain radioactively clean environments.

9A.5.4.3.1 Component Description

Filters for the TB meet the applicable efficiency rating as stated below. ASHRAE Standard 52.2 (Reference 9A.5.4-5) establishes a filter's MERV and establishes a filter's Average Atmospheric Dust Spot Efficiency. The following ASHRAE filter classifications are MERV specified (with Dust Spot Efficiency in parenthesis) below:

Low Efficiency:	MERV 1-4 (Less than 20%)
Medium Efficiency:	MERV 5-12 (As least 20% but less than 80%)
High Efficiency:	MERV 13-16 (Greater than or equal to 80%)

HEPA (High Efficiency Particulate Air) Filters are specified for various building filtration systems. Filters meet the applicable efficiency rating as stated below.

Filters with efficiency greater than MERV 16 by ASHRAE Standard 52.2 (MERV 17-20) (Reference 9A.5.4-5) are usually rated by the dioctylphthalate test method. This test is based on

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the ability of a filter to remove an aerosol consisting of 0.3 micrometer (micron) particles of a test challenge. HEPA filters are extended-medium dry-type filters in a rigid frame, having minimum particle- collection efficiency of 99.97% on 0.3-micron particles which meets ASME AG-1, Code on Nuclear Air and Gas Treatment, Section FC, HEPA Filters (Reference 9A.5.4-6).

HEPA filters are constructed, qualified, and tested per Underwriters Laboratory-586, High Efficiency, Particulate, Air Filter Units (Reference 9A.5.4-7).

Air Handler Units

Each AHU consists of an inlet area, filters (as specified by the system), heating elements (coils), cooling coils (as required) and the respective fans (supply or exhaust). The Air-Cleaning Units and Components are designed in accordance with ASME/ANSI AG-1 Code on Nuclear Air and Gas Treatment (Reference 9A.5.4-8).

Supply and Exhaust Fans

The various building ventilation systems are provided with supply and exhaust fans, which are sometimes incorporated into AHUs. These fans are either centrifugal or axial fans depending on the suitability to the specific system. The fans are designed, manufactured, and supplied in accordance with the applicable standards of AMCA (Air Movement and Control Association International). Fans are equipped with mechanisms to control airflows for the specific system application.

Heating Coils/Elements

Various AHUs are equipped with electrical heating coils/elements. Electric coils are designed and supplied to the requirements of Underwriters Laboratory, Heating and Cooling Equipment (9A.5.4-6).

Cooling Coils

The cooling coils are designed, constructed, and installed in accordance with ASHRAE 33, Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils (Reference 9A.5.4-9), and ANSI/ARI 410 (Reference 9A.5.4-10) and Underwriters Laboratory, (Reference 9A.5.4-11).

Cooling coil condensate is collected in drain pans within the air handler units with the drain pan discharge (condensate) routed to a floor drain located within the room. These floor drains connect to the EFS (Subsection 9A.9.3) subsystem.

9A.5.4.4 Materials

Refer to Subsection 9A.5.1 for information pertaining to HVS materials.

9A.5.4.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.5.4-2 for Turbine Building HVS interfaces with other equipment or systems

9A.5.4.6 System and Equipment Operation

9A.5.4.6.1 Normal Operation

The TB is normally provided heated, filtered once-through supply air from six normally operating AHUs located on the mezzanine level of the TB, which supply through ductwork to areas both inside and outside the radiological area. Air is exhausted from the same spaces through four normally operating exhaust AHUs located on the Turbine Building low bay roof, taking suction either directly through a TB roof penetration, or through suction ductwork connecting to the TB high bay. These exhaust AHUs discharge to the atmosphere from the PVS. AHU cooling provided by ventilation air is supplemented by cooling provided by FCUs located in areas that

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require extra cooling, with FCUs being controlled by local thermostats. AHU heating is supplemented by electric unit heaters. Circulating fans are provided and used to mitigate hot spots within the building.

Exhaust air from potentially contaminated areas including the TB, RB, RWB, and Plant Service Area is filtered by AHUs using HEPA filters before being exhausted to the Continuous Exhaust Air Plenum. The Continuous Exhaust Air Plenum serves as a large mixing box where potentially contaminated air will be mixed and diluted. During normal operation, up to three PVS fans take suction on the Continuous Exhaust Air Plenum and discharge to the nearby PVS. The HEPA filters from the exhaust AHUs at each building assist in ensuring radioactive material entrained in gaseous effluent will not exceed the limits specified in SOR 2000-203 (Reference 9A.5.4-4), for normal operations and anticipated operational occurrences. The PVS fans are provided ASDs to be able to maintain the Continuous Exhaust Air Plenum negatively pressurized relative to the outside and to adjust to varying flow inputs, which may vary based on the operation of the exhaust AHUs at each building.

There are also two Switchgear Rooms. Each is provided with chilled water supplied FCUs, and a small amount of supply air from two of the six TB AHUs for pressurization.

Each enclosed stairwell is provided electric cabinet heaters to maintain space minimum temperature. These heaters are controlled by local thermostat.

Each of the two Diesel Generator Rooms located in the TB are provided two inlet air motor-controlled dampers, each with upstream ductwork leading to a common outside air intake louver. Each room has an exhaust fan discharging to the outdoors, and two electric unit heaters to maintain space design temperatures, controlled by local thermostat. An additional motor-controlled damper is designed to auto-open on Diesel Generator start and auto-close on DG stop with a time delay. This motor-controlled damper is provided in the Diesel Generator radiator cooling discharge ducted flow to the outdoors through a wall louver.

9A.5.4.6.2 Off-Normal Operation

In the event of an SBO, power is lost to all the TB HVAC equipment, Supply and exhaust AHUs and fan, electric unit heaters, FCUs, and all circulating fans.

In the event of a LOOP, with the exceptions of the Diesel Generator Room exhaust fans and motor-controlled dampers, two PVS Fans, and the operating Train A or B Battery Room mini-split cooling system, power is lost to all the TB HVAC equipment; Supply and exhaust AHUs, electric unit heaters, the FCUs, and all circulating fans.

A radiation monitor alarm in the TB initiates closure of the TB isolation dampers and secures the supply and exhaust AHUs. Refer to Chapter 11, Section 11.5 for information pertaining to radiation monitoring.

9A.5.4.7 Instrumentation and Control

The following describe key control features and the operational requirements for the TB HVS:

1. The TB HVS is operated from the Temperature Control Panel located near the Diesel Generator Rooms. TB Temperature Control Panel alarms are forwarded to the Main Control Room Panels.
2. TB supply AHUs are provided with Adjustable Speed Drives to ramp down air flow as outside air temperatures lower, prior to energizing integral electric heating coils.
3. AHU starting is not permitted until all associated isolation dampers are full open.

4. An AHU start request automatically opens all associated isolation dampers.
5. Failure of an operating damper automatically trips its associated AHU.
6. Failure of an operating AHU automatically closes all its associated isolation dampers.
7. To avoid spurious standby starts and fan trips, all airflow logic incorporates time delays to allow fans to generate steady-state flow.
8. TB FCUs are operated from TCPL.
9. TB Battery Room Train A and Train B mini-split systems are operated from TCPL.
10. TB Battery Room Train A and Train B mini-split systems are operated from the TB TCPL or from nearby touchpads.
11. Train B diesel generator room outside air inlet dampers and exhaust fans are operated from TCPL.
12. The Fire Protection System (Section 9A.6) provides a signal to shut down the turbine building AHUs in the event of a fire. Smoke removal is also performed in conjunction with the FPS. Duct mounted smoke detectors provide a smoke detected signal to the FPS local fire protection panels.
13. The supply AHUs are interlocked with the exhaust AHUs to ensure that the exhaust AHUs are running before a supply fan is started.

9A.5.4.8 Monitoring, Inspection, Testing, and Maintenance

The TB HVS provides isolation features to support testing and maintenance.

Air handlers are field tested per ASME N511, In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems (Reference 9A.5.2-12).

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Personnel and lay-down access are provided around instruments to allow adequate space for maintenance purposes.

9A.5.4.9 Radiological Aspects

Chapter 12, Section 12.1 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.5.4.10 Performance and Safety Evaluation

The TB HVS is not credited for mitigation of design basis accidents and not required to be operated during off-normal events.

HEPA filters associated with the TB HVS exhaust AHUs assist in ensuring radioactive material entrained in gaseous effluent does not exceed the limits specified in SOR 2000-203 (Reference 9A.5.3-4), for normal operations and anticipated operational occurrences. The exhaust air from the TB HVS is monitored for radioactivity prior to discharge to the plant vent. Alarms annunciate in the MCR upon detection of high radiation.

9A.5.4.11 References

9A.5.4-1 ANSI/ASHRAE 62.1, "Ventilation for Acceptable Indoor Air Quality."

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- 9A.5.4-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.4-3 10 CFR 20 Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentration for Release to Sewerage."
- 9A.5.4-4 Government of Canada SOR/2020-202, "General Nuclear Safety and Control Regulations."
- 9A.5.4-5 ANSI/ASHRAE 52.2, "Method of Testing General Ventilation Air-Cleaning Devices for Removal Efficiency by Particle Size."
- 9A.5.4-6 ASME AG-1, "Code on Nuclear Air and Gas Treatment," Section FC, "HEPA Filters," American Society of Mechanical Engineers.
- 9A.5.4-7 UL-586, "High Efficiency, Particulate, Air Filter Units," Underwriters Laboratory.
- 9A.5.4-8 ASME AG-1 "Code on Nuclear Air and Gas Treatment," American Society of Mechanical Engineers.
- 9A.5.4-9 ANSI/ASHRAE 33, "Methods of Testing Forced Circulation Air-Cooling and Air-Heating Coils."
- 9A.5.4-10 AHRI Standard 410, "Forced Circulation, Air-Cooling and Air-Heating Coils," Air Conditioning, Heating, and Refrigeration Institute.
- 9A.5.4-11 UL 1995, "Heating and Cooling Equipment," Underwriters Laboratory.
- 9A.5.4-12 ASME N511, "In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems," American Society of Mechanical Engineers.

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Table 9A.5.4-1: Turbine Building, Heating, Ventilation and Air-Condition System Interfaces

Interfacing System	Interface Description	Interface Boundary
Non-Safety Instrumentation and Controls	Non-Safety Instrumentation and Controls provides distributed control and instrumentation data communication networks.	At the HVS equipment
Process Radiation and Environmental Monitoring	The PREMS provides continuous radiation monitoring of the HVS.	At the HVS equipment
Chilled Water Equipment	The CWE provides chilled water inside coils for cooling of the HVAC FCUs.	At the HVS equipment
Non-Safety Electrical Distribution System	Non-Safety Electrical Distribution System provides Non-Safety Category electrical power to the TB HVS Non-Safety Category electrical loads.	At the equipment
Equipment and Floor Drain	Provides drains for condensation off AHU/Air Conditioning cooling coils.	At the air handler or fan coil unit flange
Safety-Category (TBD) Instrumentation and Control System	Safety-Category (TBD) Instrumentation and Control System provides SC3 control to RB supply and exhaust isolation dampers, TB supply and exhaust dampers, and CRE EFUs.	At the TB HVS equipment
Safety-Category (TBD) Electrical Distribution System	Safety-Category (TBD) Electrical Distribution System provides electrical power to SDG Rooms HVS equipment, RB supply and exhaust isolation dampers, TB Supply and exhaust isolation dampers, CRE EFUs.	At the TB HVS equipment
Fire Protection System	Fire Protection System provides start/stop signals to TB HVS to shut down fans.	Fire Protection System contacts to be located within 1 m of the associated temperature control panel
Main Control Room Panels	Main Control Room Panels provide the status, alarms, and indications in the control room that are required for system monitoring.	At the TB HVS equipment

STAIRWELL PRESSURIZATION FAN (1 PER STAIRWELL)

STAIRWELL (TYP 4)

STAIRWELL ELECTRIC CABINET HEATER (1 PER STAIRWELL)

PLANT VENT/STACK

PLANT VENT/STACK FAN (QTY 4)

CONTINUOUS EXHAUST AIR PLENUM

REACTOR BUILDING EXHAUST

REACTOR BUILDING EXHAUST

MTE EXHAUST GLAND SEAL

PLSA EXHAUST

OGS OFFGASSING EXHAUST

LWM EXHAUST

TB EXHAUST AHU (QTY 2)

TB EXHAUST AHU (QTY 2)

MAIN TURBINE MOISTURE SEPARATOR REHEATER FEEDWATER HEATERS POTENTIALLY RADIOACTIVE AREAS

TB SUPPLY AHU (QTY 6)

FAN COIL UNIT (QTY 3)

FAN COIL UNIT (QTY 3)

FAN COIL UNIT (QTY 3)

TURBINE BLDG SWITCHGEAR A

TURBINE BLDG SWITCHGEAR B

TURBINE BLDG INTERIOR MEZZANINE

ELECTRIC UNIT HEATER (QTY 20)

CIRCULATING FAN (QTY 9)

FAN COIL UNIT (QTY 23)

BATTERY ROOM

BATTERY ROOM MINI-SPLIT SYSTEM (QTY 2)

TCPL

TB TCPL & PVS TCPL

DIESEL GENERATOR ROOMS

DIESEL GENERATOR ROOM HEATING & VENTILATION (TYP 2)

TURBINE BUILDING INTERIOR GROUND FLOOR

DG ROOM EXHAUST FAN (1 PER ROOM)

TCPL DG TCPL (1 PER ROOM)

DG ROOM ELECTRIC UNIT HEATER (1 PER ROOM)

TRAIN A TRAIN B

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9A.5.5 Plant Services Area Heating, and Ventilation Systems

The Plant Services Area is supplied by a 100% capacity makeup AHU and a 100% capacity exhaust AHU with HEPA filtration, functioning in a push-pull manner. Clean filtered outside air, heated when needed, is supplied through ductwork and supply registers to all spaces within the PLSAs. The exhaust AHU takes suction on these same spaces providing differential pressures resulting in flows from "clean" to "dirty" areas." AFDs are provided for each of the AHUs, permitting air flow to be ramped down in the wintertime, saving on heating load. Heating is supplemented by electric unit heaters located in most of the PLSA rooms. Baseboard heating units supplement the AHU heating in the Office and Auxiliary RCA Access Control Room. An electric unit heater is located near the exterior door of the hallway addressing the entryway into the North end of the building as well as along the various exterior walls of the rooms without baseboard heating.

Air discharged from the exhaust AHU is directed through ductwork to the Continuous Exhaust Air Plenum, where the air gets mixed with discharges from other building, before being exhausted out the PVS.

During recovery from a fire, smoke is exhausted from the PLSA by operating the supply AHU in 100% outside air mode in conjunction with operation of the exhaust AHU.

9A.5.5.1 System and Equipment Functions

9A.5.5.1.1 Normal Functions (Non-Safety Category)

1. The PLSA HVS provides a controlled environment for personnel comfort and safety and equipment operation. Sufficient outside air is provided to meet the ventilation requirements for acceptable indoor air quality consistent with the applicable requirements of ASHRAE 62.1 (Reference 9A.5.5-1).
2. The PLSA HVS provides a controlled environment for the proper operation and integrity of equipment in the PLSA during normal, startup and shutdown operations.
3. The PLSA HVS provides necessary isolation features to support testing and maintenance.
4. The PLSA HVS minimizes exposure to personnel during inspection and maintenance by locating equipment and instrumentation as far as practical from potential sources of high radiation.
5. The PLSA HVS maintains designated clean areas higher than atmospheric (positive) pressure to minimize the infiltration of air.
6. The PLSA HVS maintains negative pressurization of potentially contaminated areas to control leakage of potentially radioactive effluent to the atmosphere, or other rooms needing to be maintained at negative pressure.
7. The HVS reduces the potential spread of airborne contamination by maintaining airflow from areas of lower potential for contamination to areas of greater potential for contamination.
8. The PLSA HVS is designed such that failure of SCN equipment does not compromise or otherwise damage Safety-Class equipment.
9. The PLSA HVS provides the capability to exhaust smoke, heat, and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire.

9A.5.5.1.2 Normal Functions (Safety-Category)

The PLSA HVS does not perform Safety-Category functions.

9A.5.5.1.3 Off-Normal Functions (Non-Safety-Category)

PLSA HVS continue to operate during off-normal conditions until removed from service either manually or automatically.

9A.5.5.1.4 Off-Normal Functions (Safety-Category)

The PLSA HVS does not perform any Safety-Category functions.

The PLSA HVS is not required to operate during or after a design basis event. The design of the PLSA HVS meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.11.2 (Reference 9A.5.5-2) as related to providing a ventilation system with an appropriate filtration system capable of the following:

1. Preventing unacceptable dispersion of all airborne contaminants within the plant through the use of the exhaust AHUs to maintain a negative pressure in clean areas, resulting in air transferring from clean to "dirty" areas."
2. Reducing the concentration of airborne radioactive substances to levels compatible with the need for access to each particular area is achieved by using shielding, ventilation, monitoring instrumentation and ALARA design concepts as discussed in Chapter 12 to ensure the overall design minimizes radiation exposure to workers and to the public.
3. Keeping the level of airborne radioactive substances in the plant below prescribed limits. The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance is below 0.1 the derived air concentration as specified in 10 CFR 20 Appendix B (Reference 9A.5.2-3) during normal power operation.
4. Application of ALARA design principles in normal operation as discussed in Chapter 12, Subsection 12.1.5.4.
5. Ventilating rooms containing inert or noxious gases without impairing the capability to control radioactive releases using HEPA filters. The HEPA filters associated with the PLSA HVS exhaust AHUs assist in ensuring radioactive material entrained in gaseous effluent does not exceed the limits specified in SOR 2000-203 (Reference 9A.5.3-4), for normal operations and anticipated operational occurrences.

9A.5.5.2 Safety Design Bases

The PLSA HVS does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

9A.5.5.3 Description

Figure 9A.5.5-1 depicts the PLSA HVAC Process Flow Diagram.

The PLSA are provided filtered, heated air from a normally operating supply AHU located on the Plant Services Area roof, supplying through ductwork. Air is exhausted from the spaces by a normally operating exhaust AHU with HEPA filtration also located on the roof, taking suction through ductwork, discharging ultimately to the outdoors through the PVS. Heating is supplemented by electric unit heaters and baseboard heaters.

The PLSA Temperature Control Panel is provided in the Plant Services Area Office and is used to control the supply and exhaust AHU.

9A.5.5.3.1 Component Description

Air Handling Units

Each AHU consists of an inlet area, filters, heating elements (coils), cooling coils (as required) and the respective fans (supply or exhaust). The Air-Cleaning Units and Components are designed in accordance with ASME/ANSI AG-1 Code on Nuclear Air and Gas Treatment (Reference 9A.5.5-5).

Supply and Exhaust Fans

The various building ventilation systems are provided with supply and exhaust fans, which are sometimes incorporated into AHUs. These fans are either centrifugal or axial fans depending on the suitability to the specific system. The fans are designed, manufactured, and supplied in accordance with the standards of AMCA. Fans are equipped with ASD mechanisms as required to control airflows for the specific system application.

Heating Coils/Elements

Various AHUs are equipped with electrical heating coils/elements. Electric coils are designed and supplied to the requirements of Underwriters Laboratory, Heating and Cooling Equipment (Reference 9A.5.5-6).

9A.5.5.4 Materials

Refer to Subsection 9A.5.1 for information pertaining to HVS materials.

9A.5.5.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.5.5-1 for PLSA HVS interfaces with other equipment or systems.

9A.5.5.6 System and Equipment Operation

9A.5.5.6.1 Normal Operation

The PLSA is provided filtered, heated air from a normally operating supply AHU located on the PLSA roof, supplying through ductwork. Air is exhausted from the space by a normally operating exhaust AHU also located on the roof, taking suction through ductwork, discharging ultimately to the outdoors through the PVS. Heating in each of the ventilated rooms is supplemented by electric unit heaters or baseboard heaters located in each room. The PSA supply AHU and exhaust AHUs are controlled at the Temperature Control Panel in the PLSA Office. The unit heaters are all controlled by local thermostats.

9A.5.5.6.2 Off-Normal Operation

In the event of a LOOP or SBO, power is lost to the Plant Services Area HVAC equipment including the supply and exhaust AHUs, electric unit heaters, and baseboard heaters.

9A.5.5.7 Instrumentation and Control

Manual initiation and shutdown of the PLSA AHUs is provided from the PLSA temperature control panel located in the Plant Services Area Office. System and component operating status for the AHUs is displayed in the MCR:

1. Manual initiation and shutdown of the Plant Services Area AHUs is provided from the Plant Services Area Temperature Control Panel located in the Plant Services Area Office.

2. An AHU start request automatically opens all associated isolation dampers.
3. Failure of an operating damper automatically trips its associated AHU.
4. Failure of an operating AHU automatically closes all its associated isolation dampers.
5. To avoid spurious standby starts and fan trips, all airflow logic incorporates time delays to allow fans to generate steady-state flow.
6. Electric Unit Heaters are controlled from local wall-mounted thermostats.
7. The FPS provides a signal to shut down the Plant Services Area AHUs in the event of a fire. Duct mounted smoke detectors provide a smoke detection signal to the Fire Protection System local fire protection panels.

9A.5.5.8 Monitoring, Inspection, Testing, and Maintenance

The PLSA HVS provides necessary isolation features to support testing and maintenance

Air handlers are field tested per ASME N511, In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems (Reference 9A.5.5-7).

Component specific maintenance procedures are outlined in the vendor manuals provided as part of each procurement package.

Personnel and lay-down access are provided around instruments to allow adequate space for maintenance purposes.

9A.5.5.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.5.5.10 Performance and Safety Evaluation

The PLSA HVS does not perform any Safety-Category functions and therefore requires no nuclear safety evaluation.

The PLSA HVS is not credited for mitigation of design basis accidents and not required to be operated during off-normal events.

HEPA filters associated with the PLSA HVS exhaust AHUs assist in ensuring radioactive material entrained in gaseous effluent does not exceed the limits specified in SOR 2000-203 (Reference 9A.5.3-4), for normal operations and anticipated operational occurrences. The exhaust air from the PLSA HVS is monitored for radioactivity prior to discharge to the plant vent. Alarms annunciate in the MCR upon detection of high radiation.

9A.5.5.11 References

- 9A.5.5-1 ANSI/ASHRAE 62.1, "Ventilation for Acceptable Indoor Air Quality."
- 9A.5.5-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.5-3 10 CFR 20 Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentration for Release to Sewerage."
- 9A.5.5-4 Government of Canada SOR/2020-202, "General Nuclear Safety and Control Regulations."

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- 9A.5.5-5 ASME AG-1, "Code on Nuclear Air and Gas Treatment," American Society of Mechanical Engineers.
- 9A.5.5-6 UL 1995, "Heating and Cooling Equipment," Underwriters Laboratory.
- 9A.5.5-7 ASME N511, "In-Service Testing of Nuclear Air – Treatment, Heating, Ventilating, and Air Conditioning Systems," American Society of Mechanical Engineers.

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Table 9A.5.5-1: Plant Services Area Heating, and Ventilation System Interfaces

Interfacing System	Interface Description	Interface Boundary
Non-Safety Instrumentation and Controls	Non-Safety Instrumentation and Controls provides distributed control and instrumentation data communication networks.	At the HVS equipment
Non-Safety Electrical Distribution System	Non-Safety Electrical Distribution System provides Non-Safety Category electrical power to the HVS Non-Safety Category electrical loads.	At the HVS equipment
Fire Protection System	Fire Protection System provides start/stop signals to TB HVS to shut down fans.	Fire Protection System contacts to be located within 1 m of the associated temperature control panel
Main Control Room Panels	Main Control Room Panels provide the status, alarms, and indications in the control room that are required for system monitoring.	At the HVS equipment

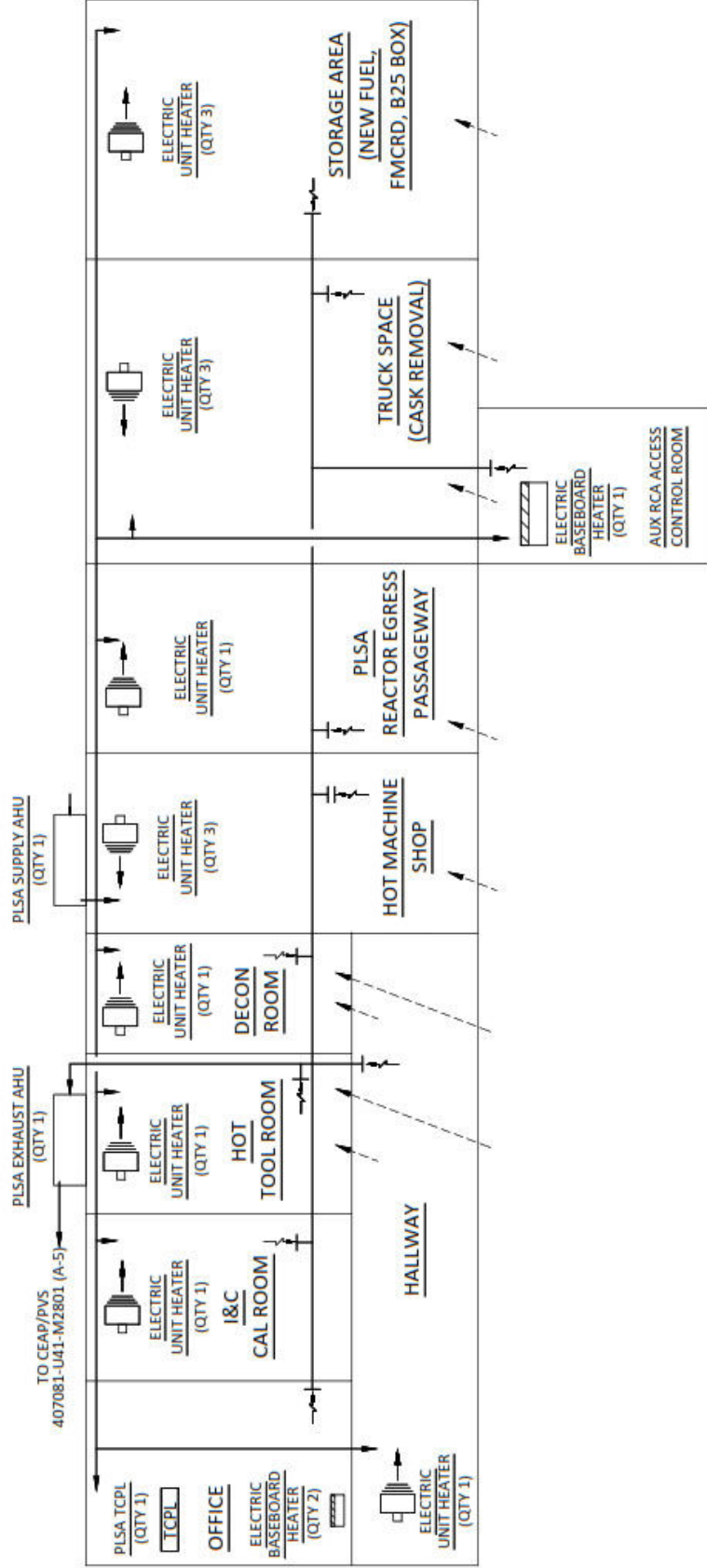


Figure 9A.5.5-1: Plant Services Area HVAC

9A.5.6 Containment Cooling System

The Containment Cooling System is a closed loop cooling system that recirculates air/nitrogen with no outside air introduced into the system except during outages.

9A.5.6.1 System and Equipment Functions

The CCS supports performance of the following functions during operating conditions through the entire operating range, from startup to full load condition to refueling:

9A.5.6.1.1 Normal Functions (Non-Safety-Category)

The CCS performs the following Non-Safety-Category functions during normal operations.

- CCS aids in purging of nitrogen from the containment during shutdown

9A.5.6.1.2 Normal Functions (Safety-Category)

The CCS performs the following Safety-Category functions during normal operations:

- CCS maintains temperature in the upper containment and control rod drive area at the required target temperature and below the specified maximum temperature limit during normal operation
- CCS maintains operation temperature envelope in the Steel-plate Composite Containment Vessel (SCCV) to support Environmental Qualification (EQ) of Safety-Category function related equipment during Modes 1-5

9A.5.6.1.3 Off-Normal Functions (Non-Safety-Category)

The CCS performs the following Non-Safety-Category functions during Off-Normal operations:

- CCS supports heat removal during outages, if available

9A.5.6.1.4 Off-Normal Functions (Safety-Category)

The CCS performs the following SC3 Safety-Category functions during off-normal operations:

- CCS assists with containment cooldown following a Loss-of-Offsite Power (LOOP) during the period from hot shutdown to cold shutdown if available
- CCS limits containment temperature during a LOOP.

9A.5.6.2 Safety Design Bases

The CCS is classified as Defense Line 2 and Safety Class 3 (DL2/SC3). The CCS performs the Safety-Category function of heat removal in the containment.

9A.5.6.3 Description

Figure 9A.5.6-1 depicts the CCS.

The design of the CCS provides cooling to the containment consistent with the requirements of REGDOC 2.5.2 Section 7.10 (Reference 9A.5.6-1).

The CCS is Safety Class 3. The CCS is used to ensure the containment conditions are maintained, which includes the EQ temperatures envelope for Safety-Category function related equipment.

The CCS provides containment cooling using AHU that reject heat to the CWE (Subsection 9A.2.4) during all modes of operation. The CCS is a closed loop recirculating cooling system.

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The CCS is comprised of four (4) 50% capacity AHUs and is divided into two trains of two (2) AHUs. Each train is cooled by a corresponding chilled water train so that SCCV cooling is still possible even with loss of one train of CWE or one train of CCS. Each AHU can be powered from one of the standby diesel generators of the Safety Class 2 and 3 Electrical Distribution System.

During normal operation, only one train of CCS is operating while the other is in standby. The train in standby automatically starts upon loss of the lead train or upon temperature exceeding target. All AHUs can be in service at the same time, if needed.

Each AHU has a condensate collection pan. The condensate collected from the AHUs is drained to a hub location inside of containment, then is piped to the Equipment and Floor Drain System (Subsection 9A.9.3) sump located outside containment where it is discharged.

The CCS air distribution ductwork is constructed of hot-dipped galvanized sheet metal and uses manual balancing dampers to control the distribution of air to the various containment locations.

Backdraft dampers are provided on each fan suction to prevent reverse rotation of a non-operating fan.

The system is classified as Seismic Interaction which requires evaluation to ensure that during a seismic event there are no adverse interactions with the ability of SSCs classified as Seismic A or B to perform their functions. Refer to Chapter 3, Subsection 3.2 for additional information pertaining to Seismic Interaction.

The CCS ductwork and equipment are designed in accordance with ASHRAE (Reference 9A.5.6-2) for the reactor building environment to support the operation and integrity of equipment during normal, startup and shutdown operations.

9A.5.6.3.1 Component Description

Air Handling Units

The CCS is comprised of two trains of AHUs. Each AHU is cooled by the corresponding CWE system train. The AHU cooling coils are to be designed, constructed, and installed in manner that facilitates coil cleaning and supports ease of replacement as required. A manual damper is installed at the AHU suction for isolation during maintenance.

Low efficiency air filter sections are provided upstream of the cooling coil sections to permit installation of temporary filters during testing and outages.

Supply Fans

There are four (4) 50% capacity recirculating supply air fans in CCS. The supply fans are located downstream of their respective AHU coils. A totally enclosed fan cooled motor is provided for the fan.

Cooling Coils

CCS has four (4) cooling coils per train. Supply fans are located downstream of the cooling coils. These coils use CWE chilled water as the heat sink.

Condensate from the cooling coils is routed to the bottom of the containment, then through piping to the EFS sump located outside of containment. All cooling coils are provided with stainless steel condensate drain pans.

Air Filters (Temporary)

Low efficiency air filter sections are provided upstream of the cooling coil sections. Low efficiency air filters are used during construction and post-construction testing of CCS to protect the cooling

coils. After the above test, the air filters are removed since it is not necessary to filter the recirculating air during normal plant operation. If necessary, during a refueling outage, temporary air filters can be installed in the filter section.

Piping, Valves, and Ductwork

CCS piping and valves are classified as follows:

CCS cooling coil condensate drain piping is SCN and designed to meet ASME B31.1 Power Piping requirements.

CCS ductwork and dampers are classified as follows:

CCS ductwork and dampers are designed, installed and inspected according to ASHRAE Fundamentals, SMACNA, HVAC System Duct Design, and SMACNA Duct Construction Standards. The ductwork is seismically supported so that in the event of a failure of CCS Structures, Systems, or Components during a seismic event does not adversely affect the ability of any Seismic Category A or B Structure, System or Component to accomplish its Safety Class function.

9A.5.6.4 Materials

Selection of radiation-resistant materials of construction is included in individual equipment specifications.

Material and equipment selection for CCS components is based on a 60-year plant life, with appropriate provisions for maintenance and replacement. Components which may require periodic replacement and/or maintenance prior to the end of plant life are dampers, damper linkages, AHUs, bearings, and motors.

Aluminum ducting is not allowed inside containment.

Materials are selected in accordance with applicable codes, standards, and industry practice for the design, service, and test conditions and expected ambient conditions.

Materials selected are compatible with the internal process and external environmental conditions during normal, abnormal, accident, and beyond design basis accident conditions as appropriate.

9A.5.6.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.5.6-2 for CCS interfaces with other equipment or systems.

9A.5.6.6 System and Equipment Operation

CCS is designed for continuous operation for the following operating modes.

Normal Operational Concept

CCS normal plant operation includes startup, power operation, hot shutdown, cold shutdown, and refueling. In normal operation mode, one (1) of the two (2) trains operates as lead while the other train is in standby. The lead train is initiated remotely from the Main Control Room. The standby train is automatically started when containment temperature reaches limit or upon tripping of the lead train to maintain the normal operation temperature envelope and support environmental qualification of Safety-Category function related equipment. The assignment for lead and standby train is switched periodically. Both trains can also operate simultaneously.

Off-Normal Operational Concept

The CCS AHUs and supporting chillers are provided with backup power from the diesel generators. The CCS remains operational during a Loss-of-Offsite Power. This supports

maintaining the containment environment from exceeding design conditions. The CCS assists with containment cooldown following a Loss-of-Offsite Power during the period from hot shutdown to cold shutdown.

9A.5.6.7 Instrumentation and Control

CCS provides the instrumentation to control and monitor system operation.

Operation of the CCS is controlled remotely from the Main Control Room. Each AHU has a local disconnect switch to facilitate maintenance. Temperature elements are provided to indicate and alarm on loss of train or temperature over target and energize the standby fan motor. Containment electrical penetrations are used to route the cable from the control elements in the ductwork to the Main Control Panel. There will be no CCS local control panels.

9A.5.6.8 Monitoring, Inspection, Testing, and Maintenance

The cooling coils can be accessed for cleaning. Only water can be used for cleaning. No compressed air is allowed. There is no need for duct cleaning since no outside air is introduced during operation.

Pick points for heavy equipment are provided. The equipment and components of the CCS are designed for inspection and maintenance accessibility. Additionally, fans, motors, coils, filter section, and dampers and damper actuators can be removed for maintenance and repair.

To minimize maintenance (e.g., changing/adjusting belts) and contribute to ALARA goals, the CCS fans are equipped with a quick disconnect on the motor leads and a motor terminal junction box. The fans are provided with full manual control for testing and maintenance. Ductwork is arranged to facilitate AHU maintenance.

To allow testing and balancing, small holes may be drilled on the side of the CCS ductwork to take readings with temporary instrumentation. The holes are capped after the testing and balancing is completed.

AHUs are inspected and maintained according to the manufacturer's requirements.

CCS piping is installed and inspected according to ASME B31.1. CCS ductwork is installed and inspected according to Sheet Metal and Air Conditioning Contractors National Association, Inc.

9A.5.6.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to ALARA design measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.5.6.10 Performance and Safety Evaluation

CCS equipment is classified as Safety Class 3 and seismic classification "Seismic Interaction" which entails an evaluation as discussed in Chapter 3, Subsection 3.2 to ensure that in the event CCS Structures, Systems and Components, fail during a seismic event there are no adverse interactions with the ability of any Seismic Category A or B Structure, System, or Component to accomplish their Safety-Class function.

The CCS is designed with N+1 redundancy for asset protection. In the event of one train failure or shut down for maintenance, the train in standby automatically starts and maintains full operation of the system to protect the EQ equipment inside the SCCV.

9A.5.6.11 References

- 9A.5.6-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.5.6-2 "ASHRAE Handbook– Fundamentals," ASHRAE.

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Table 9A.5.6-1: Containment Cooling System Interfaces

Interfacing System	Interface Description	Interface Boundary
Safety Class 2 and 3 Instrumentation and Control	Provides all Safety Class 3 I&C control, logics, and instrumentation (pressure, flow, temperature), etc.	AHUs and instruments
Chilled Water Equipment System	CCS transfers heat to the CWE	Cooling Coils flanges
Safety Class 2 and 3 Electrical Distribution System	Provides power to AHUs	AHUs and instruments
Steel-plate Composite Containment Vessel	Contains environment provided by CCS, thereby ensuring that the environmental qualification limits of the Safety-Category function related equipment inside of the SCCV are not exceeded	Safety-Category function related equipment
Equipment and Floor Drain System	The condensate collected from the AHUs drain pans is routed to the EFS sump located outside of the containment	Drain pipe

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ABBREVIATIONS:

CCS - CONTAINMENT COOLING SYSTEM
CWE - CHILLED WATER EQUIPMENT SYSTEM
EFS - EQUIPMENT AND FLOOR DRAIN SYSTEM
SA - SUPPLY AIR
RA - RETURN AIR

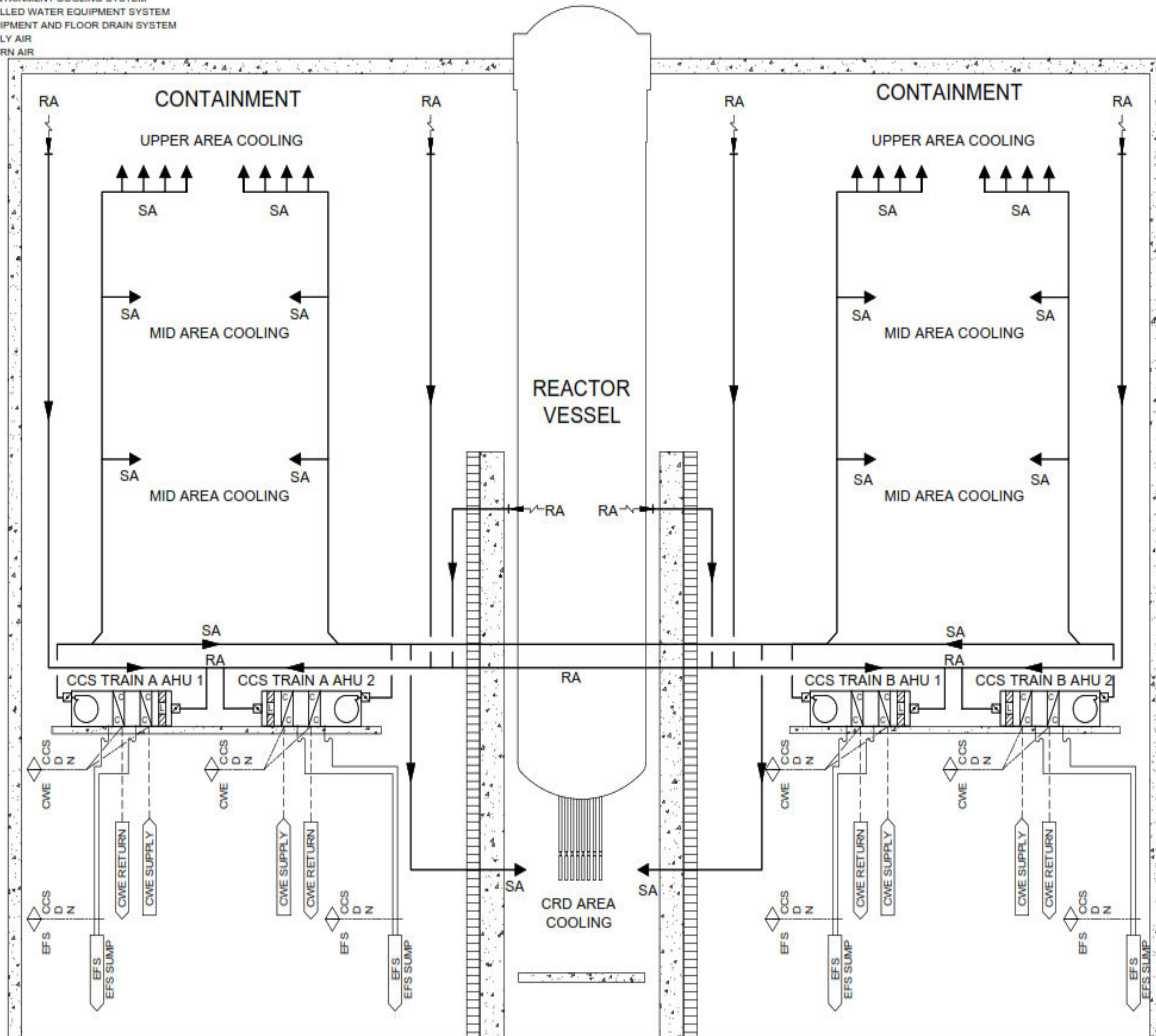


Figure 9A.5.6-1: Containment Cooling System

9A.6 Fire Protection Systems

Introduction

This section provides information on the BWRX-300 fire protection system design. It also confirms that the design meets the Canadian Nuclear Safety Commission (CNSC) expectations for fire protection system design.

The BWRX-300 fire protection system is comprised of an integrated complex of components and equipment provided for detection, notification, annunciation, and suppression of fires.

This document addresses the requirements of CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants (Reference 9A.6.11-3), by providing a set of comprehensive design requirements and guidance that is risk-informed to align with accepted international codes and practices.

The guidance in this document is consistent with international best practices and the recommendations of the International Atomic Energy Agency and International Commission on Radiological Protection.

This section provides criteria pertaining to the safe design of the BWRX-300 in Canada. Aspects of the BWRX-300 design are considered, and multiple levels of defence are promoted in design considerations.

A summary description of the BWRX-300 fire protection system design is provided below. In addition, more detailed information describing the salient features of the overall BWRX-300 Fire Protection Program (FPP).

This discussion focuses on fire protection design consisting of the following:

- General design approach and strategy for fire protection
- Structural aspects of fire protection
- Description of fire protection measures inside confinement and containment
- Strategy and measures for alerting staff of fire events or conditions that may potentially trigger a fire event
- Strategy and measures for control of fire protection systems
- Human factors considerations in the design for fire protection, and
- Description of fire protection systems interface with other systems, including inter-unit interfaces, where common systems are shared including these systems which have a Safety Class classification.

Fire Protection Program General Information

The BWRX-300 fire protection system is comprised of an integrated complex of components and equipment provided for detection, notification, annunciation, and suppression of fires. The Fire Protection Program is different than the Fire Protection System (FPS) which is a system of mechanical equipment and piping used to provide fire detection and suppression to fight a fire at the plant.

The OPG DNNP Fire Protection Program includes the concepts of design and layout implemented to prevent or mitigate fires, administrative controls and procedures, and the training of personnel to combat fires. Refer to Chapter 3, Subsection 3.4.1 for a discussion of the Fire Protection design

requirements and features for SSCs for the BWRX-300. Additional discussion of the Fire Protection Program is provided in Chapter 19, Section 19.5.

The fire protection program design addresses protection from fire by demonstrating that a Defence-in-Depth (D-in-D) approach has been implemented. The Fire Protection Assessment ensures implementation of the fire protection program through delivery of supporting documents including a comprehensive design report, code compliance review, a fire hazard assessment, and fire safe shutdown analysis.

The PSAR includes results of an independent third-party review of the design, assessing compliance against the applicable fire codes and standards used in the design for protection from fires and explosions. Refer to Section 9A.6.10 for a discussion of the third-party review.

9A.6.1 System and Equipment Functions

The FPS performs the following functions during normal and off-normal conditions.

Normal Functions (Non-Safety Category)

FPS Provides the following safety class functions:

1. Detect and locate fires by fire area or fire zone and actuate alarms promptly (excluding the inerted Primary Containment).
2. Provide automatic and manual fire suppression systems, where required by the Fire Hazard Assessment (FHA) of the BWRX-300 as well as in those areas as defined by other future life safety and fire task analysis.
3. Provide manual backup to automatic fire suppression systems.
4. Limit the spread of fires.
5. Suppress fires which do occur, thereby minimizing the adverse effects of fire on structures, systems, and components important to safety and to production of electricity and economic loss.
6. Ensure that one of the redundant divisions necessary to achieve safe shutdown is free of fire damage.
7. Minimize radioactive or hazardous chemical exposure to personnel and radioactive or hazardous chemical release to the environment as the result of a fire.
8. Provide firewater supply and distribution of firewater to all suppression circuits.

Normal Functions (Safety Category)

The FPS does not perform any safety class functions.

Off-Normal Functions (Non-Safety Category)

The FPS continues to perform its functions in off-normal conditions including a Loss-of-Offsite Power (LOOP). Electronic elements of the system have their own batteries or are electrically supplied from the uninterruptible power supply portion of the electrical system (Refer to Chapter 8).

Off-Normal Functions (Safety Category)

The FPS does not perform any Off-Normal safety category functions.

9A.6.2 Safety Design Bases

Fire Protection System Objective

The BWRX-300 Fire Protection System (FPS) is comprised of an integrated complex of components and equipment provided for detection, notification, annunciation, and suppression of fires.

Safety Design Bases

The FPS does not perform any SC1 or SC2 functions.

Power Generation Design Bases

In accordance with CSA N293, Fire protection for nuclear power plants and N293S1, Supplement No. 1 to N293 (References 9A.6.11-1 and 9A.6.11-7), Fire protection for nuclear power plants (application to small modular reactors) the FPS uses the concept of D-in-D to achieve the required degree of reactor safety by using administrative controls, fire protection systems and features, and safe shutdown capability. These D-in-D principles achieve the following objectives:

- Prevent fires from starting:
 1. And rapidly detect, control, and extinguish promptly those fires that do occur
 2. To provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities does not prevent the safe shutdown of the plant and does not significantly increase the risk of radioactive release to the environment.
- Provide timely identification and suppression of fires:
 1. To provide automatic fire detection and annunciation for selected areas of the plant as required by the FHA for personnel safety and fire brigade notification.
 2. To supply the maximum firewater demand at any point throughout the system, with one fire pump out of service.
 3. To maintain the ability to safely shut down the reactor and keep it shut down by providing the capability to control the spread of and extinguish the postulated fires in all plant areas by the use of fixed and/or portable firefighting equipment. This capability is achievable during all modes of plant operation.
 4. To prevent inadvertent operation of the FPS from jeopardizing the capability to achieve safe shutdown of the plant.
 5. To ensure a continuous firewater supply for the fire pumps in the event of failure of one firewater source. Two separate firewater sources are connected to FPS such that there is no interruption in supply and that failure of one water source or its piping does not drain the other sources.
 6. To provide manual suppression capability to all plant areas, including those that have automatic fire suppression systems.
 7. To ensure at least one effective hose stream can reach any location containing SC1 or SC2 equipment, for preventing a fire exposure hazard to the equipment.
 8. To permit isolation from the fire main or outside hydrants for maintenance or repair without interrupting FPS water supply.

- Ensure that the plant can be safely shutdown in the event of a fire:
 1. To maintain the ability to safely shut down the reactor and keep it shut down by providing adequate separation of SC1 or SC2 equipment. This capability is to be achievable during all modes of plant operation.
 2. To minimize the probability of the spread of fire by the use of fire barriers between areas of significant combustible loading.
 3. To maintain the ability to minimize the potential for radioactive releases to the environment in the event of a fire.
 4. To preclude damage to plant SC1 or SC2 structures, systems, or components caused by seismic loading of the FPS.
 5. To keep equipment required for safe shutdown free from fire damage during a Safe Shutdown Earthquake (SSE). To this end, one source of firewater supply, including a water source, two fire pumps and their associated suction and discharge lines; and firewater lines, including standpipes and hose connections, are designed, and analyzed to remain functional after an SSE. This includes analysis to the first isolation valve on all branches connected to the seismically analyzed firewater lines.
 6. To have a useful life of 60 years with normal maintenance and replacement of parts/components subject to normal wear and deterioration.

Fire Protection System Design Codes and Standards

Mechanical System Codes and Standards utilized for conceptual design are listed in PSAR Chapter 1. A subset of this list addresses relevant FP design codes and standards. The applicable BWRX-300 FP design codes and standards are provided Table 9A.6.2-1.

Chapter 9A, Table 9A.6.2-1 also provides a list of key documents applicable to BWRX-300 FPS design.

A preliminary Code Compliance Review has been performed as part of the FPP development and list the codes and standards required for BWRX-300 FP system design and program development. This document defines code compliance or lists open items, exemptions or alternatives that depart from Canadian codes and standards and how the intent of the requirements is met using equivalent or alternative means.

This Fire Hazard Assessment (FHA) (Reference 9A.6.11-5) establishes and evaluates distinct fire areas for the Reactor Building, Radwaste Building, Turbine Building, Control Building, PLSA and Fire Pump Enclosure for OPG DNNP BWRX-300 plant.

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Table 9A.6.2-1: Applicable BWRX-300 Fire Protection Design Codes and Standards and Key Documents

Code and Standard	Title
American Society of Mechanical Engineers (ASME)	
B31.1	Power Piping
American Society for Testing and Materials	
ASTM D323	Standard Test Method for Vapor Pressure of Petroleum Products (Reid Method)
CSA Group	
CSA B51	Boiler, Pressure Vessel, and Pressure Piping Code
CSA B72	Installation code for lightning protection systems
CSA C22.1 Canadian Electrical Code, Part I	Safety Standard for Electrical Installations
CAN/CSA-C22.2 NO. 0.17	Evaluation of Properties of Polymeric Materials
CSA C22.2 NO. 2556	Wire and cable test methods (Trinational standard with NMX-J-556-ANCE-2021 and UL 2556)
CSA C282	Emergency electrical power supply for buildings
CSA N289.3	Design procedures for seismic qualification of nuclear power plants
CSA N293	Fire protection for nuclear power plants
CSA N293 Supplement 1	Fire protection for nuclear power plants (Application to small modular reactors)
Electric Power Research Institute (EPRI)	
EPRI Product ID 1006756	Technical Report Fire Protection Equipment Surveillance Optimization and Maintenance Guide
EPRI NP-2660	Fire Tests in Ventilated Rooms, Extinguishment of Fire in Grouped Cable Trays,
Factory Mutual	
Factory Mutual 7-101	Fire Protection for Steam Turbines and Electric Generators

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Code and Standard	Title
International Atomic Energy Agency (IAEA)	
International Atomic Energy Agency, IAEA Safety Standards Series, Safety Guide No. NS-G-2.1	Fire Safety in the Operation of Nuclear Power Plants
IAEA, Safety Report Series No. 8	Preparation of Fire Hazard Analysis for Nuclear Power Plants
IAEA, NS-G-1.7	Protection Against Internal Fires and Explosions in the Design of Nuclear Power Plants
INSAG Series No. 10	Defence-in-Depth in Nuclear Safety
INSAG Series No. 12	Basic Safety Principles for Nuclear Power Plants 75-INSAG-3
Safety Reports Series No. 10	Treatment of Internal Fires in Probabilistic Safety Assessment for Nuclear Power Plants
Safety Reports Series No. 46	Assessment of Defence-in-Depth for Nuclear Power Plants
IAEA Safety Standards Series No. NS-G-1.7	Protection Against Internal Fires and Explosions in the Design of Nuclear Power Plants
IAEA Safety Standards Series No. NS-G-2.1	Fire Safety in the Operation of Nuclear Power Plants
National Research Council	
NBCC	National Research Council National Building Code of Canada
NFCC	National Research Council National Fire Code of Canada
Nuclear Energy Institute (NEI)	
NEI 00-01	Guidance for Post Fire Safe Shutdown Circuit Analysis, NEI Circuit Failure Issues Task Force
National Fire Protection Association (NFPA)	
NFPA 1	Uniform Fire Code Handbook
NFPA 10	Standard for Portable Fire Extinguishers
NFPA 11	Standard for Low-, Medium- and High-Expansion Foam Systems
NFPA 13	Standard for the Installation of Sprinkler Systems

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Code and Standard	Title
NFPA 13E	Fire Department Operations in Properties Protected by Sprinkler and Standpipe Systems
NFPA 14	Standard for the Installation of Standpipe and Hose Systems
NFPA 15	Standard for Water Spray Fixed Systems for Fire Protection
NFPA 16	Standard for the Installation of Foam-Water Sprinkler and Foam-Water Spray Systems
NFPA 17	Standard for Dry Chemical Extinguishing Systems
NFPA 20	Standard for the Installation of Stationary Pumps for Fire Protection
NFPA 22	Standard for Water Tanks for Private Fire Protection
NFPA 24	Standard for the Installation of Private Fire Service Mains and their Appurtenances
NFPA 25	Standard for Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems
NFPA 30	Flammable and Combustible Liquids Code
NFPA 30B	Code for Manufacture and Storage of Aerosol Products
NFPA 55	Standard for the Storage, Use, and Handling of Compressed Gases and Cryogenic Fluids in Portable and Stationary Containers, Cylinders, and Tanks
NFPA 69	Standard on Explosion-Prevention Systems
NFPA 72	National Fire Alarm and Signaling Code
NFPA 76	Standard for the Fire Protection of Telecommunications Facilities
NFPA 80A	Recommended Practice for Protection of Buildings from Exterior Fire Exposures
NFPA 90A	Standard for the Installation of Air Conditioning and Ventilating Systems
NFPA 90B	Standard for the Installation of Warm Air-Heating and Air Conditioning Systems
NFPA 91	Standard for Exhaust Systems for Air Conveying of Vapors, Gases, Mists and Noncombustible Particulate Solids

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Code and Standard	Title
NFPA 92	Standard for Smoke-Control Systems
NFPA 92A	Recommended Practice for Smoke-Control Systems
NFPA 101	Life Safety Code
NFPA 241	Standard for Safeguarding Construction, Alteration, and Demolition Operations
NFPA 600	Standard on Industrial Fire Brigades
NFPA 701	Standard Methods of Fire Tests for Flame Propagation of Textiles and Films
NFPA 780	Standard for the Installation of Lightning Protection Systems
NFPA 801	Standard for Fire Protection Practices for Facilities Handling Radioactive Materials
NFPA 804	Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants
NFPA 805	Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants
NFPA 850	Recommended Practice for Fire Protection for Electric Generating Plants and High-Voltage Direct Current Converter Stations
NFPA 855	Standard for the Installation of Stationary Energy Storage Systems
NFPA 921	Guide for Fire and Explosion Investigations
NFPA 1081	Standard for Industrial Fire Brigade Member Professional Qualifications
NFPA 1620	Recommended Practice for Pre-Incident Planning
NFPA 2001	Standard on Clean Agent Fire Extinguishing Systems
National Fire Protection Association (NFPA)	Fire Protection Handbook
SFPE	Society of Fire Protection Engineers Engineering Guide to Performance-based Fire Protection
Nuclear Insurance Pools Forum (NIPF)	International Guidelines for the Fire Protection of Nuclear Power Plants

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Code and Standard	Title
Ontario Code	
Ontario Building Code	Ontario Building Code
Ontario Fire Code	Ontario Fire Code
Underwriters Laboratory (UL)	
UL 586	UL Standard for Safety High Efficiency, Particulate, Air Filter Units
ULC CAN/ULC-S102	Method of Test for Surface Burning Characteristics of Building Materials and Assemblies
ULC CAN/ULC-S102.2	Method of Test for Surface Burning Characteristics of Flooring, Floor Covering and Miscellaneous
ULC CAN/ULC-S107	Method of Fire Tests of Roof Coverings
ULC CAN/ULC-S109	Flame Tests of Flame Resistant Fabrics and Films
ULC CAN/ULC-S111	Standard Methods of Fire Tests for Air Filter Units
ULC CAN/ULC-S114	Standard Method of Test for Determination of Non-Combustibility in Building Materials
ULC CAN/ULC-S115	Standard Methods of Fire Tests for Fire Stop Systems
ULC CAN/ULC-S126	Standard Method of Test for Fire Spread Under Roof-Deck Assemblies
ULC CAN/ULC-S524	Standard for the Installation of Fire Alarm Systems
ULC CAN/ULC-S537	Verification of Fire Alarm Systems
U.S. Nuclear Regulatory Commission (NRC)	
NRC NUREG-1852	Demonstrating the Feasibility and Reliability of Operator Manual Actions in Response to Fire
U.S. NRC, NUREG-0800, Section 9.5.1.1	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR edition - Fire Protection Program
U.S. NRC, Regulatory Guide 1.189	Fire Protection for Operating Nuclear Power Plants

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Table 9A.6.2-2: System Interfaces

Name	Interfacing	Interface Description	Interface Boundary
C20	Safety Class 2 and 3 Instrumentation and Control System	C20 provides the instrumentation signals for the system to the fire panels in the MCR/SCR and in areas where safety class 2 and 3 instrumentation is required.	In the MCR/SCR
C30	Non safety class Distributed Control and Instrumentation System (SCN- DCIS)	C30 receives instrumentation signals from the FPS through the fire panels in the MCR/SCR.	In the MCR/SCR
H11	Main Control Room (MCR) Panels	The MCR Panels provides a remote manual actuation capability for the U43 FPS pumps and valves.	In the MCR
K30	Offgas System	The FPS provides firewater for the manual spray on the charcoal absorbers.	In the Radwaste building at each charcoal absorber
P52	Plant Pneumatics System (PPS)	PPS supplies dry compressed air to the preaction and dry pipe sprinkler systems.	At dry pipe and preaction FPS sprinklers
R20	Safety Class 2 and 3 Electrical Distribution System	R20 provides diesel backed electrical power to FPS equipment.	At the Fire Pump Enclosure
R30	Non-safety class Electrical Distribution System	R30 provides non-safety power to FPS equipment.	At the Fire Pump Enclosure
U31	Load Handling System (LHS)	The FPS supervises fire detection in elevator lobbies, shafts, and machine rooms and sends signals to elevator controllers for recall and to shunt trip breakers for power shunt.	At the Local Fire Control Panels nearest to each elevator
U41	Heating, Ventilation	The Plantwide HVS removes smoke from fire areas. Duct- mounted smoke detectors transmit an alarm signal to the local fire control panel upon detection of smoke. Fire/Smoke dampers are considered U41 SSC.	Plantwide, having multiple interfaces in each building for the U41 HVAC system
U50	Equipment and Floor Drain System (EFS)	EFS collects fire protection water during FPS testing and actuation.	Located in the lowest levels of the turbine, radwaste, and reactor building

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Name	Interfacing	Interface Description	Interface Boundary
Y53	Water, Gas, and Chemical Pads (WGC)	Onsite Fuel Storage provides supply for refilling the diesel pump day tanks.	Later
Y99	BWRX Yard	Provides municipality water supply for the firewater make- up to the firewater storage tanks.	Located at the border of the protected area

9A.6.3 Description

Fire Safety System Design Objectives

The primary objectives of BWRX-300 FPS are to minimize both the probability of occurrence and the consequences of fire. To meet these objectives, the FPP is designed to provide reasonable assurance, through D-in-D, that a fire does not prevent the necessary safe shutdown functions from being performed and that radioactive releases to the environment in the event of a fire is minimized.

Development of Fire Protection Performance Criteria and Assessment Metrics

Nuclear Safety Objectives

In the event of a fire, the plant is capable of:

- Achieving and maintaining the reactor in subcritical conditions
- Achieving and maintaining decay heat removal
- Maintaining the integrity of the fission product boundaries
- Limiting the release of radioactive materials

9A.6.4 Materials

Noncombustible and heat-resistant materials are used wherever practical throughout the unit, particularly in locations such as the containment and control rooms. Fire detection and firefighting systems of appropriate capacity and capability are provided and designed to minimize the adverse effects of fires on SSCs important to safety. The firefighting systems are designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs.

9A.6.5 Interfaces with Other Equipment or Systems

Fire Protection System Interdependencies with Other Systems

Fire Protection interfaces are listed in PSAR Table, System Interfaces (Chapter 9A, Table 9A.6.2-2)

9A.6.6 Fire Protection System and Equipment Operation

Codes, Standards, and Regulatory Guidance

Table 9A.6.2-1 lists the codes, standards and guidelines used in the fire protection FPS design.

System Description

Figure 9A.6.6-1 shows the FPS simplified system diagram for the BWRX-300 Standard Plant facilities. Table 9A.6.2-3 lists the major FPS component design characteristics.

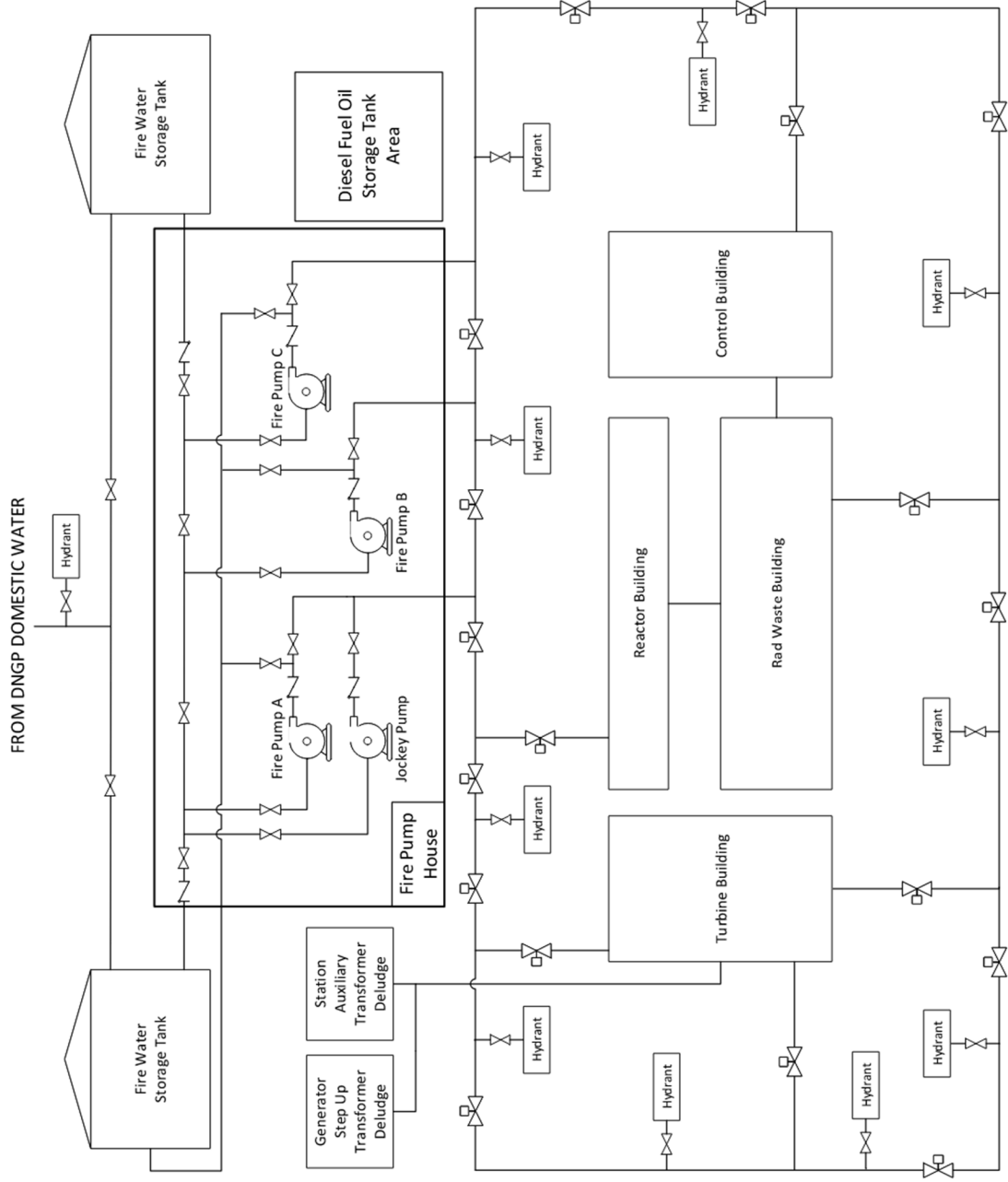


Figure 9A.6.6-1: BWRX-300 Fire Protection System Schematic

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The FPS is the integrated complex of equipment and components that provides early fire detection and suppression to limit the spread of fires. The FPS is part of the overall fire protection program including the plant design and layout to prevent or mitigate fires and includes administrative controls and procedures.

The type of fire suppression is based on the combustible loading and the extent of SC1 or SC2 equipment within a fire area. Use of automatic fire suppression is also determined based on life safety requirements defined in NFPA 101, NFPA 804, CSA N293, CSAN293S1, NBCC or NFCC. Fixed automatic fire suppression systems are installed in areas identified as having a high fire hazard rating by the FHA. Building standpipes and hose stations are provided in major buildings. Portable fire extinguishers are strategically located throughout the plant in accordance with NFPA 10, except in highly radioactive areas.

An automatic fire detection, alarm, supervisory control, and indication system is also provided in selected areas of the plant, as required by the FHA for personnel safety and fire brigade notification.

A Main Fire Alarm Panel (MFAP) located in the Main Control Room (MCR), monitors, and receives system actuation, supervisory, and trouble alarm signals from the individual local panels.

Facility Feature for Fire Protection

Consistent with applicable SC1 or SC2 requirements, structures, systems, and components are designed and located to minimize the probability and effect of fires. To the maximum extent practical, noncombustible, and fire-resistant materials minimize the combustible loading and thereby reduce the expected duration, severity, and intensity of fires.

Exposed structural steel protecting areas containing SC1 or SC2 equipment is fireproofed with material with a fire rating of up to three hours as determined from the FHA.

Access stairwells are enclosed in minimum 2-hour rated firewalls and equipped with self-closing fire-rated doors. Openings in fire barriers or firewalls are equipped with fire doors, frames, and hardware rated the same as the barriers they penetrate.

Seismically supported SC1 or SC2 circuits and circuit routing that contain raceways comply with Branch Technical Position (BTP) SPLB 9.5 1, unless justified under the FHA.

A general intent for BWRX-300 fire protection is to avoid the use of electrical raceway fire barrier systems, relying instead on divisional separation by fire area and structural fire barriers.

Fire Protection Water Supply System

Figure 9A.6.2-1 provides a simplified diagram of the firewater supply piping and supply piping for BWRX-300 Standard Plant facilities.

Water Source

The FPS includes two (2) firewater storage tanks. These tanks are designed in accordance with NFPA 22. These tanks meet NFPA 804/ CSA N293 requirements for the total water demand for a period of 120 minutes. They have a capacity greater than 2,000,000 liters (526,000 gals.) per NFPA 804/ CSA N293.

Physical separation of redundant storage tanks ensures protection against common cause failures such as seismic events and tank rupture. Each tank is equipped with a freeze protection system as necessary based on local climate conditions.

Fire department connections on all major buildings allow a fire department pumper truck to pump water into the FPS as an additional fire protection water supply source.

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Freeze protection is provided for the firewater storage tanks and exposed piping.

Fire Pumps

Three fire pumps each provide 60% of the firewater demand to the worst-case fire within the RB, TB, CB, RWB or Facility Annex Building. Fire pumps are capable of delivering the flow and pressure required to the location that is the most hydraulically remote from the firewater supply source. The fire pumps are located in the yard near the firewater storage tanks. Two of the three fire pumps are required to supply sprinklers and standpipe system. There is a combination of electric and diesel-driven fire pumps. The diesel-driven pump is an air-cooled pump with skid-mounted auxiliaries and a gravity-drain fuel oil supply.

There is a total of three (3) firewater pumps being either electric or diesel. Per CSA N293 (Reference 9A.6.11-1), where more than two pumps are provided, and the largest pump is unavailable for service, then the remaining pumps combined should be capable of providing at least 120% of the necessary, designed flow rate. The designed flow rate is to be based on the scenario of a Turbine fire, which has been determined to be the bounding worst-case fire within the Power Block. Any combination of 2 of the 3 pumps is able to achieve 120% of the needed design flow rate. All fire pumps are capable of delivering the flow and pressure required to the location that has the most limiting combination of distance, line size and/or flow rate. Firewater pumps and accessory components comply with the requirements of NFPA 20. Diesel engines comply with NFPA 20.

An electric fire pump is to be located in the fire-rated firewater pump enclosure (FPE), adjacent to the firewater storage tank and near the power block. The FPE is mounted on a separate foundation. A diesel-driven fire pump is to be located in a fire-rated structure. All pumps are separated by a 3-hour fire-rated fire barrier. Diesel-driven fire pump(s) provide(s) firewater in the event of failure of any electrically driven fire pump or LOOP.

Fire barriers between the fire pumps and their associated equipment as necessary to prevent common cause failures. The diesel-driven fire pump provides firewater in the event of failure of the electrically driven fire pump of the same load group.

The fuel oil day tank for a diesel-driven fire pump is designed in accordance with NFPA 20 and have a capacity sufficient to allow operation of the diesel engine for a minimum of 8 hours before refilling based on fuel consumption at rated pump capacity.

The fuel oil day tank(s) are/is included for the diesel-driven fire pump(s). The fuel oil day tank(s) for the diesel-driven fire pump(s) are/is designed in accordance with NFPA 20 and has a capacity sufficient to allow operation of the diesel engine(s) for a minimum of 8 hours before refilling based on fuel consumption at rated pump capacity. Diesel-driven fire pumps contain chillers that function passively during operation of the pumps.

For a fire within the protected area, an electric pump is designed to start first followed by a second pump, if the first pump fails to start or extra capacity is needed. The third fire pump is designed to start in the event of a failure in either of the other two pumps. Any fire pump may be started manually from the Main Fire Alarm Panel in the control building control room (MCR), from the secondary control room (SCR), or locally. All fire pumps can only be stopped manually.

The associated pressure maintenance (jockey) pump is used to maintain pressure in the system at all times.

Remote signals of fire pump status are provided in the MCR and SCR in accordance with NFPA 20.

The diesel-driven fire pump meets the emissions requirements for the year they are manufactured per NFPA 20.

The non-safety power Distribution System provides power to the motor-driven fire pumps, booster pumps, and jockey pumps. The pumps are powered by the controllers.

Firewater Supply Piping, Yard Piping, Valves, Sprinklers, and Yard Hydrants

Figure 9A.6.6-1 provides a simplified diagram of the firewater supply piping and supply piping for BWRX-300 Standard Plant.

The fire water supply piping consists of a buried yard main loop that feeds supplemental loops in the power block. Check valves are provided between each building connection from the main yard piping loop.

The reactor building piping consists of welded carbon steel piping in accordance with ASME B31.1. The yard main consists of buried Class 200 high-density polyethylene (HDPE) piping or other code compliant pipe material, FM approved for fire main service, in accordance with NFPA 24. Other piping located in buildings consist of carbon steel piping in accordance with NFPA 13, 14 and 15.

The sprinkler systems are designed and installed to meet the requirements of NFPA 13. The deluge systems are designed and installed to meet the requirements of NFPA 15 and/or NFPA 16. Fuel tank foundations and supports meet NFPA 30. Foam injection systems, if installed, are provided to inject foam concentrate into the fire protection water of a foam-water system for distribution over areas protected by foam-water. Foam injection systems are designed in accordance with NFPA 11. Additionally, fixed suppression piping located within the power block is designed and installed in accordance with ASME B31.1. The systems are designed so that an SSE does not cause failures which could impair the functionality of nearby safety class and safe shutdown related structures, systems, and components. Isolation valves which, if closed, would prevent fixed suppression system operation, have tamper switches to monitor their normal position as required by NFPA 13/CSA N293. Isolation valves that control only water supplies to a fire hydrant are post indicating valves and may be locked open in lieu of fire alarm supervision to satisfy CSA N293 (Reference 9A.6.11-1). Fixed suppression systems have provisions to properly drain all parts of the systems. Clean Agent Fire Extinguishing Systems that meet the requirements of NFPA 2001 may be used in addition to the sprinkler-based systems noted below or in areas where sprinkler-based systems are not required per NFPA 804/CSA N293 Fire suppression protection for Canadian BWRX-300 plants meet the provisions given under CSAN293 (Reference 9A.6.11-1).

Standpipes and hose connections in the reactor building are designed to maintain pressure integrity and operability following an SSE in accordance with NFPA 14 and CSA N293. The piping in these areas is designed and installed in accordance with ASME B31.1. Standpipe sizes are at least 150 mm (6 inch) if the building height exceeds 30.5 metres (100 feet). Standpipe sizes for other buildings are at least 100 mm (4 inch) however designed to perform per CSA N293 and NFPA 14 requirements. Drain valves are provided at the bottom of each standpipe to allow for drainage. Fire hoses and accessories are provided in accordance with CSA N293 (Reference 9A.6.11-1) and the FHA. Each hose station has a 65 mm (2-1/2 inch) hose valve with cap rooms and inside entrances for normally occupied rooms.

Fire hydrants located at approximately 75 m (250 ft) intervals along the yard main loop provide firefighting capability, especially in the vicinity of buildings or structures containing combustible materials. The fire hydrants are located no closer than 12.2m (40 ft) from the buildings and structures protected by the hydrants.

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All yard fire main piping is designed and installed to NFPA 24. Fire hydrants are designed and installed to comply with NFPA 24, NFPA 804, and local fire department requirements.

Fire hydrants are protected against freezing and damage from vehicles.

Manual Suppression Means

Manual fire suppression means are provided for all plant areas. The sprinkler systems and the hose station standpipes have separate connections to the firewater main; therefore, no single failure can impair both systems.

Standpipe and Hose Systems

Standpipe and hose stations are provided in all major buildings. Standpipes in areas adjacent to stairways and other locations provide sufficient hose coverage.

The wet standpipes and hose stations are designed to NFPA 14 Class III Service.

Areas containing equipment required for safe shutdown, standpipes and hose connections for manual firefighting remain functional following an SSE. Provisions are made to supply water to at least two standpipes and hose connections for manual firefighting in areas containing equipment required for safe plant shutdown in the event of an SSE. The piping system serving such hose stations is analyzed for SSE loading and is provided with supports to ensure system pressure integrity. The piping and valves for the portion of the hose standpipe system affected by this functional requirement, as a minimum, satisfy ASME B31.1 Requirements.

All rooms within the plant buildings are within the reach of at least one effective hose stream from a Class III hose station unless evaluated as acceptable under the FHA. Effective hose streams from two separate hose stations cover each room that contains equipment required for safe shutdown that is not protected by a fixed fire suppression system. The need for two-hose-station coverage is also based upon the fire hazard present. Rooms not covered by a fixed fire suppression system and with coverage by only one hose station are furnished with portable fire extinguishers for secondary coverage.

Hose stations also provide secondary coverage for fixed suppression systems.

Hose stations are located outside of highly radioactive areas where possible; however, hose stations are located such that any location that contains or could present a hazard to SC1 or SC2 equipment can be reached by at least one effective hose stream 30.5 metres (100 feet) with a maximum of 30.5 metres (100 feet) of hose.

Standpipes and hose stations external to containment and portable extinguishers provide protection during refueling and maintenance operations. Hose stations are located such that any location within containment can be reached by two effective hose streams with a maximum of 30.5 metres (100 feet) of hose.

Fixed fog hose nozzles protecting high-voltage electrical equipment rooms preclude electrical shock hazards with shutoff capability isolation valves. Adjustable fog and straight stream nozzles are provided for all hose stations located away from high-voltage electrical equipment.

Fire Extinguishers

Portable fire extinguishers for manually extinguishing fires are strategically placed throughout the plant in accordance with NFPA 10 and NBCC, except in highly radioactive areas. Portable fire extinguishers are readily available outside the highly radioactive areas, so that they can be accessible when needed.

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Portable Class ABC multipurpose dry chemical-type fire extinguishers are provided for all general hazard areas throughout the buildings in the plant.

Portable Carbon Dioxide Class B, C fire extinguishers are provided for electrical areas.

Special use portable fire extinguishers are provided based upon the hazard present (e.g. – Class D fire extinguishers for a Hot Machine Shop).

Water Spray for Charcoal Filters

Water spray systems are provided for the fire protection of charcoal filters in the HVS, and offgas systems as required. The water is supplied to the filters by means of a fixed piping network extending to a valve station near the exterior of the protected equipment (with manual shutoff valves) and then to open sprinklers or nozzles in the filter enclosure. Drainage is provided for the enclosure.

Fixed Automatic Water Extinguishing Systems

The selection of specific types of fire suppression systems and the areas requiring protection are based on equipment arrangements and combustible loading in each fire area.

Sprinkler piping for areas containing SC1 or SC2 equipment meets the requirements of NFPA 13 (assurance that any failure of FPS piping caused by an earthquake does not damage a SC1 or SC2 item).

Fixed automatic fire suppression systems are provided in all areas identified as having a high fire hazard rating where the combustible loading exceeds 700 MJ/m² for non-electrical areas or 1400 MJ/m² for electrical equipment areas or elsewhere as required by the FHA.

Wet Pipe Sprinkler System

Automatic sprinklers provide protection for the areas identified in the FHA, except where conditions dictate the use of other types of systems or fire suppression agents.

Each wet pipe sprinkler system consists of an alarm check valve, thermally actuated closed-head sprinklers, and attached piping network containing water under pressure. Water discharges immediately from sprinklers opened by heat from a fire. The wet pipe sprinkler system meets the requirements of NFPA 13.

Preaction Sprinkler System (Manual or Automatic)

Preaction sprinkler systems are provided for plant areas where it is necessary to prevent serious water damage to equipment or contents that could occur with other sprinkler systems as a result of leaking automatic sprinkler heads, pipe breaks, or inadvertent actuation. Each preaction sprinkler system consists of a deluge valve, means to contain preaction air (such as a preaction check valve), thermally actuated closed-head sprinklers or spray nozzles, and attached piping containing air under pressure (the air pressure serves to monitor the integrity of the piping and provides a supervisory alarm in the event of trouble). Fire detectors are installed in the same room as the sprinklers.

Deluge System (Manual and/or Automatic)

Deluge systems are provided in high fire hazard areas where required by the system design, such as transformers. Deluge systems apply water or foam-water solutions immediately over the entire hazard. Foam-water solutions may be used for areas that contain Class II combustible liquids (such as fuel oil) or flammable liquids (except diesel-driven fire pumps). Each deluge system consists of a deluge valve, open-head sprinklers or nozzles, and a dry piping network which is connected to the firewater supply. Fire detectors are installed to monitor the protected equipment or be in the same room as the nozzles and activate the system.

Foam System (As Required)

A foam system is provided for concentrated fuel oil or lube oil hazards.

Fire detectors are used for fire detection that actuates preaction foam-water sprinkler or deluge foam-water spray systems to protect against inadvertent actuation.

A foam hose rack is located outside each diesel generator area for manual fire protection. Each foam hose rack consists of a foam playpipe, hose, and rack. Foam hose racks are designed in accordance with NFPA 14.

A foam hose rack is located outside each diesel-driven fire pumps enclosure area for manual fire protection. Each foam hose rack consists of a foam playpipe, hose, and rack. Foam hose racks are designed in accordance with NFPA 14.

A foam hose rack is located outside the turbine lube oil skid for manual fire protection. Each foam hose rack consists of a foam playpipe, hose, and rack. Foam hose racks are designed in accordance with NFPA 14.

Fire Detection and Fire Alarm System

The fire alarm system is to provide a two-stage operation, as follows: (a) first stage — an alert signal, and (b) second stage — an alarm signal. The alert signal is directed to MCR staff and may remain silent throughout the balance of the building to suit the requirements of the plant's emergency notification procedures. On receipt of an alert signal, MCR staff has the capability to immediately provide a voice announcement over the fire alarm system, throughout the protected

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area and external areas of the BWRX-300 plant with the exception of the main and secondary control rooms. The alarm signal is activated automatically in the event that MCR staff do not acknowledge the signal within 5 min of initial fire alarm system activation. The alarm signal is supplemented by voice announcements. There is essentially no delay in the ability to override the alarm signal and operate voice communication functions. This fire alarm and voice communication systems with integrated, supervised, one-way voice communications is provided in all structures and exterior areas within the protected area, as well as structures and areas external to the protected area where SSCs directly support the plant. Fire and voice signals are to be distinctive and not easily confused with other alarm signals. Accessible spaces, with the exception of the main and secondary control rooms, are to be equipped with audible and/or visual fire alarm signal devices. Design, installation, and inspection of the fire alarm system comply with CAN/ULC-S524 and CAN/ULC-S537.

Fire detectors for very early warning and annunciation of fire conditions are separate from fire detection and releasing devices for suppression system actuation. The fire detection system is electrically supervised to detect circuit breaks, ground faults, and power failure.

Deluge and preaction fire detection systems have a 90-hour (minimum) backup battery packs (with 10 minutes of alarm) located at the local fire control panel (releasing panels). The remainder of the fire detection and alarm system has a 24-hour (minimum) backup battery packs (with 5 minutes of alarm) located at each local fire control panel (supervisory panels) and at the MFAP. Immediately following this 24-hour period, emergency battery power under full load is available for not less than 2 hours. Primary power is provided from the plant Uninterruptible AC Power Supply.

Manual fire alarm stations (pull box stations) are located at all exits as required by the NBCC. Where the 60-metre (200-feet) exit rule is used, manual fire alarm stations are located along each main aisle so that the maximum travel distance from the aisle to the manual fire alarm station is not more than 30 metres (100 feet) for areas without sprinklers and not more than 45 metres (150 ft) in areas with sprinklers throughout all normally occupied buildings to meet CSA N293 (Reference 9A.6.11-1). Visible fire notification is provided in manned office areas and other areas of the plant where reasonable.

Smoke detection in ventilation systems is provided to meet NFPA 90A/92A requirements. Smoke detection for elevator equipment is provided to meet ASME A17.1/CSA B44 requirements.

Primary Containment is inerted during normal operation. Portable detection equipment and fire watches, as required, are used inside containment during maintenance outages when the space is not inerted.

Manual fire alarm stations (pull box stations) are provided at the normal exit paths or every 61 metres (200 feet), whichever is less, throughout normally occupied buildings. Manual fire alarm stations (pull box stations) are provided at exit paths only for unoccupied buildings. Visible fire notification is provided in manned office areas such as in the CB and the Plant Service Area.

Fire Barriers

Fire barriers of three-hour fire resistance rating are provided separating:

SC1 or SC2 systems from any potential fires in SC3 areas that could affect the ability of SC1 or SC2 systems to perform their safety function.

Redundant divisions or trains of SC1 or SC2 systems from each other to prevent damage that could adversely affect a safe shutdown function from a single fire.

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Components within a single SC1 or SC2 electrical division that present a fire hazard to components in another SC1 or SC2 division.

Electrical circuits (SC1, SC2 and SC3) whose fire-induced failure could cause a spurious actuation that could adversely affect a safe shutdown function.

Penetrations through fire barriers are sealed or closed to provide fire resistance ratings at least equal to that of the barrier. Only noncombustible materials qualified per CAN/ULC-S102 are used for construction of fire barriers. Fire dampers protect ventilation duct openings in fire barriers as required by CSA N293 (Reference 9A.6.11-1) and CSA N293S1 (Reference 9A.6.11-7).

Design features that prevent or mitigate spurious actuations include:

The FPS is part of the overall plant Fire Protection Program, which ensures the safety of personnel/public, the protection of property/environment, and the continuity of electric power production. In addition to the FPS equipment and components, the Fire Protection Program (FPP) includes combustible inventory control, divisional separation, plant layout and equipment location considerations, fire barriers, training, and administrative controls. The overall Fire Protection Program provides assurance, through defense in depth, that the plant can achieve and maintain a safe shutdown condition in the event of a fire and that radioactive releases to the environment in the event of a fire are minimized.

The Fire Safe Shutdown Analysis (9A.6.11-5) considers multiple hot shorts for safe shutdown equipment that has hard wires for its power or control circuits that could cause equipment to either actuate or not actuate which could result in the equipment not being able to be placed in its safe shutdown position. Hard wires that are in conduit do not have to consider hot shorts from conductors/cables that are outside of the conduit. Hard wires in cabinets/panels that are in wire bundles are considered for hot shorts. CSA N293 / N293S1 separation criteria such as three-hour fire barrier separation between redundant success paths is considered in the safe-shutdown circuit analysis.

The Fire Safe Shutdown Analysis uses a deterministic analysis approach for the safe-shutdown circuit analysis and, therefore, does not use any performance-based approach like fire modeling that considers the location of cables and equipment.

Building Ventilation

Fire protection/smoke control provisions for ventilation for the various building areas are designed as follows:

The Plantwide HVS removes smoke from fire areas. Duct mounted smoke detectors transmit an alarm signal to the local fire control panel upon detection of smoke.

FPS provides fire suppression for the Secondary Control Room (SCR) Makeup AHU filters, annunciation and isolation of Main Control Room (MCR) and Secondary Control Room (SCR) ventilation under fire, smoke purge mode for U41 systems, and interfaces for down-powering ventilation fans from Fire Alarm Panels. The Turbine Building, Reactor Building, Radwaste Building, Control Building and Facility Annex Building are provided with pressurized stairwells to create smoke free egress routes in the event of a fire.

Safe egress and safe smoke refuge areas during a fire incident are provided in accordance with CSA N293 / N293S1 (References 9A.6.11-1 and 9A.6.11-7), NBCC and NFCC guidelines for building occupants and the fire brigade. The Turbine, Reactor, Radwaste Control and Plant Service Area Buildings are provided with pressurized stairwells, in accordance with NFPA 101 and CSA N293 / N293S1 Guidelines are utilized for the design and labeling of safe egress routes.

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Smoke removal meets CSA N293 / N293S1 requirements. Automatic sprinkler protection is provided as directed by the FHA for the high-density cable tunnels, fuel oil tank rooms, diesel generator rooms, and a significant portion of the Turbine Building to limit heat and smoke generation.

Procedures for manual smoke control by manual actions of the fire brigade for all plant areas are in accordance with CSA N293 / N293S1 guidelines.

9A.6.7 Instrumentation and Control

The Fire Protection System (FPS) provides detection, suppression and notification of smoke and fire incidents within the power block and supporting facilities of the BWRX-300.

Each fire suppression system automatically actuated by a fire detection system has the control logic and the capability for manual actuation available at the local fire panel for the protected room. Automatic sprinkler systems which do not require separate detection systems for actuation are generally not equipped with manual actuation means. All instrumentation for fire detection or automatically actuated fire suppression systems is either FM Approved or UL Listed or both.

There are three primary types of FPS instrumentation: instrumentation supporting fire detection, instrumentation supporting automatic suppression systems, and instrumentation supporting firewater delivery.

Instrumentation for the fire detection system provides early detection and warning alarms of fires. Each room containing safety-related equipment has a fire detection system comprised of multiple detectors. Heat and smoke detectors are supervised by local fire panels per NFPA 72/CSA N293 / N293S1 guidelines. The local panels are in turn be connected to the MFAP via a dedicated FPS data link. Signals transmitted include detector status (normal, alarm, supervisory, trouble) as well as local fire panel status. A complete listing of fire detector selection by fire area and/or room number is found in the FHA.

Instrumentation for fixed fire suppression systems provide local and remote monitoring capability for the suppression system status. Alarm check valves are used to indicate that a sprinkler head has opened for wet-pipe sprinkler systems. Pressure/flow sensors are also be used to provide local and remote indication of wet-pipe sprinkler systems. High and low pressure sensors monitor the instrument air pressure in preaction sprinkler systems. High pressure sensors monitor the system pressures downstream of deluge valves. Pressure and level sensors are used for remote and local monitoring of the foam injection systems status. All instruments for automatic suppression systems are wired to the local fire panels for control.

Instrumentation supporting firewater delivery provides status indication of firewater tank level, firewater main pressure, jockey pump status, and main fire pump status conditions.

The MFAP is directly connected to the MCR panels to display common fire alarm, supervisory, and trouble conditions.

The safety class 2 and 3 distributed control and instrumentation system provides HVS system shutdown interfaces from the local fire panels, upon fire detection within HVS ductwork where required by end equipment safety classification.

The non-safety class distributed control and instrumentation system provides HVS system shutdown interfaces from the local fire panels, upon fire detection within HVS ductwork, for equipment not supported by the C20 system.

9A.6.8 Monitoring, Inspection, Testing, and Maintenance

Monitoring

The FPS is monitored to continue to meet Canadian and NFPA requirements during all modes of construction, operation, and decommissioning.

Inspection

The FPS is periodically inspected to continue to meet Canadian and NFPA requirements during all modes of construction, operation, and decommissioning.

Testing and Maintenance

The FPS is maintained as necessary to continue to meet Canadian and NFPA requirements during all modes of construction, operation, and decommissioning.

The following NFPA tests, which are required by code, are performed as a minimum part of the inspection and maintenance requirements for the system:

- a. Waterflow alarm tests of sprinkler systems – NFPA 25
- b. Foam concentration tests, as required – NFPA 25
- c. Deluge waterflow alarm and operational tests – NFPA 25
- d. Foam injection system tests, as required – NFPA 25
- e. Fire detection and alarm system tests – NFPA 72/ CAN/ULC-S537
- f. Smoke removal system/fire damper/smoke dampers – NFPA 92A
- g. Flow testing of fire pumps – NFPA 25

Temporary Configurations

Temporary configurations controlled by plant procedures.

Required Surveillances

There are no Technical Specification surveillances required for fire protection.

Maintenance

The equipment and components of the FPS are designed for easy inspection and maintenance during plant operation. System and equipment manuals are supplied that provide instructions and procedures for installation, operation, and maintenance, and include identification of recommended tools and spare parts.

9A.6.8.1 Post-Maintenance Testing

Post maintenance testing is controlled by plant procedures.

9A.6.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are As Low As Reasonably Achievable (ALARA) in operational states.

9A.6.10 Performance and Safety Evaluation

The Fire Protection System does not perform a Safety Category function. The Fire Protection system is sufficiently separated and/or isolated from Safety Category function SSC to preclude

the possibility of a failure in the FPS from impacting the ability of a Safety Category function SSC from performing its Safety Category function. A failure of the FPS cannot initiate an event more severe than has been considered in the safety analyses and cannot degrade the operation of Safety Category function systems.

Nuclear Safety Performance

Reactor shutdown - Means are provided to rapidly insert negative reactivity into the reactor core to achieve and maintain subcritical conditions and ensure fuel design limits are not exceeded.

Decay heat removal - Means are provided to ensure that fuel is in a safe and stable condition, through the maintenance of sufficient coolant levels and the removal of decay heat from the reactor.

Barrier to fission product release – Means are provided to ensure that nuclear reactor systems that contain radioactive materials or fission products, including the reactor coolant system and reactor auxiliary systems, are not breached. There is no leakage of coolant beyond the capability of the pressure and inventory makeup system.

Support services – Means for supply the necessary power, water, compressed air, and other support functions is provided to ensure that the above criteria are met.

Nuclear Safety Assessment Metrics

1. Protect the operability of the success path to fire safe shutdown for systems and subsystems, including components and cables, of the fire safe shutdown systems that together are sufficient to meet the nuclear safety performance criteria.
2. Meet the criteria under National Building Code of Canada (NBCC) and National Fire code of Canada (NFCC) for fire barrier, detection, suppression systems and maintain criteria in FHA for areas containing radioactive materials.

Life Safety Objectives

The following life safety performance objectives are met during all operational modes and plant configurations:

1. Fire hazard controls are included in design and operational stages
2. Fire notification means are provided
3. Safe egress and/or areas of refuge are provided to occupants for use in the event of a fire
4. A safe environment and other required supports are provided for essential staff so that they can perform all necessary plant control functions during and following a fire
5. Protection for personnel performing emergency services are provided both during and following a fire
6. Access and emergency lighting is provided for all areas where manual firefighting, evacuations, or operator field actions are expected

Life Safety Criteria

The life safety objectives are met using either the prescriptive requirements or performance-based criteria outlined in the NBCC and NFCC.

Life Safety Assessment Metrics

Life Safety Assessment Metrics are met with the criteria under the prescriptive requirements or performance-based criteria outlined in the NBCC and NFCC.

Fire Protection Critical Design Features

Fire protection systems and practices, and design features to prevent the spread of fires in the BWRX-300, are provided for assurance of safe shutdown capability, prevention of radioactive release to the public, personnel safety, investment protection, and plant availability. These design features include all the following:

1. Noncombustible and heat-resistant materials are used wherever practical throughout the unit.
2. Fire barriers that are rated by approved laboratories in hours of resistance to fire and are used to prevent the spread of fire.
3. SSC important to safety are designed and located to minimize the probability and effect of fires and explosions. The concept of compartmentalization uses passive fire barriers to subdivide the plant into separate areas or to confine the effects of fires to a single compartment or area minimizing the potential for adverse effects from fires on redundant SSC important to safety.
4. The BWRX-300 safety assessment demonstrates that an adequate level of safety is achieved by the design and it can be used to satisfy the CNSC safety objectives. This assessment is also used to determine whether region or province specific design changes are needed. One key safety objective established for the BWRX-300 is the implementation of an explicitly defined D-in-D concept to ensure multiple, independent layers of protection against radiation releases outside the bounds of regulatory limits. As such, the BWRX-300 safety assessment includes an approach to deterministic safety analysis involving layered analyses that correspond to levels of defence.
5. Considering the consequences of a fire in a given fire area during the evaluation of the safe shutdown capabilities of the plant, the BWRX-300 design demonstrates that one success path of SSC that can be used to bring the reactor to hot shutdown or hot-standby conditions remains free of fire damage.
6. The BWRX-300 design layout provides adequate means of access to all plant areas for manual fire suppression. The plant layout allows for safe access and egress to areas for personnel performing safe shutdown operations. Considerations include fire and post-fire habitability in safe shutdown areas, protection or separation from fire conditions of access and egress pathways to safe shutdown SSC, and potential restrictions or delays to safe shutdown area access potentially caused by security locking systems.
7. Systems containing flammable or combustible liquids are designed to minimize leakage of these liquids. In locations where an uncontrolled leakage of the liquid could jeopardize fire safe shutdown systems, the design provides devices to collect, divert, and safely contain leakages from pressurized and non-pressurized components in order to prevent ignition or limit the size of fire and achieve fire safe shutdown.
8. Suitable design of the ventilation systems limit the consequences of a fire by preventing the spread of the products of combustion to other fire areas. Means are provided to ventilate, exhaust, or isolate the fire area as required with consideration

given to the consequences of ventilation system failure caused by the fire, resulting in a loss of control for ventilating, exhausting, or isolating a given fire area.

Refer to Chapter 3, Section 3.5 for additional discussion of Fire Protection Design Features for the BWRX-300 SSCs.

Models Used for Design Decisions and Activities

Deterministic Fire Model

The BWRX-300 FPA complies with the FPA described in CSA N293, Fire Protection for NPPs [Reference 9A.6.11-1] and CSA N293S1, Supplement No. 1 to N293, Fire Protection for Nuclear Power Plants (application to small modular reactors) [Reference 9A.6.11-7] and takes a deterministic approach that when combustible material is present, a fire is postulated and damage to a Safety Class SSC assessed. As described in CNSC REGDOC-2.4.1, Deterministic Safety Analysis [Reference 9A.6.11-2], deterministic safety analysis methods can be applied to a wide range of plant operating modes and events, including normal operation and abnormal operation resulting from equipment failure, operator errors and challenges arising from events like fires, floods, or earthquakes.

Probabilistic Fire Model

The BWRX-300 Internal Fires Probabilistic Safety Assessment (IFPSA) is performed using the current information available from the BWRX-300 plant design and procedures, meeting Requirement 4.3 of CNSC REGDOC-2.4.2 [Reference 9A.6.11-4]. Periodic update of the Probabilistic Safety Assessments (PSA) (Requirement 4.4 of REGDOC 2.4.2 [Reference 9A.6.11-4]) will be performed after the plant begins operation.

Probabilistic FP models are also applied to the BWRX-300 design.

These assessments are used in most cases to supplement a deterministic FPA and provide valuable insights into plant design and operation, including the identification of dominant risk contributors, comparison of the options for risk reduction, and consideration of the cost versus risk and benefit analysis.

Fire PSA does not necessarily address compliance with the FP Codes, Standards, and Regulations.

Guidelines for the preparation and regulatory review of PSAs are available in documents such as NFPA 805 and IAEA Safety Report Series No. 10, "Treatment of Internal Fires in Probabilistic Safety Assessment."

Development of Fire Protection Program

The BWRX-300 FPP is developed and implemented in a coordinated manner that considers the various fire protection activities of different engineering disciplines, functional groups, and other organizations.

The BWRX-300 FPP details how the program is implemented, managed, monitored, and modified during each phase of the life cycle of a plant. Activities specified in the FPP for each phase is discussed in PSAR Chapters 3 and 19.

Methodology for Fire Hazard Assessment

The BWRX-300 design utilizes the methodology described in CSA N293 [Reference 9A.6.11-1] for the development of the BWRX-300 Fire Hazard Assessment (FHA). This methodology is illustrated in Figure 9A.6.10-1, BWRX-300 Fire Hazard Assessment Methodology.

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The FHA objective is to identify the specific fire hazards and fire protection capabilities in each area of the plant to demonstrate that potential damage is limited by various active and passive fire protection measures, such that the fire protection goals are achieved.

The FHA addresses all plant building areas and outdoor areas with equipment or storage. The FHA address the goals of nuclear safety, life safety, and impact to the environment. The FHA could also address the goal of minimizing economic loss.

There are specific cases of SSC that are exceptions to the application of CSA N293, classified as two types of exceptions. SSC located outside of the PA and directly support the nuclear facility are included in the scope of CSA N293, and SSCs located inside of the PA that have no nuclear safety concerns and do not directly support the nuclear facility are excluded from the scope of CSA N293.

The content of the BWRX-300 FHA includes the following elements and attributes:

The applicability of CSA and CNSC FP requirements and guidance are evaluated.

1. In situ and potential transient fire and explosion hazards, including amounts, types, configurations, and locations of flammable and combustible materials (e.g., electric cable insulation and jacketing material, lube oil, diesel fuel oil, flammable gases, chemicals, building materials and finishes) associated with operations, maintenance, and refueling activities are identified. The continuity of combustible materials (e.g., exposed electrical cables that span the distance between redundant trains), the potential for fire spread, and sources of ignition are identified and described in the analysis.
2. External exposure hazards (e.g., flammable and combustible liquid or gas storage, adjacent industrial facilities or transportation systems, natural vegetation, and adjacent plant support facilities) that could potentially expose SSC important to safety to damage from the effects (e.g., heat, flame, smoke) of fires are identified.
3. The design, installation, operation, testing, and maintenance of automatic fire detection and suppression capabilities is addressed. The FHA describes the level of automatic protection (e.g., water spray density, gaseous agent concentration) provided relative to the specific fire hazards that are identified.
4. The layout and configurations of SSC important to safety are depicted. The protection for safe shutdown systems within a fire area are determined on the basis of the worst-case fire that is likely to occur and the resulting damage. The FHA explains and documents the extent of such damage. The analysis considers the degree of spatial separation between redundant shutdown systems, the presence of in situ and transient combustibles, the available FP systems and features, sources of ignition, and the susceptibility to fire damage of the cables, equipment, systems, and features in the area that are related to safe shutdown.
5. Reliance on and qualifications of fire barriers, including fire test results, the quality of the materials and barrier system, and the quality of the barrier installation is described.
6. Fire area construction (walls, floor, and ceiling materials, including coatings and thicknesses; fireproofing of structural members; area dimensions and volume; normal ventilation and smoke removal capability; and level of congestion as it applies to access for manual firefighting activities) is described. The FHA provides sufficient information to determine that fire areas are properly selected, based on the fire hazards present and the need for separation of SSC important to safety.

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7. Manual fire suppression capability, including systems (e.g., hydrants, standpipes, extinguishers), fire brigades, manual firefighting equipment, plans and procedures, training, drills, mutual aid, and accessibility of plant areas for manual firefighting are identified. The FHA lists the location and type of manual firefighting equipment and accessibility for manual firefighting.

Potential fire impacts on operations are identified, including the following:

1. Fire in control rooms or other locations where operations important to safety are performed
2. Fire conditions that may necessitate evacuation from areas that are required to be attended for safe shutdown, and
3. Lack of adequate access or smoke removal facilities that impede plant operations or fire extinguishment in plant areas important to safety
4. Potential disabling effects of fire suppression systems on safe shutdown capability are identified, including damage to equipment from the normal or inadvertent operation of fire suppression systems. The FHA addresses the effects of firefighting activities.
5. Explosion-prevention measures in areas subject to potentially explosive environments from flammable gases or other potentially energetic sources (e.g., chemical treatment systems, ion exchange columns, high-voltage electrical equipment) are listed.
6. The availability of oxygen (e.g., inerted containment) is identified
7. Alternative or dedicated shutdown capability for those fire areas where adequate separation of redundant safe shutdown systems cannot be achieved is identified.
8. The analysis assumes fire initiation at the location within each fire area or zone that produces the most severe fire with the potential to adversely affect SSC important to safety. Fire development considers the potential for involvement of other combustibles, both fixed and transient, in the fire area. Where automatic suppression systems are installed, the analysis evaluates the effects of the assumed fire, with and without actuation of the automatic suppression system.

The FHA separately identifies hazards and provides appropriate protection in locations where losses of SSC important to safety can occur as a result of the following:

1. Concentrations of combustible contents, including transient fire hazards of combustibles expected to be used in normal operations, such as refueling, maintenance, and modifications
2. Continuity of combustible contents, furnishings, building materials, or combinations thereof in configurations conducive to fire spread
3. Exposure to fire, heat, smoke, or water, including those that may necessitate evacuation from areas that are required to be staffed for safe shutdown
4. Fire in control rooms or other locations having critical functions important to safety
5. Lack of adequate access or smoke removal facilities that impeded plant operations or fire extinguishment in plant areas important to safety
6. Lack of explosion-prevention measures
7. Loss of electric power or control and instrumentation circuits

8. Inadvertent operation of fire suppression systems

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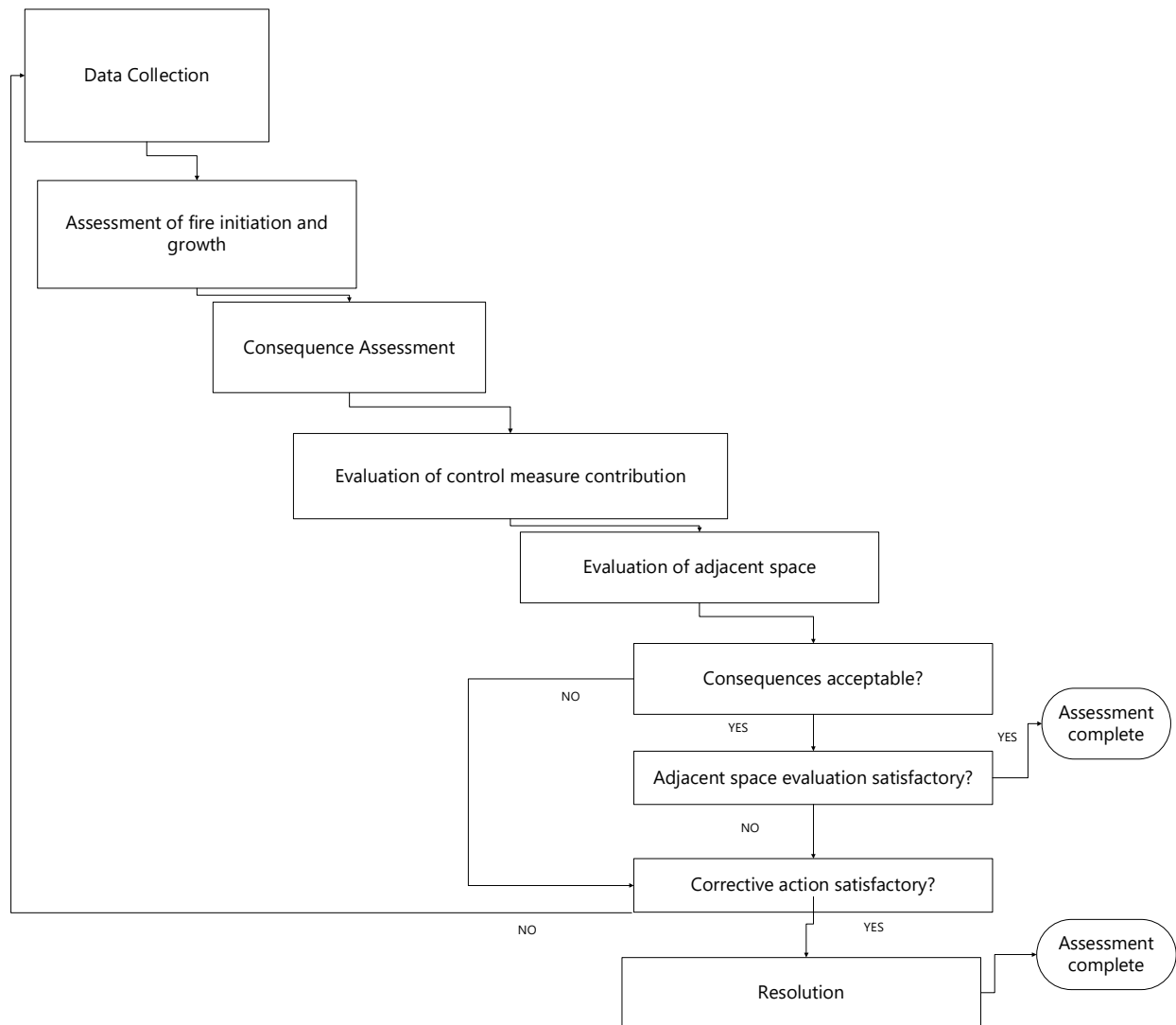


Figure 9A.6.10-1: BWRX-300 Fire Hazard Assessment Methodology

Methodology for Fire Safe Shutdown Analysis

The BWRX-300 design utilizes the methodology described in CSA N293 [Reference 9A.6.11-1] for the development of the BWRX-300 Fire Safe Shutdown Analysis (FSSA) (9A.6.11-5). This methodology is illustrated in Figure 9A.6.10-2, BWRX-300 Fire Safe Shutdown Analysis Methodology. The FSSA process includes the following steps further defined in NEI 00-01.

Identify fire safe shutdown systems:

1. Perform consequence assessment to identify the impact of each design basis fire on fire safe shutdown systems and demonstrate that one success path to fire safe shutdown remains available and radioactive materials listed in FHA do not result in unacceptable radiological exposure,
2. Assess manual actions and emergency procedures step required to achieve fire safe shutdown,
3. Consequence acceptability to determine that remaining SSC and cables provide at least one success path to fire safe shutdown
4. Corrective actions if FSSA can not ensure at least one success path remains available for fire safe shutdown including removal or reduction of fire hazards, enhanced fire protection control measures, relocation or protection of exposed FSS component or addition of alternate components

The FSSA demonstrates that at least one means of achieving nuclear safety objectives and performance criteria is available in the event of a fire. The FSSA scope is limited to plant areas where fires have a potential impact on SSC required to perform the fire safe shutdown functions.

It is a general conclusion of this FSSA (Reference 9A.6.11-5) that the station has the capability to safely shutdown the operating unit in the event of a fire. The results of this conceptual study show that the BWRX-300 plant is inherently safe with respect to internal fire events. This is due in large part to the passive safety features of the BWRX-300 plant design which builds on prior ESBWR and Boiling Water Reactor (BWR) Nuclear Power Plant (NPP) design insights.

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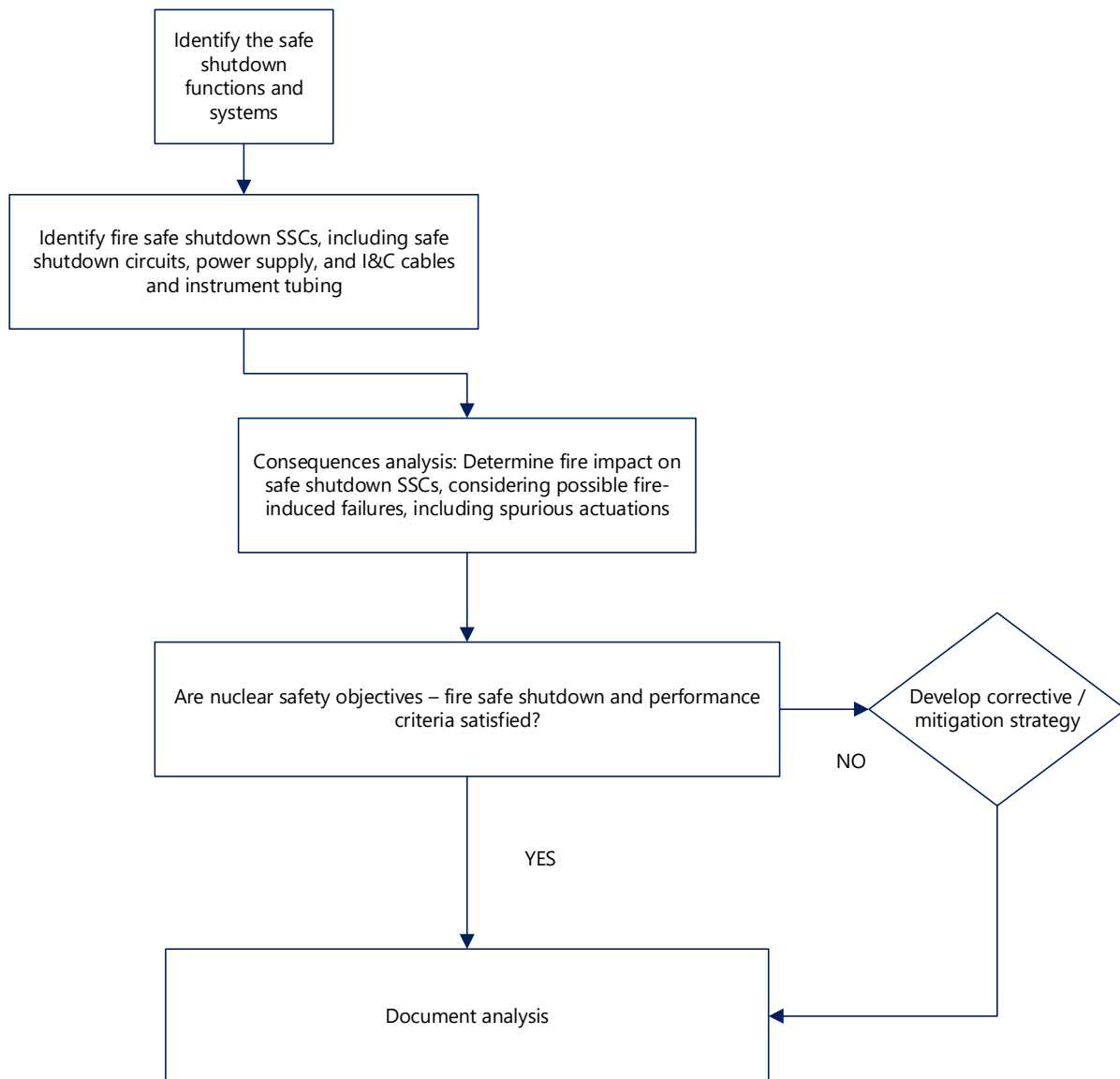


Figure 9A.6.10-2: BWRX-300 Fire Safe Shutdown Analysis Methodology

9A.6.10.1 Results of the Third-Party Review of BWRX-300 Fire Protection System

A third-party review of the BWRX-300 Fire Protection system was conducted by Hatch LTD. (Reference 9A.6.11-9) to ensure that the applicable codes and standards were being applied and that adequate design principles were being utilized as specified by the codes and standards.

Hatch LTD. reviewed the following documents OPG DNNP BWRX-300 Fire Hazard Assessment Requirements Document (Reference 9A.6.11-8), OPG DNNP-1 BWRX-300 Fire Safety Shutdown Requirement and Analysis Document (Reference 9A.6.11-5), and DNNP Preliminary Fire Protection Code Compliance Review Report (Reference 9A.6.11-6). All these documents are currently at the Revision 0 status for Licensing use only.

The Fire Hazards Assessment and Code Compliance Review Report met the requirements described in CSA N293 and CSA N293 Supplement 1 but required more detailed information for final determination of compliance with the Canadian standards. These details will be available as the design of the Fire Protection System and program are completed.

The Fire Safety Shutdown Requirement and Analysis Document complies with Canadian Regulations, Codes and Standards. It follows the methodology in NEI 00-01, Guidance for Post Fire Safe Shutdown Circuit Analysis. The documentation for Fire Safe Shutdown is still being developed, but no issue was identified that could impose a significant hurdle to achieving and maintaining fire safe shutdown.

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9A.6.11 References

- 9A.6.11-1 CSA N293," Fire Protection for Nuclear Power Plants, American National Standards Institute," CSA Group.
- 9A.6.11-2 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 9A.6.11-3 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.6.11-4 CNSC Regulatory Document REGDOC-2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants".
- 9A.6.11-5 NEDC-33978P, "BWRX-300 Darlington New Nuclear Project (DNNP) Fire Safe Shutdown Analysis Report," GE-Hitachi Nuclear Energy Americas, LLC.
- 9A.6.11-6 NEDC-33980P, "BWRX-300 Darlington New Nuclear Project (DNNP) Preliminary Fire Protection Code Compliance Review Report," GE-Hitachi Nuclear Energy Americas, LLC.
- 9A.6.11-7 CSA N293S1, "Supplement #1 to N293-12, Fire Protection for Nuclear Power Plants (Application to Small Modular Reactors)," CSA Group.
- 9A.6.11-8 NEDC-33979P, "BWRX-300 Darlington New Nuclear Project (DNNP) Fire Hazard Assessment Requirements Document," GE-Hitachi Nuclear Energy Americas, LLC.
- 9A.6.11-9 NEDC-33981P, " BWRX-300 DNNP Independent Third-Party Review Report of Preliminary Fire Protection Design," GE-Hitachi Nuclear Energy Americas, LLC.

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Table 9A.6.2-3: FPS Component Design Characteristics

Principal Components	Safety Class	Location	Quality Group	Supply Category	Seismic Category	Notes
U43 Fire Protection System	SCN					
Primary Firewater Pump	SCN	FPH	D	TBD	TBD	
Secondary and Tertiary Firewater Pumps	SCN	FPH	D	TBD	TBD	
Firewater Storage Tanks	SCN	YARD	D	TBD	TBD	
Piping, Valves, and Sprinklers	SCN	CB, SCR, seismically qualified access/egress paths, seismically qualified inst. rooms, RB	D	TBD	TBD	
Piping, Valves, and Sprinklers	SCN	Areas containing SSC	D	TBD	TBD	DL 2 for coverage of SSC
Piping, Valves, and Sprinklers	SCN	All other areas	D	TBD	TBD	
Main Fire Alarm Panel	SCN	CB	D	TBD	TBD	
Secondary Fire Alarm Panel	SCN	RB	D	TBD	TBD	

- (1) Fire pump provided in accordance with NFPA 20 with the FP water pumping system design capable of 120% of total required flow at design pressure.
- (2) As a minimum, the fire protection water pumping system consists of at least one diesel-engine-drive fire pump set and one electric-motor-drive fire pump set.
- (3) The water supply system for fire protection is provided with an automatic pressure maintenance method such as jockey pumps independent of the fire pumps.
- (4) The fire protection water supply volume is calculated based on the largest expected flow rate for a period of 2 hr. The expected flow rate is based on the largest concurrent design demand of any automatic water-based suppression system designed in accordance with CSA N293, taking into account that the allowance for manual hose streams (minimum hose stream demand of 2850 L/min (750gpm)).
- (5) The OPG BWRX-300 FPS design meets the requirements of N293 section 7.3.2 for the water source. This includes redundancy. Where reservoirs or tanks are used, two separate reservoirs or tanks, each having 100% of the supply volume required in Clause 7.3.2.1.2, is provided unless a lesser redundancy is deemed acceptable as demonstrated by the FPA. A SMR should require less fire water than a legacy nuclear plant. It is expected that a lesser volume from CSA N293 section A7.3.2.1 recommending 2,000,000 L (526,000 gal) water storage can be justified in the BWRX-300 FHA.

9A.7 Supporting Systems for Diesel Generators (Storage and Transfer, Cooling Water and Cooling Air, Starting, Lubrications, Combustion Air Intake and Exhaust)

9A.7.1 System and Equipment Functions

The BWRX-300 contains two Safety Class 3 (SC3) Standby Diesel Generators (SDGs). These diesel generators provide SC3 electrical power directly supporting DL2 and DL4a functions. The SDGs backup the SC3 busses which directly power SC1 equipment. Ideally, the SDGs power SC1, SC2, and SC3 loads in the same manner (by providing backup power to the SC3 busses). Whether its normal AC power or diesel AC power, it is the same path from the SC3 busses to SC1 equipment, the SC2 equipment, and the SC3 equipment. Further details are provided in Chapter 8.

The auxiliary systems supporting the SDGs include:

- Generator fuel oil storage and transfer system
- Generator cooling water or cooling air system
- Generator starting system
- Generator lubrication system
- Generator combustion air intake system
- Generator emissions

Except for the fuel oil storage and transfer system, these subsystems are provided with the engine skid and do not require auxiliary plant system connections.

The SC3 standby diesel generators are electric start and radiator cooled and do not require plant mechanical support services for operation (such as plant auxiliary cooling or instrument air). Either of the standby diesel generators can support all the SC2 and 3 EDS loads and SC1 EDS loads (three divisions) that require power for completion of Safety Category 1, 2, and 3 functions (i.e., either standby diesel generator can support active decay heat removal). There is onsite fuel storage in large storage tanks and day tanks for each standby diesel generator. The large storage tanks hold enough fuel for at least seven days of operation of the standby diesel generators at required load. The local day tanks are in addition to the seven-day supply of fuel in order to support online testing; the day tanks alone provide up to eight hours of full power operation at rated load.

Standby diesel generator starting time is not important to plant safety. However, when the associated bus loses power, the standby diesel generator automatically starts after a short delay to allow any fast transfer schemes to work. The standby diesel generator start and achievement of rated voltage and frequency is expected (although not credited) to be accomplished within 30 seconds. Specific loads on the A21 and B21 LV busses (and the derivative MCCs) are automatically sequenced after the standby diesel generator successfully starts. The completion of automatic sequencing is expected (but not credited) to take no longer than two minutes. The standby diesel generators can also be manually started/controlled from the MCR, the SCR, and locally.

9A.7.2 Safety Design Bases

For BWRX-300, the SC1 EDS provides at least 72 hours of battery backed AC and DC power, after which it is supported by the standby diesel generators for at least one week, or alternatively, is supported by connection of portable generators. In a complete loss-of-offsite/generator AC

power, batteries are charged via the standby diesel generators or the portable generator connections while also powering the required loads for safety functions.

Although the BWRX-300 does not require any onsite or offsite AC power for safety, the permanently installed standby diesel generators provide standby AC power and generally meet all requirements of the referenced section of the REGDOC. Restoring power as soon as possible is desirable. Following a plant LOOP, the standby diesel generators are expected (although not credited) to automatically start and achieve rated voltage and frequency within 30 seconds. The completion of automatic sequencing is expected (but not credited) to take no longer than two minutes. The permanently installed batteries provide the emergency power required for the SC1, SC2, and SC3 loads to meet plant needs for 72-hours. If offsite power is not restored, AC power is provided via the standby diesel generators.

The SDGs and their auxiliary systems are SC3, who's primary function is to provide power directly to SC3 loads supporting DL2 functions. The SDGs also provides a backup source of power SC1 and SC2 loads.

9A.7.3 Description

Fuel Storage and Transfer System

A storage tank is provided, which holds sufficient fuel oil to operate the SDGs for a minimum of seven days without refueling. A day storage tank is also provided for fuel oil. Transfer pumps supply fuel oil to each day tank from the fuel oil storage tank. An engine-driven fuel oil booster pump supplies fuel from the day tank to the diesel engine fuel manifold then to the engine fuel injector pumps and injectors.

Generator Cooling Water or Cooling Air System

Each of the SDGs is equipped with an engine radiator cooling system. This system is self-contained on the engine skid-mounted package with no plant auxiliary connections required.

Generator Starting System

Each of the SDGs is equipped with its own dedicated electric start system. This system is self-contained on the engine skid-mounted package with no plant auxiliary connections required. Dedicated batteries, mounted inside, are Lead Acid batteries.

Generator Lubrication System

Each of the SDGs is equipped with a dedicated lubrication system. This system is self-contained on the engine skid-mounted package with no plant auxiliary connections required. All lubricating and fuel handling pumps are API pumps with double seals.

Generator Combustion Air Intake System

Each of the SDGs is equipped with its own air intake system. This system is self-contained on the engine skid-mounted package with no plant auxiliary connections required.

Generator Exhaust System

The Standby Diesel Generator Exhaust system directs diesel exhaust gases away from the SDGs and out of the building. The purpose of this building is to provide protection of the SDGs from ambient environmental effects and to allow the SDGs to perform their functions. The exhaust system design is capable to meet occupational requirements for personnel hearing, emissions, and pollution. Refer to 9A.7.11 on emissions.

9A.7.4 Materials

Materials for the SDGs and fuel system are purchased in accordance with the requirements of Safety Class 3.

9A.7.5 Interfaces with Other Equipment or Systems

The SDGs are package skid mounted systems requiring limited interface with other equipment or systems. The SDGs interfaces with plant protective relaying and the DCIS for protection, control, and instrumentation.

The fuel system interfaces with the plant DCIS for control and monitoring.

9A.7.6 System and Equipment Operation

The SDGs are capable of being manually started and aligned to their respective busses via a local control panel or remotely from the Main and (Secondary) control room. During a loss-of-offsite power, the SDGs automatically start and load on their respective bus via the automatic load sequencer.

Diesel fuel oil is supplied by a delivery system from a 7-day storage tank that is monitored periodically to ensure sufficient fuel oil is available. A supply of lubricating oil is available for the SDGs operating for 7-day operation.

9A.7.7 Instrumentation and Control

Sufficient instrumentation and controls are available locally to start and operate the SDGs. Additionally, control and instrumentation interfaces are provided to the DCIS such that remote control and monitoring from the Main and (Secondary) Control Room is possible.

Indication and alarms for the fuel supply levels in the fuel storage and day tanks are provided.

9A.7.8 Monitoring, Inspection, Testing, and Maintenance

Monitoring, Inspection, Testing, and Maintenance activities are provided to ensure that the SC3 SDGs and fuel system can perform their intended support functions.

The SDGs are also redundant and normally in standby. Any required maintenance can be done online. Specific circumstances of standby diesel generator maintenance are dictated by plant technical specifications and controlled by plant procedures.

9A.7.9 Radiological Aspects

There are no radiological aspects to this system.

9A.7.10 Performance and Safety Evaluation

The SDGs are sized to carry 100% of the required load following an SBO with sufficient fuel to carry the required load for 7 days.

9A.7.11 Diesel Engine Emissions

The diesel engine exhaust configuration is covered by NFPA 20 S 11.5.2. Emission data is supplied by the diesel engine manufacturer. Required emission limits are given by local jurisdictions. Diesel engine emissions are rated with the applicable EPA regulations in the year of supply.

9A.7.12 References

9A.7.12-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

9A.8 Overhead Lifting Equipment

9A.8.1 Cranes, Hoists and Elevators

Cranes, Hoists, and Elevators (CHE) are located throughout the plant and provide the means to lift and lower equipment and materials and move them horizontally along safe load paths. The principal equipment is the Reactor Building Polar Crane, Turbine Building Overhead Bridge Crane and various hoists and monorails. Table 9A.8.1-1 identifies the CHE components.

The Reactor Building Polar Crane is designed to lift heavy loads. A heavy load is defined as a load whose weight is greater than the combined weight of a fuel assembly and the associated handling device. Refer to Chapter 3, Subsection 3.4.4.3 for additional information pertaining to heavy loads.

The CHE provide a safe and effective means for transporting loads including the handling of new and spent fuel, plant equipment, service tools and spent fuel transfer casks and canisters. Safe handling includes design considerations for maintaining occupational radiation exposure as low as practicable during transportation and handling.

9A.8.1.1 System and Equipment Functions

The system and equipment functions associated with the lifting of loads includes the following:

9A.8.1.1.1 Normal Functions (Non-Safety-Category)

The CHE system continuously operates during all modes of normal power plant operation, including startup and shutdown, controlling the movement of lifted loads throughout the plant. The CHE system carries out the following functions:

- System for lifting and lowering a load and moving it horizontally, with the hoisting mechanism being an integral part of the system
- Load handling system includes rigging components such as slings, shackles, and eyebolts which connect a load to a lifting device and any lift fixture
 - The CHE system also includes building elevators for passenger and freight movement.

9A.8.1.1.2 Normal Functions (Safety-Category)

The system does not perform any Safety-Category functions during normal conditions.

9A.8.1.1.3 Off-Normal Functions (Non-Safety-Category)

The CHE equipment is capable of operation during all modes of off-normal power plant operation.

9A.8.1.1.4 Off-Normal Functions (Safety-Category)

The system does not perform any Safety-Category functions during off-normal conditions.

The design of the CHE meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 7.15.3 (Reference 9A.8.1-1) as related to lifting and handling of large and heavy loads and ensuring design margin exists as well as interlocks to accommodate lifting of loads. Information pertaining to the impact of hard objects upon SSCs is presented in Chapter 3, Subsection 3.4.4.3.

9A.8.1.2 Safety Design Bases

The CHE system is classified as SCN. The Reactor Building Polar Crane is designed, fabricated, erected, and tested to appropriate quality standards such that its failure does not impact the function of other Safety-Category function systems.

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The Reactor Building Polar Crane is designed to meet the applicable requirements provided in NUREG-0554 (Reference 9A.8.1-2). Structures, Systems, and Components compliant with NUREG-0554 are considered single failure proof. The Reactor Building Polar Crane is designed to support a critical load regardless of the failure of any single component used in normal operation. A critical load as discussed in NUREG-0554 (Reference 9A.8.1-2) is a load being handled by a crane that can be a direct or indirect cause of release of radioactivity.

The consequences of a postulated load drop are considered acceptable when the four evaluation criteria of NUREG-0612 (Reference 9A.8.1-3), Paragraph 5.1, are satisfied.

Plant arrangement and the design of heavy load handling systems are based on the following criteria:

- To the extent practicable, heavy loads are not carried over or near Safety-Category function components, including irradiated fuel and safe shutdown components. Safe load paths are designated for heavy load handling in Safety-Category function areas.
- The likelihood of a load drop is extremely small (that is, the handling system is single failure proof), or the consequences of a postulated load drop are within acceptable limits.
- Single-failure-proof systems can stop and hold a critical load following the credible failure of a single component.
- Single-failure-proof systems can support a critical load during and after a safe shutdown earthquake.

9A.8.1.3 Description

A description of the CHE system is provided below. Location, safety class and seismic category for system components are summarized in Table 9A.8.1-1.

9A.8.1.3.1 Reactor Building Lifting Devices

Reactor Building Polar Crane

The Reactor Building Polar Crane is designed according to NUREG-0554 (Reference 9A.8.1-2) supplemented by ASME NOG-1 (Reference 9A.8.1-4) for a Type I single failure proof crane. Crane Manufacturers Association of America Specification 70 (Reference 9A.8.1-5), and to the applicable ANSI standards.

The Reactor Building Polar Crane is composed of an overhead bridge of two deep girders supporting a trolley with a main and auxiliary hook. The top of a circular rail is located at approximately elevation 22.7 m. This rail supports the two girders allowing the crane bridge 360 degrees of rotation. The bridge structure spans the full width of the refueling floor. The crane is classified as Seismic Interaction to maintain structural integrity and single failure proof to mitigate the probability of a load drop event. Seismic Interaction evaluations ensure that in the event SSCs, fail during a seismic event there are no adverse interactions with the ability of any Seismic Category A or B SSC to accomplish its Safety Class function. Refer to Chapter 3, Subsection 3.2 for additional information pertaining to Seismic Interaction. Design for both the main and auxiliary hoists is in accordance with ASME NOG-1 requirements which provides a high degree of reliability and safety. Redundancy and other features ensure that the failure of a single component in the load path does not result in the loss of capability to stop and hold a critical load. The principal heavy loads handled by this crane consist of the components associated with reactor vessel refueling, including the reactor head; reactor vessel internals including dryer/separators; fuel assemblies (associated with new fuel receipt), and various containment support components. During spent fuel handling activities, the cranes maximum load is the combination of the spent

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fuel cask, transfer cask, and lifting yoke, which weighs less than the cranes overall capacity. Weight of the spent fuel cask is dependent upon the selected manufacturer. The crane will have access to a truck bay for fuel transfer to spent fuel shipping and/or transfer casks, new fuel assemblies, and replacement components.

Jib Crane

A jib crane is located in the truck bay near the equipment hatch. The jib crane performs rigging services for replacement of components, materials, and supplies to and from the truck bay to the lower levels of the Reactor Building.

Load Handling

Overhead pad eyes and rigging beams are located below the concrete slab at elevation 4.9 m. These components are used to manipulate the piping, valves, and equipment at the top of the reactor vessel in the primary containment during outage maintenance activities.

Battery Room Monorails

A monorail system located above battery racks for battery handling during maintenance activities is provided.

Elevator

A service elevator is provided.

9A.8.1.3.2 Turbine Building Lifting Devices

Turbine Building Overhead Bridge Crane

The Turbine Building Overhead Bridge Crane spans the operating deck above the generator, High Pressure Turbine, Low Pressure Turbines and Condenser. The bridge structure consists of steel girders supporting a trolley with a main hoist and auxiliary hoist. The principal heavy loads handled by this crane consist of components of the turbine-generator system. The heaviest component being the Low Pressure Turbine bladed rotor.

Vertical and horizontal hook travel limits are based on Turbine-Generator equipment requirements as well as the other components located on the operating deck.

CSA B167, (Reference 9A.8.1-6), as well as ASME/ANSI B30.2 (Reference 9A.8.1-7) are used for general construction and installation requirements.

Turbine Building Jib Crane

The Turbine Building Jib Crane is located at the north end of the operating floor. The Turbine Building Jib Crane performs lifts from ground level to the operating deck for maintenance and operation activities such as transport of inspection materials, toolboxes, and other components.

Diesel Generator Monorails

Overhead monorails are located above each Diesel Generator. The monorails are used for engine maintenance activities.

Condenser Monorails

Overhead monorails are located above the circulating water piping elbows directly east of the condenser.

Battery Room Monorails

A monorail system located above battery racks for battery handling during maintenance activities is provided.

Hot Machine Shop Overhead Bridge Crane

An overhead bridge crane is provided in the hot machine shop for maintenance support services.

Elevator

A service elevator is provided.

9A.8.1.3.3 Radwaste Building Lifting Devices

Tank Filter Monorail

A Tank Filter Monorail hoist is located above concrete hatches adjacent to the concrete tank wall on elevation 13m of the Radwaste Building. The Tank Filter Monorail hoist is used to lift concrete hatches above the condensate pre-filter tanks.

Elevator

A service elevator is provided.

9A.8.1.3.4 Control Building Lifting Devices

Battery room monorails in the Control Building are provided for battery handling during maintenance activities.

9A.8.1.3.5 Power Block Pre-Qualified Lift Points

Existing steel or concrete structures throughout the plant have pre-qualified lift points for maintenance activities. The rated load for each lift point is appropriately tagged or stenciled to the building concrete or steel member. Rigging attached to lift points allow for equipment to be moved for repair and/or replacement maintenance activities.

9A.8.1.3.6 Component Description

This subsection describes CHE components.

Hook

The hook is a device for grabbing and lifting loads by means of a device such as a hoist or crane. A lifting hook is typically equipped with a safety latch to prevent the disengagement of the lifting wire rope sling, chain or rope to which the load is attached.

The hook is designed to withstand stress imposed under normal operating conditions while handling loads within the rated load. The minimum hook design factor conforms to those specified for the equipment or system in which the hook is a component.

Monorails

Monorails are used for lifting and lowering a load and moving the load horizontally, suspended from a single track. The combined load on all hoists on the monorail do not exceed the rated load of the monorail.

Monorail systems conform to the minimum design parameters as specified in The Steel Construction Manual (Reference 9.8.1-8), Crane Manufacturers Association of America No. 74 (Reference 9A.8.1-9), ANSI MH27.1 (Reference 9A.8.1-10), or ANSI MH27.2 (Reference 9A.8.1-11), as applicable.

Hoist

A hoist is a suspended machinery unit that is used for lifting or lowering a freely suspended (unguided) load. The hoist and appurtenances are designed to withstand the stresses imposed under normal operating conditions while handling loads within the rated load.

Hand Chain-Operated Chain Hoist

A hand chain-operated chain hoist is a suspended machinery unit that is used for lifting or lowering a freely suspended (unguided) load.

Electric Powered Chain Hoist

An electric powered chain hoist is a suspended machinery unit that is used for lifting or lowering a freely suspended (unguided) load.

Below-the-Hook Lifting Devices

A below-the-hook lifting device is used for attaching loads to a hoist. The device contains components such as slings, hooks, and rigging hardware. The design of below-the-hook-lifting devices are in accordance with ASME BTH-1 (Reference 9A.8.1-12).

9A.8.1.4 Materials

Cranes are fabricated using materials that are designed to meet operating stress limits plus margin. The CHE System equipment and associated structural components are designed to meet the 60-year plant life, with appropriate provisions for maintenance and replacement.

9A.8.1.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.8.1-2 for Cranes, Hoists, and Elevators interfaces with other equipment or systems.

9A.8.1.6 System and Equipment Operation

9A.8.1.6.1 Normal Operations

Operators are trained, qualified, and cognizant of the load handling equipment interlocks and protective devices. Interlocks and protective devices are not overridden or bypassed unless authorized by an approved Work Order. Elevators that serve floor elevations above the first floor that are equipped with an automatic emergency recall feature are equipped with smoke detectors installed in the elevator lobbies on the recall level. In-car emergency switches are provided for in all elevator cars.

9A.8.1.6.2 Off-Normal Operations

Load handling systems are operational unless scheduled for a maintenance.

9A.8.1.7 Instrumentation and Control

9A.8.1.7.1 Instrumentation

The following information does not pertain to manual or portable hoists in the CHE system.

The Reactor Building Polar Crane and Turbine Building Overhead Crane control circuits are arranged so that tripping of an overload relay or limit switch defeats the permissive to all crane controllers. Overload relay or limit switch operation does not interrupt lighting circuits.

The avoidance of two-blocking for the RB Polar Crane is accomplished using single-failure-proof features and does not rely on any action by the operator. The normal hoist limit switch is

supplemented by an independent final hoist limit switch operated by the load block to remove power from the hoist motor and brakes.

The prescribed path for Reactor Building Polar Crane hook travel is enforced by limit switches for hook travel in both vertical and horizontal directions.

9A.8.1.7.2 Control

The following information does not pertain to manual or portable hoists in the CHE system.

The Reactor Building Polar Crane and Turbine Building Overhead Cranes are controlled by a local master control panel on the crane bridge. Primary control of crane motions is directed from a cab located on the crane bridge and remote control with backup pendant control. The cab, remote and pendant controls provide for bridge, trolley, main hoist, hook rotation (cab and radio control only), and auxiliary hoist motions. Pendant controls are back up for cab/remote and are isolated during normal operation by a transfer switch on the pendant. Remote control utilizes lever switch technology.

The Reactor Building Polar Crane electrical system is designed so it is possible for the operator to stop and hold a critical load regardless of the failure of any single component used in normal operation. The operator can stop all motors without a time delay using the emergency stop.

The Reactor Building Polar Crane control system uses a programmable logic controller, housed in a bridge control panel. The programmable logic controller is integral to a restricted zone scheme for the crane which is activated during critical lift evolutions such as movement of the spent fuel cask.

9A.8.1.8 Monitoring, Inspection, Testing, and Maintenance

Elevators are registered in compliance with Technical Standards and Safety Authority.

Surveillances of a load handling system are performed in accordance with plant load handling procedures. The surveillance is a visual check of the overall configuration of the load handling system. Inspections and tests of the load handling system is performed in accordance with Operations and Maintenance Manual requirements as well as load handling procedures.

A maintenance program based on manufacturers' recommendations, integrating proactive, reactive, preventive, and predictive maintenance and operating experience, is established to increase the probability that CHE structures, systems and components function in the required manner over the design life cycle. The program includes procedures which ensure that records are retained, and test and inspection discrepancies are documented and corrected. Any crane, hoist or monorail found in an unsafe operating condition is tagged out and removed from service until repaired. All repairs are made by qualified personnel in accordance with the manufacturers' instructions. Rigging hardware used for load handling systems is maintained in a similar fashion.

Maintenance activities for CHE load handling systems which are either preventive or the result of adverse conditions determined through an inspection program are corrected by adjustment, repair, or replacement of components before continuing the use of the system/component. Operational testing is performed in accordance with plant procedures.

CHE load handling systems which are altered, repaired, and modified are required to be operationally tested for the functions affected by the alteration, repair, or modification, as determined by a qualified person. In addition, load testing of altered, repaired, and modified load handling systems, may be limited to the functions affected by the alteration, repair, or modification, as determined by a qualified person.

9A.8.1.9 Radiological Aspects

For purposes of meeting ALARA guidelines the Cranes, Hoists and Elevators include hoisting/transport mechanisms to handle parts and components in decontamination areas, in the contaminated storage areas and in the active workshops.

The Cranes, Hoists and Elevators use remotely operated electric hoists where possible for tasks/equipment associated with general area dose rates in excess of 1 mSv/hr.

Refer to Chapter 12, Subsection 12.1.5.4 for information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.8.1.10 Performance and Safety Evaluation

The Reactor Building Polar Crane is designed to be single failure proof to improve operational reliability, which is generally implemented by adding design margin, and by increasing material inspection and testing above that found in Crane Manufacturers Association of America Specification 70 (Reference 9A.8.1-5) and other standards invoked by Crane Manufacturers Association of America Specification 70. The RB Polar Crane electrical system is designed so that it is possible for the operator to stop and hold a critical load regardless of the failure of any single component used in normal operation.

Based upon guidance provided in NUREG-0612 (Reference 9A.8.1-3); Safe load paths are defined. Use of load handling procedures, training of crane operators, guidelines on slings and special lifting devices, periodic inspection and maintenance of the crane are provided to ensure safe crane operations.

9A.8.1.11 References

- 9A.8.1-1 Canadian Nuclear Safety Commission REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.8.1-2 USNRC NUREG-0554 "Single-Failure-Proof Cranes for Nuclear Power Plants."
- 9A.8.1-3 USNRC NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."
- 9A.8.1-4 ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," American Society of Mechanical Engineers.
- 9A.8.1-5 Crane Manufacturers Association of America, Specification 70 "Specification for Top Running Bridge and Gantry-Type Multiple Girder Electric Overhead Traveling Cranes."
- 9A.8.1-6 CSA B167, "Overhead Cranes, Gantry Cranes, Monorails, Hoists and Jib Cranes," CSA Group.
- 9A.8.1-7 ASME/ANSI B30.2, "Overhead and Gantry Cranes - Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist," American Society of Mechanical Engineers.
- 9A.8.1-8 American Institute of Steel Construction, "Steel Construction Manual."
- 9A.8.1-9 Crane Manufacturers Association of America, Specification 74, "Specification for Top Running and Under Running Single Girder Electric Traveling Cranes Utilizing Under Running Trolley Hoist."
- 9A.8.1-10 ANSI MH27.1, "Specifications for Patented Track Underhung Cranes and Monorail Systems," American National Standards Institute.

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- 9A.8.1-11 ANSI MH27.2, "Specifications for Enclosed Track Underhung Cranes and Monorail Systems," American National Standards Institute.
- 9A.8.1-12 ASME BTH-1 "Design of Below-the-Hook Lifting Devices," American Society of Mechanical Engineers.

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Table 9A.8.1-1: Cranes, Hoists, and Elevators Components

Principal Components	Safety Class	Building	Seismic Category ⁽¹⁾
RB Polar Crane – Main Hook	SCN	RB	Interaction
RB Polar Crane – Auxiliary Hook	SCN	RB	Interaction
Jib Crane	SCN	RB	Interaction
Monorail - Battery Rooms	SCN	RB	Interaction
Elevator	SCN	RB	Interaction
Overhead Bridge Crane Main Hook	SCN	TB	Non-Seismic
Overhead Bridge Crane Auxiliary Hook	SCN	TB	Non-Seismic
Diesel Generator Monorails	SCN	TB	Non-Seismic
Jib Crane	SCN	TB	Non-Seismic
Monorail - Battery Room	SCN	TB	Non-Seismic
Overhead Bridge Crane Hot Machine Shop	SCN	TB	Non-Seismic
Elevator	SCN	TB	Non-Seismic
Monorail - Condenser	SCN	TB	Non-Seismic
Monorail – Filters	SCN	RWB	Non-Seismic
Elevator	SCN	RWB	Non-Seismic
Monorail - Battery Room	SCN	CB	Non-Seismic

(1) Refer to Chapter 3, Subsection 3.2.3 for information pertaining to seismic classification.

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Table 9A.8.1-2: Cranes, Hoists and Elevators System Interfaces

Interfacing System	Interface Description	Interface Boundary
Non-Safety Electrical Distribution System	Non-Safety Electrical Distribution System provides power	Crane bridge or Elevator control panel
	The Turbine Building Overhead crane and the Reactor Building Polar Crane are supplied with an A or B train electrical power transfer switch	Crane bridge
	Communications to elevators	Elevator car enclosure
Reactor Building Structure	The Reactor Building provides shelter and structural support to the components of the Cranes, Hoists and Elevators System	Runway for crane or monorail, Elevator supporting steel or concrete structure
Turbine Building Structure	The Turbine Building provides shelter and structural support to the components of the Cranes, Hoists and Elevators System	Runway for crane or monorail, Elevator supporting steel or concrete structure
Control Building Structure	The Control Building provides shelter and structural support to the components of the Cranes, Hoists and Elevators System	Runway for monorail
Radwaste Building Structure	The Radwaste Building provides shelter and structural support to the components of the Cranes, Hoists and Elevators System	Runway for monorail, Elevator supporting steel or concrete structure
Fire Protection System	The Fire Protection System provides for elevator emergency functions, i.e., smoke detectors installed in elevator lobbies for auto return	Elevator car enclosure

9A.8.2 Fuel Building Crane

The BWRX-300 does not have a Fuel Building and as such does not have a Fuel Building Crane. New fuel is received onsite and is brought into the Reactor Building (RB) through RB hatch. Spent fuel is removed from the Fuel Pool in a cask. Cask loading and transport activities involve use of the Reactor Building Polar Crane located inside of the Reactor Building. Refer to Subsections 9A.1.1 "New Fuel Storage and Handling System," 9A.1.2 "Fuel Storage and Handling System," and 9A.1.4 "Handling Systems for Fuel Cask Loading," and 9A.8.1 "Cranes, Hoists and Elevators," for additional information pertaining to new and spent fuel handling operations.

9A.9 Miscellaneous Auxiliary Systems

9A.9.1 Communication Systems

The communications systems provide effective intraplant communications and effective plant-to-offsite communications during normal operation, maintenance, transients, fire, and accidents conditions including LOOP.

A typical Communication System consists of the following subsystems:

- Wireless telephone system
- Telephone/page system
- Private branch exchange system
- Sound-powered system
- Emergency offsite communications
- Security communication system

9A.9.1.1 System and Equipment Functions

System and equipment functions associated with the Communication System include the following:

Normal Functions (Non-Safety-Category)

The communications systems provide intraplant and plant-to-offsite communications during normal operation.

Normal Functions (Safety-Category)

The communications systems provide no Normal Safety-Category functions.

Off-Normal Functions (Non-Safety-Category)

The communications systems provide intraplant and plant-to-offsite communications during transients, fire, accidents, off-normal phenomena, and security-related events.

Off-Normal Functions (Safety-Category)

The communications systems provide no Off-Normal Safety-Category functions.

The Communications System is designed considering the requirements of REGDOC-2.5.2 (Reference 9A.9.1-1) Section 7.20 "Escape routes and means of communication" by providing diverse methods of communication within the plant and in the immediate vicinity as well as to offsite agencies. In addition, the Communications Systems are designed to provide secure communications channels to emergency support facilities and offsite response organizations consistent with the requirements of REGDOC-2.5.2 Section 8.10 "Control Facilities."

9A.9.1.2 Safety Design Bases

The Communication System does not perform or ensure any Safety-Category function, and thus has no safety design bases.

9A.9.1.3 Description

The communication system provides the means to conveniently and effectively communicate between various plant locations and with offsite locations during normal, maintenance, transient, fire, and accident conditions under maximum potential noise levels. The communication system

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allows guards and watchmen on duty to maintain continuous communication with personnel in manned alarm stations, and offsite/onsite agencies. This is typically accomplished by either private automatic branch (telephone) exchange or wireless communication system.

The Communication System is designed to support Emergency Preparedness and Response requirements for accident conditions, including notification of personnel and implementation of evacuation procedures. This capability includes communications support to both onsite and offsite emergency centers as well as provisions for onsite and offsite communications system, each with a backup power source. Refer to Chapter 19 for information related to Emergency Preparedness and Response.

The Communication System is designed to support the coordination between plant personnel during accident or incident conditions under maximum potential noise levels. This capability also includes communications support for firefighting, including support of alternative and dedicated shutdown capabilities. Refer to Subsection 9A.6 for information related to Fire Protection.

The Communication Systems are designed to support coordination of security activities both within the plant and with external security and law enforcement organizations.

The Communications Systems are designed to generally recognized codes and standards.

The Communication Systems are separated from one another, such that a failure in one system does not degrade the performance of the other systems.

The Communication System is designed to remain functional during a LOOP.

The Communications Systems use conventional equipment that have a history of successful operation at similar industrial settings.

On-site plant communications systems are backed up by the standby diesel generators during a LOOP. Most of the communications equipment is located in the Control Building and the Communications Room both of which have standby diesel generator backed redundant power panels and Uninterruptible Power Supply (UPS) backed redundant power panels. Portions of the Communication Systems are operated on battery backed 72-hour uninterruptible power supplies.

9A.9.1.3.1 Component Description

Wireless Telephone System

A typical wireless telephone system consists of wireless portable handsets, hands-free type portable headsets, a comprehensive antenna system, and necessary electronics equipment. The wireless telephone system is the primary means of communication for plant operations and maintenance personnel. The telephone-page, private branch exchange telephone, and sound-powered communication systems are for general plant communications and serve as a backup to the wireless system.

A typical wireless telephone system has the ability to dial fixed private branch exchange telephone stations and vice versa. The wireless system has the capability to access the telephone-page system and the capability to access offsite emergency communication links.

Telephone/Page System

A typical telephone/page system consists of handsets, amplifiers, loudspeakers, and associated electronic equipment. The system supports paging and party lines without crosstalk or interference.

Private Branch Exchange System

A typical private branch exchange system provides communication between the system stations, with capability for transferring calls and providing conference calls. The private branch exchange system typically interfaces with the following communication systems:

- The wireless telephone system
- Redundant connectivity to commercial telephone services
- Access to the telephone page system
- Direct extensions from the private branch exchange locations exterior to the plant

The separate, redundant connections between the private branch exchange and commercial telephone service provides communications between the Main Control Room and the headquarters or other facilities in case of a single fault.

Commercial telephone lines are provided by the local area telephone company.

Sound-Powered System

The sound-powered telephone system provides voice service to key locations throughout the plant. This system uses portable sound-power telephones that can plug into local terminal jacks. The system allows uninterrupted communications. The system does not rely on external power supply for operation.

Security Communication

A typical security communication system utilizes the common site private branch exchange, public address, and sound power phone circuits to facilitate a portion of their communications needs. Typically, Security has a dedicated radio system providing continuous communications between each onsite security officer, watchman, or armed response individual and an individual in each continuously manned alarm station. These radio systems are typically powered by the security power system.

Emergency Communications

There is an emergency communications workspace provided in the MCR to accommodate two people. The emergency communications workspace contains space for communications equipment and administrative tasks.

9A.9.1.4 Materials

The Communication System is designed using materials that ensure operability in nuclear and industrial environments.

9A.9.1.5 Interfaces with Other Equipment or Systems

The communication system derives its power from the Electrical Distribution System. Refer to Chapter 8, Subsection 8.1 for information pertaining to the Electrical Distribution System.

9A.9.1.6 System and Equipment Operation

Normal Operation

The Communication Systems operation is checked as part of normal daily usage.

Voice and data communications systems and equipment are provided to support all phases of plant operations and maintenance, including emergency operations.

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Portable, wireless communication capability is provided using equipment typically supported by base stations, antennas, amplifiers, and repeaters. This is typically the means of communication among control room operators, equipment operators, radiation protection, and maintenance technicians for routine and emergency operations, including surveillance tests, startup and shutdown operations, refueling, job coverage, and emergency or accident conditions.

Communication channels accommodate the expected message load including task requirements in critical or emergency situations, plus allow margin for expansion and contingency. The system provides the capability for open channel, or "party line" communication.

The Communications Systems function in all ambient noise level environments. The equipment allows communication from high-noise areas consistent with performing other tasks in those areas. Adequate means are provided to alert personnel to the use of communication equipment.

The design of the Communication Systems takes into account the applicable requirements of NUREG-0700 (Reference 9A.9.1-2).

Off-Normal Operation

The Communication Systems will fail in a predictable manner with failure alarms alerting users to failed or degraded system status. The system is checked for functionality as part of daily usage; therefore, system degradation is self-revealing.

9A.9.1.7 Instrumentation and Control

No special instrumentation is required for the Communication System.

9A.9.1.8 Monitoring, Inspection, Testing, and Maintenance

Communication Systems equipment is designed to operate reliably within the environment in which it is installed including environmental conditions such as temperature, humidity, radiation, and noise. Furthermore, the Communication Systems are designed to operate taking into account placement of barriers such as shield walls. Communication System equipment is accessible to personnel for operation, inspection, maintenance, and testing.

The Communications System is pre-operational tested. The systems described above are conventional and have a history of successful operation at similar plants. These systems are used and maintained routinely to ensure their availability.

The power sources for the plant page/party-line telephone system and the private branch exchange are tested separately during the pre-operational and startup test program. Measurements or tests required to identify long-term deterioration are performed on a periodic basis.

9A.9.1.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.9.1.10 Performance and Safety Evaluation

The Communication Systems do not perform Safety-Category functions. The failure of any Communication System does not adversely affect safe shutdown capability. It is not necessary for plant personnel in Safety-Category function areas of the plant to communicate with the control room to achieve safe shutdown of the plant.

Diverse Non-Safety-Category function power supplies connected to the plant standby diesel generators, power the plant page/party-line telephone, Private automatic branch (telephone) exchange and plant radio systems. Failure of any or all of its components does not affect any Safety-Category functions.

9A.9.1.11 References

9A.9.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

9A.9.2-1 NUREG-0700, "Human-System Interface Design Review Guidelines"

9A.9.2 Lighting and Servicing Power System

The plant Lighting and Servicing Power System includes normal and emergency lighting in addition to providing service power throughout the plant. The normal lighting provides illumination during plant operating, maintenance, and test conditions. The emergency lighting provides illumination in areas where emergency operations are performed upon loss of normal lighting. Security aspects of the Lighting and Servicing Power System are considered Security Related Information and are addressed in the Security Annex.

9A.9.2.1 System and Equipment Functions

Normal Functions (Non-Safety Category)

The Lighting and Servicing Power System provides illumination throughout the plant as required during Non-Safety Category normal conditions.

Normal Functions (Safety Category)

The Lighting and Servicing Power System does not perform a Safety Category function during normal conditions.

Off-Normal Functions (Non-Safety Category)

The Lighting and Servicing Power System provides illumination during Non-Safety Category, off-normal conditions, including fire, transient and accident conditions.

Off-Normal Functions (Safety Category)

The Lighting and Servicing Power System does not perform a Safety Category function during off-normal conditions.

The design of the Lighting and Servicing Power System is consistent with the requirements of REGDOC 2.5.2, (Reference 9A.9.2-1) Section 7.12.2 relative to lighting infrastructure, Section 7.20 relative to emergency lighting, and Section 8.10 relative to Control Facility lighting.

9A.9.2.2 Safety Design Bases

The Lighting and Servicing Power System does not perform or ensure any Safety Category function, and thus has no safety design bases.

The Lighting and Servicing Power System does not perform any Safety Category functions during or after a design basis accident. Failure of the lighting systems does not compromise Safety Category functions from being performed, nor does it prevent the safe shutdown of the reactor.

The plant illumination levels provided by the lighting systems are in accordance with the applicable lighting levels specified in NUREG-0700 (Reference 9A.9.2-2).

9A.9.2.3 Description

The lighting equipment is designed to provide illumination throughout the plant that is equal to or greater than those recommended by the Illuminating Engineering Society of North America.

Emergency lighting is provided in areas where emergency operations are performed and for personnel safety during a power failure. The individual rooms receive lighting appropriate to their contents. The Lighting and Servicing Power System is comprised of the following lighting systems:

- Normal lighting system - The normal lighting system is used to provide illumination in all areas of the plant except the Main Control Room and the Secondary Control Room (the emergency lighting system provides both normal and off-normal lighting in the control room). The normal lighting system is available under normal plant operating, maintenance, and testing conditions and during off-normal conditions if normal AC power remains available. The normal lighting system does not have a backup power source.
- Emergency lighting system - The emergency lighting system is used to provide acceptable levels of illumination in the control rooms during normal operating conditions while also providing acceptable levels of illumination in the control rooms and other areas of the plant when the normal lighting system is not available. Emergency lighting is available in areas where emergency operations are performed, including the access and egress routes to and from those areas. Upon loss of the normal lighting system (including loss of power events), the emergency lighting system provides illumination for the control rooms, remote shutdown areas, battery rooms, and containment. Emergency lighting has a backup source of power that is either diesel backed or battery backed.

The BWRX-300 uses Light Emitting Diode (LED) lighting that is operated from low voltage power panels where the lighting loads are shared with other loads (e.g., controllers and convenience outlets). These various power panels are powered by three sources:

- Offsite power (derived from busses A1 and B1 – Non-Safety Electrical Distribution System)
- Standby diesel generator (derived from busses A21 and B21 – Safety Class 2 and 3 Electrical Distribution System)
- Uninterruptible Power Supplies – either Safety Class 2 and 3 Electrical Distribution System or Safety Class 1 Electrical Distribution System that are backed by 72-hour batteries and most closely associated with emergency lighting

These panels are distributed in most rooms such that lighting for that room can be easily obtained. Extensive emergency lighting is provided since LED loads are small and insignificant to the diesel or battery loading.

The lighting is designed in accordance with applicable industry standards and is provided in all areas using an appropriate mixture of SCN, SC3 diesel backed and SC2 and SC1 battery backed lighting for the specific plant areas.

The lighting equipment and installation inside and outside the control room is designed to remain functional during design basis events and withstand the seismic loads of a design basis earthquake. Lighting fixtures located in the vicinity of Safety-Category function equipment is supported so that they do not adversely impact the equipment when subjected to seismic loading of a safe shutdown earthquake.

The MCR and SCR lighting over the SC1 video display units and the communications console are powered from the Safety Class 1 Electrical Distribution System. The remaining MCR and SCR lighting is split between the redundant diesel backed Safety Class 2 and 3 Electrical Distribution System power feeds. All control room lighting is part of the emergency lighting system since all control room lighting has a backup power source.

9A.9.2.4 Materials

The plant lighting is designed using materials that ensure operability in nuclear and industrial environments.

9A.9.2.5 Interfaces with Other Equipment or Systems

The plant lighting derives its power from the Electrical Distribution System (Chapter 8, Subsection 8.1).

9A.9.2.6 System and Equipment Operation

The plant lighting includes normal and emergency lighting. The normal lighting provides illumination during all plant operating conditions, including off-normal conditions if normal AC power is available. The emergency lighting provides illumination in the control rooms during all normal and off-normal conditions while also providing illumination in other areas of the plant during off-normal plant operating conditions when the normal lighting system is not available. Off-normal plant conditions for emergency lighting include fire, transient and accident conditions.

9A.9.2.7 Instrumentation and Control

No special instrumentation is required for the lighting system. The emergency lighting automatically actuates on the loss of normal plant lighting.

9A.9.2.8 Monitoring, Inspection, Testing, and Maintenance

Pre-operational testing verifies that the normal lighting system provides illumination under normal plant operating, maintenance, and testing conditions, and that the emergency lighting system provides illumination where required throughout the station, including areas where emergency operations are performed. System operability is demonstrated by normal use during plant operation. The normal lighting system and the control room portion of the emergency lighting system is normally energized continuously and requires only routine maintenance or testing.

9A.9.2.9 Radiological Aspects

There are no radiological aspects associated with the design and operation of the lighting system.

9A.9.2.10 Performance and Safety Evaluation

The lighting equipment used is conventional equipment that has a history of successful operation at similar nuclear and industrial settings. This equipment is used and maintained routinely to ensure its availability.

The lighting system has no Safety-Category functions and therefore requires no nuclear safety evaluation. Failure of the lighting system or its components will not affect the ability of Safety Category systems to perform their intended function.

9A.9.2.11 References

- 9A.9.2-1 CNAS Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.9.2-2 USNRC NUREG-0700, "Human-System Interface Design Review Guidelines."

9A.9.3 Floor Drain System

The purpose of the Equipment and Floor Drain System is to provide equipment to drain, collect, and transport liquid waste from the floor drains to the Liquid Waste Management System (Chapter 11, Subsection 11.2) for processing.

9A.9.3.1 System and Equipment Functions

9A.9.3.1.1 Normal Functions (Non-Safety-Category)

The EFS is designed to perform the following:

Collect radioactive or potentially radioactive liquid wastes generated in the plant during normal operation, anticipated operational occurrences, startup, hot shutdown, cold shutdown, and refueling, and transfer these wastes to the Liquid Waste Management System for processing or to an appropriate disposal system.

Collect oily wastes from the Electro-Hydraulic Control Unit, Diesel Generators, and Turbine Lube Oil skid during all modes of plant operation for transfer to the SWM (Chapter 11, Subsection 11.4).

9A.9.3.1.2 Normal Functions (Safety-Category)

The system does not perform any Safety-Category functions during normal conditions.

9A.9.3.1.3 Off-Normal Functions (Non-Safety-Category)

The system does not perform any Non-Safety-Category functions during off-normal conditions. The system is designed to collect spills and leaks via floor drains during off-normal conditions.

9A.9.3.1.4 Off-Normal Functions (Safety-Category)

As part of Defense Line 3 the EFS provides containment isolation valves on piping that penetrates the containment boundary. These valves are designed to close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System (Chapter 7, Subsection 7.3.1). The EFS can perform this function during and after a Design Basis Event requiring containment isolation.

The design of the EFS meets CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.6.6 as related to containment isolation. (Reference 9A.9.3-1).

9A.9.3.2 Safety Design Bases

As part of Defense Line 3 the EFS provides containment isolation valves on piping that penetrates the containment boundary.

9A.9.3.3 Description

Refer to Figure 9A.9.3-1a which depicts a typical drain sump and Figure 9A.9.3-1b which depicts the containment drain sump.

The EFS consists of the drainpipes, collection sumps, sump pumps, interconnecting piping, and instrumentation to provide for the collection and removal of liquid waste in the plant. The EFS is provided for the Reactor Building, Turbine Building, Radwaste Building and PLSA.

General Floor Drains and Sumps

All liquid waste collected in the radioactive drains are treated as High Conductivity Waste (HCW). Liquid waste is floor drain wastes, equipment drains, and process drains collected throughout the entire facility. These wastes are collected by drains and drain headers, routed by gravity drainage piping to a sump, and pumped to LWM for processing. All liquid waste is considered HCW in the

BWRX-300 with the exception of a small subset of potentially clean drains and sumps that are used to keep chemicals from mixing with radwaste and to reduce the amount of waste sent to processing. Liquid waste testing is utilized in order to test for radioactive contamination among the potentially clean collection samples.

The EFS continuously collects the liquid waste that is generated throughout the plant during all modes of plant operation. The sumps receive and store the liquid wastes until the contents are pumped to the LWM. Each sump has a cover to prevent debris from entering the sump. The sumps are vented to their associated HVAC exhaust systems for control of airborne contamination. The sumps are also provided with flush capability to minimize the solid deposit within the sump and transfer piping. Clean flush water can be added to clean out a sump prior to maintenance activities.

Open drainage lines from areas that are required to maintain an air pressure differential but drain to the system are provided with a water seal.

Oily Wastes Drains and Sump

Dedicated oily waste collection sumps are provided at the base of the EHC unit, lube oil skid and diesel generators in case of a leak. A temporary pump is required to pump oily waste from the sump pit to a collection barrel for transport to the SWM (Chapter 11, Subsection 11.4) drum evaporator for de-watering and disposal.

Pressurized Containment Sump

Three divisionally separated sumps are included in the basement floor of the RB. One of the sumps is a pressurized containment sump. The EFS has drain piping in the SCCV floor to drain any leakage or condensation collected in containment to the pressurized sump. The pressurized sump is also used for containment leak detection and measurement. The pressurized sump's upstream piping penetrates the SCCV which requires the use of redundant isolation valves to ensure containment isolation. The isolation valves fail closed and automatically close upon receiving an isolation signal from the Safety Class 1 Instrumentation and Control System (Chapter 7, Section 7.3).

9A.9.3.3.1 Component Description

Sump Pumps

The sump pumps are vertical, centrifugal types, driven by electric motors. Two parallel 100% capacity pumps (run pump and standby pump) are provided for each sump.

Maintenance valves and flanged connections are provided for sump pumps so that they can be isolated and removed from the sumps for maintenance and repair. Check valves in the pump discharge piping prevent backflow of waste into the sump.

Collection Sumps

The sumps are sized to accommodate the normal anticipated daily inputs without overflow in conjunction with operation of one sump pump.

Collection sumps are lined cavities in the floor slab of their respective building. The sumps have tight fitting, but not gastight, steel plate covers to prevent the entrance of debris as well as to minimize airborne contamination. The sumps have vents that are piped to the building HVAC interfacing exhaust. The sumps are provided with grab sampling connections.

The pressurized containment sump is designed in accordance with ASME Boiler and Pressure Vessel Code Section VIII, Division 1 (Reference 9A.9.3-3). The pressurized sump vessel is steel

lined and shielded due to its interface with the primary containment. Any leakage from the pressurized sump is contained in the lined cavity, eliminating direct contact with the bare concrete.

A pressurized vessel is used in place of a traditional sump in the Reactor Building basement due to its open interface with primary containment during normal operation. The pressurized sump is located outside containment to maintain a dry containment. The pressurized vessel provides an interface for a drain line, flush water, level transmitters and pump connections.

Piping and Valves

All piping is in accordance with the requirements of ASME B31.1 (Reference 9A.9.3-4) except for the containment penetration. As part of Defense Line 3, the EFS provides containment isolation valves on piping that penetrates the containment boundary. Accordingly, the piping penetrating containment, up to and including both isolation valves, is Safety-Class 1.

The EFS piping penetrating containment is an extension of the containment boundary and is designed to Category A; ASME BPVC Section III, Division 1, Class 2 requirements. The arrangement of the isolation valves and connecting piping is such that a single active failure or passive failure in the connecting piping or an outboard valve, cannot prevent isolation of the EFS containment penetrations. The containment isolation valves are classified as Seismic Category B.

The floor drain line in containment connects directly to the containment atmosphere, penetrates the primary reactor containment, and is provided with containment isolation valves. There are two containment isolation valves placed in series, located outside the containment vessel, and placed as close to the primary containment wall as practical.

These isolation valves are located to provide accessibility for maintenance, inspection, and testing during all modes of reactor operation.

The pressurized sump system is credited for unidentified leak measurement, so the isolation valves on the EFS containment drain line are designed to be normally open to connect the drain line to the pressurized sump. To ensure a power source availability for the containment isolation valves, the isolation valves are supplied by different power source divisions. The EFS containment isolation valves have provisions to close automatically. Upon loss of actuating power, the automatic isolation valves are designed to fail closed, taking the position that provides greater safety.

All ESF piping is stainless steel to minimize corrosion.

Piping entering the collection sumps is turned down and terminated below the lowest sump fluid level to which the sump pump can draw. This provides a water seal to prevent gas flow and cross-contamination of building areas.

9A.9.3.4 Materials

Material and process control requirements for the BWRX-300 components ensure the reliability of plant operations through its design life by minimizing corrodents and mitigating the degradation of materials through material chemistry, heat treatment, material processes controls and periodic inspection. The ESF piping is manufactured from stainless steel and the sumps are manufactured using materials which have a high degree of resistance to effluents treated by the system. Proper selection of radiation-resistant materials of construction is included in individual equipment specifications as all liquid collected is treated as potentially contaminated.

9A.9.3.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.9.3-1 for ESF interfaces.

9A.9.3.6 System and Equipment Operation

Normal Operational Concept

The ESF operates automatically in the same manner under all normal operating modes of the plant (startup, power operation, hot shutdown, cold shutdown, and refueling). Liquid wastes generated in the plant are directed into drain fixtures and drain piping. The ESF continually collects these radioactive or potentially radioactive liquid wastes and convey the wastes via gravity drainage piping systems to their respective collection sumps. Each pump is sized to accommodate the normal anticipated daily inputs into the sump. The system is not designed to accommodate fire suppression water. However, the system design features prevent drain or flood water from backing up in the drainage system into areas housing Safety-Category function SSC's.

No credit is taken for the drainage system in the internal flooding analysis.

Dual 100% capacity sump pumps are provided on each sump pump to pump liquid waste to LWM. The second pump automatically starts on sump high-high water level, provides redundancy in the event of sump pump failure, and assists the single pump when abnormally high inputs into the sump occur.

Off-Normal Operational Concept

In the event of a Loss-of-Coolant Accident (LOCA) signal, the containment isolation valves perform a Safety-Category function and close, thereby maintaining the integrity of the SCCV.

Under Design Basis Event (DBE) conditions and assuming the unavailability of normal power, the ESF does not operate, nor is it required to be functional. The containment isolation valves fail closed.

All drainage piping whose collapse could result in a loss of function of Safety-Category function SSC equipment, is seismically analyzed to remain intact following an SSE.

9A.9.3.7 Instrument and Control

Instrumentation

Redundant level sensors and transmitters are provided for each sump. Redundancy is appropriate for ensuring reliability and availability of the I&C systems. Level signals are used to start and stop the sump pumps based on High and Low setpoints. The level transmitters on the pressurized containment sump are also used to detect, measure, and trend leakage inside containment. Pump run timers are provided for each sump pump.

Local pressure transmitters are provided in the discharge piping of each sump pump which display the pressure remotely in the Main Control Room. Additionally, the pressurized sump that penetrates the SCCV is provided with discharge flow transmitters that displays the flow remotely in the Main Control Room.

Isolation diaphragms are installed between all gauges or transmitters and the piping, to avoid contaminating the instrument.

Controls

The following describes the key control features of the EFS.

Sump pump operations automatically maintain the sump level. The lead sump pump starts upon a high level indication and stops on low level indication. When the capacity of one sump pump is exceeded and the sump level rises above the high level, a high-high level setting is reached, and the standby pump starts. Both pumps operate until the low level is reached, causing both pumps

to stop. Alarms are annunciated at the high-high level to notify the operator of potential sump overflow and at the low-low level to notify the operator of potential air cross-contamination due to the loss of water seal. Sump pump controls are in the Main Control Room.

Where needed, sumps are provided with remote flush capability to clean out sumps prior to maintenance activities to minimize solid deposits.

The EFS containment isolation valves are controlled via the Safety Class 1 Instrumentation and Control System, and close upon receiving an isolation signal. These valves can also be remotely operated from the control rooms.

9A.9.3.8 Monitoring, Inspection, Testing, and Maintenance

The containment isolation valve closure time is measured during the valve operability test and the leakage is measured during the valve leakage test as specified in the containment leakage testing procedures. Each sump in the EFS utilizes recirculation lines with a grab sample tap to test/analyze the contents of the sump. Leak detection and inspection for primary containment isolation features comply with the requirements of ASME Boiler and Pressure Vessel Code Section III, Division 1, Class 2 (Reference 9A.9.3-2). A test connection is provided between the isolation valves to support local leak rate testing of the primary containment boundary.

9A.9.3.9 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.9.3.10 Performance and Safety Evaluation

As part of DL3, the EFS performs a containment isolation Safety-Category function. The CIVs are designed to close upon receipt of an isolation signal from the Safety Class 1 Instrument and Control System. The CIVs are designed to maintain the leak tightness of the containment in the event of an accident and prevent radioactive releases to the environment that exceed prescribed limits.

9A.9.3.11 References

- 9A.9.3-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.9.3-2 ASME Boiler and Pressure Vessel Code (BPVC), Section III, "Rules for Construction of Nuclear Power Plant Components, Division 1 - Subsection NC, Class 2 Components," American Society of Mechanical Engineers.
- 9A.9.3-3 ASME Boiler and Pressure Vessel Code (BPVC), Section VIII, "Rules for Construction of Pressure Vessels, Division 1," American Society of Mechanical Engineers.
- 9A.9.3-4 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.

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Table 9A.9.3-1: Equipment and Floor Drain System Interfaces

Interfacing System	Interface Description	Interface Boundary
Safety Class 1 Instrumentation and Control System	Provides all Safety Class 1 I&C control (Containment Isolation Valves)	Primary Containment Isolation Valves
Non-safety Instrumentation and Control System	Provides all Non-Safety Category I&C control. Sump pump logics, valve logics, instrumentation (pressure, flow, temperature, level) etc.	Instruments
Process Radiation and Environmental Monitoring System	Provides process sampling for radiological analysis radiation detection and the instrumentation used for containment leak detection and monitoring	Sampling locations
Control Rod Drive System	Provides drain piping for CRD pumps and the Hydraulic Control Unit room. The CRD wash down drains into the pressurized containment sump	Drain hub
Liquid Waste Management	Sump pumps discharge effluent to LWM for processing (LWM collection tank) Provides seal water and flushing water for sumps	Collection Tank
Solid Waste Management	Oily waste is collected and pumped to SWM drum evaporator for dewatering and disposal	Drum Evaporator
Condensate and Feedwater Heating	Provides drain piping from CFS to EFS	Drain hub
Condensate Filters and Demineralizers	Provides drain piping from CFD to EFS	Drain hub
Main Turbine Equipment	Provides drain piping from MTE to EFS	Drain hub
Main Condenser and Auxiliaries	Provides drain piping from MCA to EFS	Drain hub
Plant Cooling Water	Dedicated clean sump for PCW system. The PCW dedicated sump discharges the collected water back to the makeup water surge tank	Drain hub
Plant Pneumatics System	Provides air to EFS Air Operated Valves	PPS Tank/Air Operated Valves

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Interfacing System	Interface Description	Interface Boundary
Safety Class 2 and 3 Electrical Distribution System	Provides diesel generator oil catching from Safety Class 2 and 3 Electrical Distribution System	Drain Hub
Non-Safety Electrical Distribution System	Provides power to pumps and valves	Pump motors and valve motors
Steel-plate Construction Containment Vessel	EFS provides piping to drain the SCCV	SCCV wall and piping penetration
Containment Cooling System	Pressurized containment sump collects condensate from the air handling unit's drain pans	Air Handling Units drain pan
Heating Ventilation and Cooling System	The collection sumps are vented to Heating, Ventilation and Air Conditioning ductwork	Sump vent duct
Fire Protection System	EFS collects FPS water EFS sump overflow due to Fire Protection System water is allowed with a maximum flood depth required in all associated buildings	Collection Sumps
Reactor Building Structure	Sloped drain piping	Reactor Building structure and floor
Turbine Building Structure	Sloped drain piping embedded in concrete floor	Slab on grade concrete floor
Radwaste Building Structure	Sloped drain piping embedded in concrete floor	Slab on grade concrete floor
Yard/Balance of Plant	EFS discharges PPS condensate from the PPS to the yard storm water drain	Drain hub

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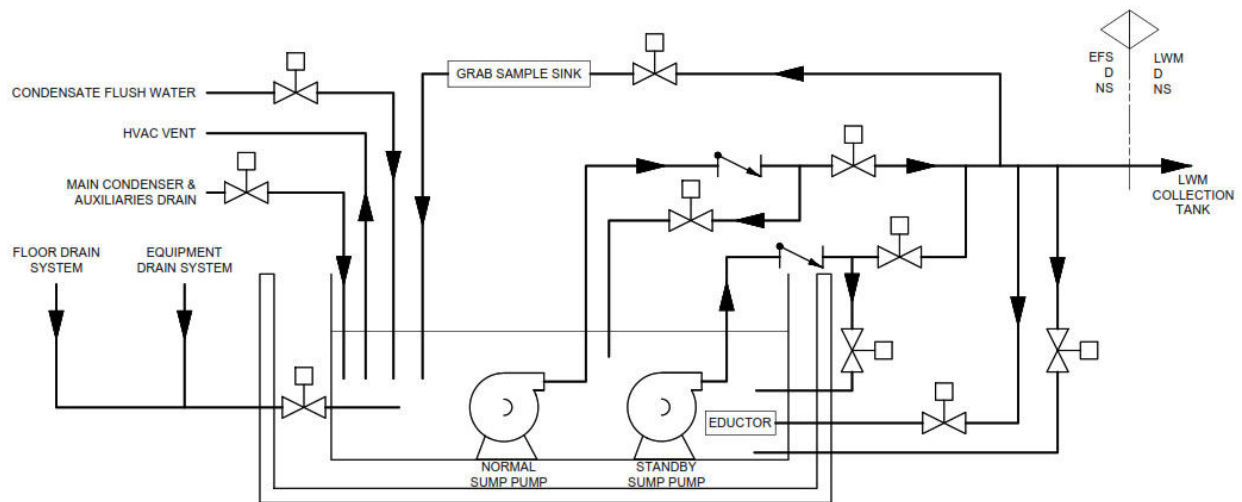


Figure 9A.9.3-1a: Floor Drain System Drain Sump (Typical)

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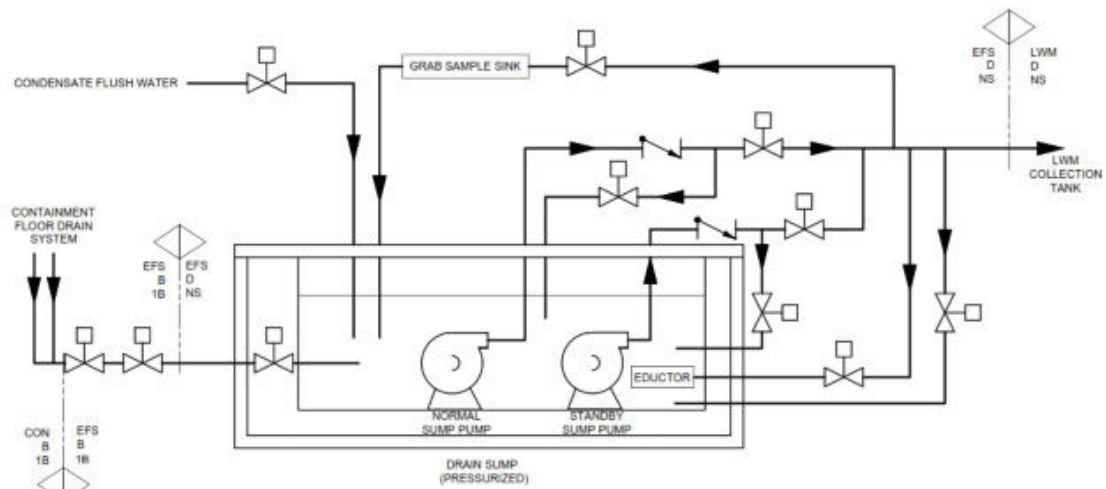


Figure 9A.9.3-1b: Floor Drain System Containment Drain Sump (Pressurized)

9A.9.4 Interfacing Water Systems

9A.9.4.1 Potable Water System

The Potable Water System design is dependent on the site-specific water pathways. The system is designed to supply the required volume of potable water during peak demand periods.

9A.9.4.2 System and Equipment Functions

Normal Functions (Non-Safety-Category)

Provides for potable water distribution inside the power block. The Potable Water System brings potable water to the power block from the city water source.

Normal Function (Safety Category)

There are no Safety-Category functions related to the Potable Water System.

Off-Normal Functions (Non-Safety Category)

There are no specific off-normal functions related to the Potable Water System.

Off-Normal Functions (Safety Category)

There are no Safety-Category functions related to the Potable Water System.

The safety classification of the Potable Water System is consistent with the requirements of CNSC REGDOC-2.5.2, Section 7.1 (Reference 9A.9.4.12-1).

9A.9.4.3 Safety Design Bases

The Potable Water System does not perform, ensure, or support any Safety-Category function, and thus has no safety design bases.

9A.9.4.4 Description

The potable water system provides potable water to the power block. The system includes faucets, toilets, showers, and hot water heaters.

The potable water system provides makeup water to the demineralized water trailers if needed.

The Potable Water System consists of potable water supply piping from the Municipality of Durham. The supply line includes isolation valves, flow totalizer and instrumentation. Heat tracing is provided for all outside above ground piping and instrumentation. A distribution header supplies the power block and other onsite buildings. Provisions are made in the distribution header to accommodate future demand.

Potable water supplies the following areas:

1. Mobile demineralized water trailers as required to supplement Darlington Nuclear Generating Station (DNGS) demineralized water supply
2. Restrooms, food service equipment and battery room safety showers in the Control Building
3. Water is supplied to the toilet in the Main Control Room

9A.9.4.4.1 Component Description

The Potable Water System components include: underground supply lines, isolation valves, and necessary instrumentation. Heat tracing is provided for all outside above ground piping and

instrumentation. Where required, back flow preventers are installed. Hot water heaters, faucets and showers are installed where needed.

Piping and valves associated with the Potable Water System are designed and manufactured in accordance with applicable codes and standards for the applications involved. These codes and standards include but are not limited to ASME B31.1 Power Piping (Reference 9A.9.4.12-2), National Plumbing Code of Canada (Reference 9A.9.4.12-3), Regulation 332/12 Ontario Building Code (Reference 9A.9.4.12-4), and American Society for Testing and Materials D3350-14 Standard Specification for Polyethylene Plastics Pipe and Fittings Material (Reference 9A.9.4.12-5).

9A.9.4.5 Materials

The Potable Water System provides piping, valves, and other control components to distribute potable water to final use locations. Potable water system component materials are built to the Canadian National Building Code.

9A.9.4.6 Interfaces with Other Equipment or Systems

The potable water system interfaces with the following other systems:

Demin Water – There is a potable water connection to provide for temporary potable water supply to portable demineralizer trailers.

Sanitary Sewer – Potable water supplies water to toilets, sinks, showers which drain to the sanitary sewer system.

Electrical – Low voltage power from the Non-Safety Electrical Distribution System is supplied to components in the system to provide for heating of water.

9A.9.4.7 System and Equipment Operation

The potable water system relies on system pressure being supplied from the Municipality of Durham. Hot water heaters are provided where necessary. The hot water heaters temperature is automatically controlled.

The Potable Water System is designed for operation in all modes.

9A.9.4.8 Instrumentation and Control

Instrumentation for the potable water system includes flow metres to track usage of water, back flow preventers where required by code, temperature control thermostats in the hot water heaters, and relief valves in the system where required.

Instrumentation such as valve position switches, pressure gauges, and temperature sensors are located throughout the system.

9A.9.4.9 Monitoring, Inspection, Testing, and Maintenance

The potable water system is tested and commissioned in accordance with the National Plumbing Code of Canada and Ontario's Watermain Disinfection Procedure. Commissioning includes hydrostatically tested for leak-tightness, disinfection, and flushing of the system.

Periodic testing for residual chlorine and micro biologics is done in accordance with the Safe Drinking Water Act, the Ontario Drinking Water Quality Management Standard, and Guidelines for Canadian Drinking Water Quality.

9A.9.4.10 Radiological Aspects

The Potable Water System has no interconnections to systems with the potential for containing radioactive material.

Chapter 12, Subsection 12.3 provides information pertaining to design measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.9.4.11 Performance and Safety Evaluation

There is no potable water piping in the reactor building. Potable water is limited to Non-Safety-Category function areas of the plant. The MCR washroom and kitchen area sinks, and potable water piping are sufficiently remote from the control system to eliminate the potential for interactions.

9A.9.4.12 References

- 9A.9.4.12-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.9.4.12-2 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 9A.9.4.12-3 "National Plumbing Code of Canada," Canadian Commission on Building and Fire Codes, National Research Council of Canada.
- 9A.9.4.12-4 O. Reg. 332/12, "Building Code," Government of Ontario.
- 9A.9.4.12-5 American Society for Testing and Materials D3350-14, "Standard Specification for Polyethylene Plastics Pipe and Fittings Material."

9A.9.5 Makeup Water System

The Makeup Water System consists of supply piping from the Darlington Nuclear Generating Station (DNGS) site, trailer hookup piping, storage tank, transfer pumps, and distribution piping.

9A.9.5.1 System and Equipment Functions

Normal Functions (Non-Safety-Category)

Provides for demineralized water storage and distribution.

Normal Function (Safety Category)

The system does not perform any Safety-Category functions during normal conditions.

Off-Normal Functions (Non-Safety Category)

Following a reactor trip with no condenser available, the Makeup Water System provides makeup to the Isolation Condenser pools (Chapter 5, Subsection 5.8).

Off-Normal Functions (Safety Category)

The Makeup Water System provides containment isolation of Makeup Water System piping penetrating the SCCV. This isolation is provided by two locked closed manual isolation valves which are Safety Class 1 and Defense Line 3 – one valve inboard and one valve outboard of the Primary Containment. The containment isolation valves are only open during Mode 5 or 6.

The Makeup Water System is not required to operate during or after a design basis event except for performing the containment isolation function. The design of the Makeup Water System meets

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CNSC requirements specified in CNSC REGDOC-2.5.2 Section 8.6.6 as related to containment isolation. (Reference 9A.9.5.11-1).

9A.9.5.2 Safety Design Bases

The Makeup Water System containment penetration and isolation valves perform the safety function of maintaining containment integrity.

9A.9.5.3 Description

The makeup water supply from the DNGS site is sized to provide for normal makeup water needs for the unit as well as additional capacity to refill the ICS pools in a reasonable timeframe following a typical initiation. During certain shutdown/refueling/startup mode evolutions the increases in plant water consumption may require use of a temporary demineralization subsystem to be used as a supplemental water source.

Demineralized water of the proper water quality specification is supplied to the single demineralized water storage tank located in the site yard. The tank is heated to prevent freezing. Two 100% pumps supply the demineralized water system.

The demineralized water system provides makeup water to: Condensate Storage Tank, Isolation Condenser Pools, Ventilation Humidification, Chilled Water and Plant Cooling Water surge tanks, Boron Injection Tank, and washdown stations on the refuel floor and in containment. Additional demineralized water connections are located where needed for maintenance of systems.

The makeup water equipment and associated piping in contact with demineralized water are fabricated from corrosion-resistant materials to prevent contamination of the makeup water.

Piping and valves associated with the Makeup Water System are designed and manufactured in accordance with applicable codes and standards for the applications involved. These codes and standards include but are not limited to ASME B31.1 Power Piping (Reference 9A.9.5.12-2), National Plumbing Code of Canada (Reference 9A.9.5.12-3) – American Water Works Association D100 (Reference 9A.9.5.12-4).

9A.9.5.4 Materials

Water quality requirements are used in the selection of Makeup Water System components.

9A.9.5.5 Interfaces with Other Equipment or Systems

Refer to Table 9A.9.5-1 for Makeup Water System interfaces with other systems.

9A.9.5.6 System and Equipment Operation

The demineralized water storage tank is filled as necessary from the DNGS water treatment plant. Filling of the tanks is a batch process that is manually controlled. Level alarms prompt the operator to secure filling when needed.

Normal plant operation includes Power Operation, Startup Operation, Hot Shutdown Operation, Stable Shutdown Operation, Cold Shutdown Operation, and Refueling Operation. During normal operations the Makeup Water System provides demineralized water storage and distribution to required systems and equipment in the plant.

The demineralized water in containment is only used when the plant is shutdown. During plant operation both inside and outside manual containment isolation valves are locked closed.

9A.9.5.7 Instrumentation and Control

Level alarms on the demineralized water storage tank alert the operator to high or low level situations.

The storage tank heater as well as heat trace on above ground yard piping is automatically controlled.

9A.9.5.8 Monitoring, Inspection, Testing, and Maintenance

Containment Isolation Valves are periodically tested to validate operability and determine if valve leakage is within acceptable limits. Test and vent connections are provided at the containment isolation valves to verify that the valves meet the local leak rate limits.

Makeup water is monitored and tested to ensure it meets the water quality requirements for interfacing systems.

Periodic and condition-based maintenance are completed for instrumentation, pump vibrations, and other equipment to ensure the proper performance of the system.

9A.9.5.9 Radiological Aspects

Chapter 12, Subsection 12.3 provides information pertaining to design measures that can be taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are as ALARA in operational states and in accident or post-accident conditions.

9A.9.5.10 Performance and Safety Evaluation

The Makeup Water System performs the containment isolation Safety-Category function using two manual valves. The containment isolation valve arrangement meets the requirements of Section 8.6.6 of REGDOC-2.5.2 (Reference 9A.9.5-1).

9A.9.5.11 References

- 9A.9.5-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.9.5.11-2 ASME B31.1, "Power Piping," American Society of Mechanical Engineers.
- 9A.9.5.11-3 "National Plumbing Code of Canada," Canadian Commission on Building and Fire Codes, National Research Council of Canada.
- 9A.9.5.11-4 AWWA D100, "Welded Carbon Steel Tanks for Water Storage," American Water Works Association.

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Table 9A.9.5-1: Makeup Water System Interfaces

Interfacing System	Interface Description	Interface Boundary
Non-Safety Instrumentation and Control System	Provides all SCN Instrumentation and Control for the demineralized water equipment, potable water, sewage treatment, and hydrogen subsystems	Non-Safety Class Instruments
Non-Safety Electrical Distribution System	Provides power to demineralized water transfer pumps and tank/piping heating	SCN Pumps and heater, and controls
Process Radiation and Environmental Monitoring System	Provides demineralized water to Process and Radiation Monitoring for process sampling	Sampling points and RWB chemical Laboratory
ICS Pool Cooling and Cleanup System	Provides Demineralized water to the IC pools through the ICS Cooling and Cleanup system	Interface valve
Boron Injection System	Provides demineralized water to the Boron Injection Sodium Pentaborate Storage Tank for filling	Interface valve
Plant Cooling Water System	Provides demineralized water to the surge tanks	Interface valve
Chilled Water Equipment System	Provides demineralized water to the glycol auto fill unit	Interface valve
Heating, Ventilation and Cooling System	Provides demineralized water to CB AHUs used for space humidification	AHUs
Liquid Waste Management System	Provides demineralized water to the Condensate storage tank	Interface valve
Refueling Equipment and Servicing	Provides demineralized water connections on the refueling floor for use during outages	Interface valve
Main Condenser and Auxiliaries	Provides fill and makeup water to the condenser vacuum skids for seal water separators	Interface valve
DNGS Demineralized Water System	System is supplied demineralized water from the DNGS Demineralized Water System	Yard demineralized water tie-point connection

9A.9.6 Sanitary Water Systems

9A.9.6.1 System and Equipment Functions

Normal Functions (Non-Safety Category)

Provides for sewage water collection and delivery to the existing DNGS East Sewage Lift Station.

Normal Function (Safety Category)

There are no normal Safety-Category functions related to the Sanitary Water System.

Off-Normal Functions (Non-Safety Category)

There are no off-normal functions related to the Sanitary Water System.

Off-Normal Functions (Safety Category)

There are no specific off-normal functions related to the Sanitary Water System.

The safety classification of the Sanitary Waste System is consistent with the requirements of CNSC REGDOC-2.5.2, Section 7.1 (Reference 9A.9.6.11-1).

9A.9.6.2 Safety Design Bases

The Sanitary Water System does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

9A.9.6.3 Description

The Sanitary Sewer Collection and Delivery Subsystem collects sewage from the facility washrooms and break areas and transfers the sewage to the existing DNGS East Sewage Lift Station. The sewage system is a non-radiologically contaminated system and collects sewage only from areas outside of any radiologically controlled areas in the CB. This subsystem is sized for normal plant operation of the CB. The Administration Building and the Security Building sewage system is provided by OPG.

The Sanitary Sewage Handling Subsystem is designed to prevent raw sewage overflow in the event of a power outage.

Piping and valves are designed and manufactured in accordance with applicable codes and standards for the applications involved. These codes and standards include National Plumbing Code of Canada (Reference 9A.9.6.11-2), and Regulation 332/12 Ontario Building Code (Reference 9A.9.6.11-3).

9A.9.6.4 Materials

The water quality requirements are used in the material selection and design of the water treatment systems.

9A.9.6.5 Interfaces with Other Equipment or Systems

The sanitary water system interfaces with the DNGS waste treatment system.

9A.9.6.6 System and Equipment Operation

The Sanitary Water System is designed for operation in all modes.

Sanitary waste drains are gravity drained through underground piping to a lift station which is part of the DNGS waste treatment system.

9A.9.6.7 Instrumentation and Control

Alarm indication is provided as needed to inform the operator of the status of the DNGS East Lift Station pump.

9A.9.6.8 Monitoring, Inspection, Testing, and Maintenance

Inspection and testing of the Sanitary Water System is performed in accordance with the applicable requirements of the Ontario Building Code.

9A.9.6.9 Radiological Aspects

The Sanitary Water System is a non-radiologically contaminated system and only collects sanitary waste from areas outside of any radiologically controlled areas in the CB.

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.9.6.10 Performance and Safety Evaluation

The Sanitary Water System does not perform, ensure, or support any Safety-Category function; therefore a Safety Evaluation is not required.

9A.9.6.11 References

- 9A.9.6.11-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 9A.9.6.11-2 "National Plumbing Code of Canada," Canadian Commission on Building and Fire Codes, National Research Council of Canada.
- 9A.9.6.11-3 O. Reg. 332/12, "Building Code," Government of Ontario.

9A.9.7 Chemistry Systems

9A.9.7.1 Primary Coolant

Not Applicable, the terminology "Primary Coolant" refers to a Pressurized Water Reactor.

9A.9.7.2 Secondary Coolant

Not Applicable, the terminology "Secondary Coolant" refers to a Pressurized Water Reactor.

9A.9.8 Other Process Media and Other Materials

9A.9.8.1 Hydrogen Water Chemistry

9A.9.8.1.1 System and Equipment Functions

The Hydrogen Water Chemistry System adds hydrogen into the feedwater system at the Reactor Feed Pump suction and oxygen (as a constituent of air) into the Offgas System (Chapter 11, Section 11.3) via the Service Air System (Subsection 9A.4.1). The Hydrogen Water Chemistry System is included in the BWRX-300 design for the purpose of reducing and mitigating IGSCC in reactor vessel internals.

9A.9.8.1.2 Safety Design Bases

The Hydrogen Water Chemistry System does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

9A.9.8.1.3 Description

The Hydrogen Water Chemistry System provides for an established technique for reducing and preventing the growth rates of Intergranular Stress Corrosion Cracking (IGSCC) in reactor vessel internals. The mitigation of IGSCC is achieved from the reduction of oxygen and other oxidizing species (oxidants) in the reactor coolant. This is accomplished by injecting hydrogen into the feedwater at the feedwater pump suction. The injected hydrogen suppresses the radiolytic formation of oxidants in the reactor core. To compensate for any excess hydrogen which may travel downstream and be removed from the main condenser by the Offgas System, a corresponding amount of oxygen, as a constituent of injected air provided by the Service Air System, is injected into the Offgas System prior to Offgas Recombiner.

Reduction of oxidants results in a reduction of the Electrochemical Corrosion Potential, which is the measurement that is used to predict initiation and growth of IGSCC. When the Electrochemical Corrosion Potential is reduced to an acceptable level, IGSCC crack initiation stops, and crack growth rates are minimized. The BWRX-300 design employs NobleChem™ technology as a means to provide for the injection of noble metal(s) into the reactor to aid in the protection of reactor vessel internals from IGSCC in combination with the addition of hydrogen by the Hydrogen Water Chemistry System. As a result of NobleChem™ injection, the noble metal deposition in the reactor provides a catalyst effect on vessel surfaces to facilitate the recombination of free hydrogen and oxygen molecules to minimize the oxygen available to initiate and encourage IGSCC crack growth. Refer to Subsection 9A.9.9 for a description of the application of NobleChem™ technology.

The Hydrogen Water Chemistry System injects hydrogen into the feedwater system at the Reactor Feed Pump (RFP) suction to mitigate IGSCC in the reactor internals. The injected hydrogen causes a reduction in dissolved oxygen and other oxidants within the reactor internals by lowering the radiolytic production of hydrogen and oxygen in the vessel core region.

Hydrogen addition to the feedwater results in an excess ratio of hydrogen to oxygen at the entrance to the Offgas System. The Hydrogen Water Chemistry System injects a stoichiometric

amount of oxygen (as a constituent of air) upstream of the offgas recombiner to combine with the hydrogen reaching the recombiner. The amount of air (oxygen) can be determined from the hydrogen injection rate and is controlled by the Hydrogen Water Chemistry System utilizing time delays in the controls to ensure that hydrogen and oxygen reach the offgas recombiner in balanced amounts. The net result is that with air (oxygen) injection, the Offgas System operates at conditions (relative to gas constituents needed to recombine to form water vapor) very much like the conditions without Hydrogen Water Chemistry System. For most BWRs that use NobleChem™ technology with a Hydrogen Water Chemistry System featured with air injection into the offgas system, the actual air flow injection can be determined from the hydrogen injection rate and condenser air inleakage and, is controlled by the Hydrogen Water Chemistry System controller to maintain sufficient percent oxygen exit downstream of the offgas recombiner.

The Hydrogen Water Chemistry System consists of a Main Control Panel, a Hydrogen Isolation Module (to isolate the hydrogen supply from the power plant buildings if necessary), a Hydrogen Flow Control Module, which controls the flow of hydrogen into the Feedwater System, and an Air Flow Control Module, which controls air flow to the Offgas System. Note that Hydrogen Area Monitors are also included in the system which monitor for local hydrogen in the areas of susceptible system equipment.

9A.9.8.1.4 System and Equipment Operation

Power Operation

The Hydrogen Water Chemistry System can only be operated during power operations. The Hydrogen Water Chemistry System begins operation when the RFP pump is in service and Steam Jet Air Ejectors (SJAE) and Offgas System are in-service during plant startup.

9A.9.8.1.5 Instrumentation and Control

Instrumentation is provided to control the injection of hydrogen and the injection of oxygen (as a constituent of air) via the Service Air System. Automatic control features in the Hydrogen Water Chemistry System minimize the need for operator attention and improves performance.

9A.9.8.1.6 Monitoring, Inspection, Testing, and Maintenance

The Hydrogen Water Chemistry System is demonstrated functional by its use during normal operation.

9A.9.8.1.7 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.9.8.1.8 Performance and Safety Evaluation

The Hydrogen Water Chemistry System does not perform a Safety-Category function. However, the Hydrogen Water Chemistry System is used, along with other measures, to reduce the likelihood of corrosion failures that would adversely affect plant availability. The means for storing and handling hydrogen utilizes the guidelines in Electric Power Research Institute (EPRI) Report "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations" (Reference 9A.9.8-1).

9A.9.8.1.9 References

- 9A.9.8-1 EPRI Report NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," Electric Power Research Institute.

9A.9.9 NobleChem™ Injection

9A.9.9.1 System and Equipment Functions

On-Line NobleChem™ (OLNC) provides a means for the injection of noble metal salt solution directly into the reactor coolant flow path. The purpose of injecting noble metal into the reactor is to aid in the protection of reactor vessel internals from IGSCC in combination with the addition of hydrogen by the Hydrogen Water Chemistry System (Subsection 9A.9.8.1).

9A.9.9.2 Safety Design Bases

The OLNC does not perform, ensure, or support any Safety-Category function, and thus, has no safety design bases.

9A.9.9.3 Description

The OLNC application process deposits noble metal salt solution directly into the reactor coolant flow path. The resulting noble metal deposition in the reactor provides a catalyst effect on vessel surfaces to facilitate the recombination of free hydrogen and oxygen molecules to minimize the oxygen available to initiate and encourage IGSCC crack growth. Due to the catalyst effect from noble metal loading, utilizing OLNC in conjunction with a Hydrogen Water Chemistry System mitigation strategy results in a much lower amount of injected hydrogen required to achieve IGSCC mitigation levels than the use of Hydrogen Water Chemistry System alone. This need for less hydrogen to provide mitigation results in lower main steam line dose rates.

OLNC operates in conjunction with the Hydrogen Water Chemistry System although they are both separate systems which have no mechanical or electronic interrelationships. The Hydrogen Water Chemistry System is used to provide a ratio of hydrogen to oxygen in excess of the stoichiometric balance which causes a significant decrease in the electrochemical corrosion potential of the reactor vessel and internal components. As long as excess hydrogen is maintained, crack initiation and growth are greatly reduced, even at high bulk liquid oxidant levels. OLNC injection provides a catalytic surface on reactor vessel internals to enhance the function of the Hydrogen Water Chemistry System. Based on the catalytic surface recombination efficiency provided by OLNC, less hydrogen is required (from the Hydrogen Water Chemistry System) to establish the Electrochemical Corrosion Potential to mitigate IGSCC initiation and significantly reduce IGSCC crack growth rates.

The OLNC injection skid is installed as permanent plant equipment outside of the Steel-plate Composite Containment Vessel in a Non-Safety-Category area as close as possible to the Feedwater injection tap(s). A direct connection to a supply of demineralized water provides the carrier flow to lessen the residence time of the noble metal solution in the injection line. The effluent from the OLNC injection skid is injected into a tap (or taps) in the Feedwater System (via an injection quill) at a point that lessens the loss of noble metal prior to entering the reactor.

Once the flow from the injection skid reaches the Feedwater System, the process employs the reactor coolant as the transport medium to deposit noble metal on the surface of all wetted reactor components and inside existing cracks.

The boundary between OLNC and the Condensate and Feedwater Heating System (Chapter 10, Subsection 10.3.2) is downstream of the isolation and root valves to the CFS to a shared flange with the CFS (and also includes the injection quill). A single input tie for demineralized water as well as plant electrical power from the Non-Safety Electrical Distribution System is provided to support operation of the system.

9A.9.9.4 System and Equipment Operation

Power Operation

The OLNC injection is designed to be performed during power operations.

9A.9.9.5 Instrumentation and Control

Instrumentation is provided to monitor and control the injection of a noble metal solution into the reactor coolant flow path at the injection skid.

9A.9.9.6 Monitoring, Inspection, Testing, and Maintenance

OLNC injection is demonstrated functional by its use during normal operation.

9A.9.9.7 Radiological Aspects

Chapter 12, Subsection 12.1.5.4 provides information pertaining to measures taken to ensure that occupational exposures arising from the operation or maintenance of the equipment or system are ALARA in operational states and in accident or post-accident conditions.

9A.9.9.8 Performance and Safety Evaluation

OLNC does not perform a Safety-Category function. The OLNC Injection System is used, along with the Hydrogen Water Chemistry System to reduce the likelihood of IGSCC.

9A.9.10 Chemical Bases of Water Treatment

Refer to Chapter 13, Subsection 13.3.1 for information pertaining to the Chemistry Control Program.

9A.10 Storage System for Non-Permanent Equipment Used in Design Extension Conditions

The storage system for non-permanent equipment used in Design Extension Conditions is developed based upon Emergency Mitigating Equipment requirements. The design of the storage system takes into consideration the aggregate set of on-site and off-site resource considerations for the hazards that are applicable to the site as well as final BWRX-300 design considerations. These factors plus consideration for the following are used as design inputs to size, locate, deploy, and operate the storage system”

- Protection of EME
- Deployment of EME
- Procedural Interfaces
- Utilization of off-site resources
- Use of existing site facilities for storage



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**Ontario Power Generation Inc.
Darlington New Nuclear Project
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 12
Radiation Protection**

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
0	All	Initial Release

ACRONYM LIST

Acronym	Explanation
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
ARM	Area Radiation Monitoring Subsystem
BWR	Boiling Water Reactor
CAM	Continuous Air Monitor
CB	Control Building
CFD	Condensate Filters and Demineralizers System
CIS	Containment Inerting System
CMon	Containment Monitoring Subsystem
CNSC	Canadian Nuclear Safety Commission
CRD	Control Rod Drive
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
DBA	Design Basis Accident
DNNP	Darlington New Nuclear Project
EFU	Emergency Filter Unit
FMCRD	Fine Motion Control Rod Drive
FPC	Fuel Pool Cooling and Cleanup System
GEH	GE Hitachi Nuclear Energy
HVAC	Heating, Ventilation, and Air Conditioning
HVS	Heating Ventilation and Cooling System
HEPA	High Efficiency Particulate Air
ICRP	International Commission on Radiological Protection
ICS	Isolation Condenser System
ISI	In-Service Inspection
LOCA	Loss-of-Coolant Accident
LWM	Liquid Waste Management System
MCR	Main Control Room
OGS	Offgas System
OLC	Operational Limits and Conditions
OPEX	Operating Experience

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Acronym	Explanation
PAM	Post-Accident Monitoring
PLSA	Plant Services Area
POSAR	Pre-Operational Safety Analysis Report
PPE	Personal Protective Equipment
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PS	Process Sampling Subsystem
RB	Reactor Building
RHX	Regenerative Heat Exchanger
RPV	Reactor Pressure Vessel
RW	Radwaste
RWB	Radwaste Building
SC	Safety Class
SCCV	Steel-Plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SJAE	Steam Jet Air Ejector
TB	Turbine Building
URD	Utility Requirements Document
USNRC	U.S. Nuclear Regulatory Commission

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12.0 RADIATION PROTECTION

Administrative programs and procedures, in conjunction with facility design, ensure that the occupational radiation exposure to personnel is kept As Low As Reasonably Achievable (ALARA). The systematic application of the ALARA principle during the design phase of the BWRX-300 establishes the basic design criteria observed to reducing occupational exposure during plant operation and maintenance, decommissioning and post-accident ALARA.

ALARA design requirements are established to improve the layout of enclosures, accesses, and exits from controlled areas of the plant structures that confine radioactive material. The design of plant Structures, Systems, and Components (SSCs) minimizes personnel exposure to radiation during operation, inspection, maintenance, or plant design modifications.

The ALARA design requirements keep radiation exposures ALARA during normal operation or Anticipated Operational Occurrences (AOOs) and planned radioactive material releases below regulatory limits. The ALARA design criteria includes provisions for mitigating the radiological consequences of design basis accidents in accordance with Canadian Nuclear Safety Commission (CNSC) REGDOC-2.5.2, Section 4.1.1 (Reference 12.1-1).

The plant design:

- Precludes the release of radioactive material to the public and the environment that exceeds the limits of applicable regulations for normal operations, transients, and accidents
- Minimizes personnel exposure
- Minimizes the generation of radioactive contamination and waste

12.1 Optimization of Protection and Safety

12.1.1 Design and Construction Policies

ALARA is applied during the initial design of the plant and implemented via internal design reviews. The design is reviewed in detail for ALARA considerations and is reviewed, updated, and modified during the design phase to apply best practices operating experience. Engineers review the plant design and integrate the layout, shielding, ventilation and instrument monitoring designs with traffic control, security, access control, and health physics aspects to ensure the overall design maintains exposures ALARA.

All radioactive fluid pipe expected to contain significant radiation sources are designed with adequate shielding and properly routed to reduce personnel exposure Operating Experiences (OPEX) from industry that includes the operating Boiling Water Reactor (BWR) fleet are continuously integrated during the design phase.

12.1.2 Operational Policies

The ALARA principle is applied during the operation of the BWRX-300 throughout the completion of the plant life cycle by implementing the Radiation Protection Program discussed in Section 12.7.

12.1.3 Compliance With Standards for Radiation Protection to Ensure Occupational Exposures are ALARA

The BWRX-300 plant is designed for Radiation Protection to comply with Government of Canada Radiation Protection Regulations (Reference 12.1-2) and to meet the requirements and guidance in CNSC REGDOC-2.7.1 (Reference 12.1-3). Additionally, guidance from International Atomic

Energy Agency (IAEA) international safety standards, and U.S. Nuclear Regulatory Commission (USNRC) guidance informs the design.

12.1.4 Ensuring Occupation Radiation Exposures Remain ALARA

The BWRX-300 design considers that the majority of station personnel radiation exposure of is received during:

- Maintenance
- Radwaste (RW) work
- In-service inspection
- Refueling
- Non-routine operations
- Decommissioning

The BWRX-300 SSC that contain, collect, store, process, or transport radioactive materials follow the ALARA principle.

The BWRX-300 is designed to meet requirements in REGDOC 2.5.2, Section 8.13 (Reference 12.1-1). REGDOC-2.7.1, Section 4.1 (Reference 12.1-3) provides measures for the application of ALARA with reference to international standards for facility, equipment, and instrumentation design features delineated in Section 12.3. Cited standards for implementing and for maintaining ALARA include IAEA NS-G-1.13 (Reference 12.1-4) and International Commission on Radiological Protection (ICRP) standards such as ICRP Publication 101(b), (Reference 12.1-5) and ICRP Publication 103, (Reference 12.1-6).

The Radiation Protection Program discussed in Section 12.7 is implemented in accordance with REGDOC 2.7.1 (Reference 12.1-3) to ensure that radiation exposures to personnel and the public are ALARA, and that personnel are qualified, trained and monitored to keep exposures ALARA.

12.1.5 ALARA Design Considerations

This section describes the methods and features incorporated into the design that keep exposures ALARA.

12.1.5.1 General Design Considerations for ALARA Exposures

Consistent with the recommendations of IAEA and ICRP standards, general design considerations and methods employed to maintain in-plant radiation exposures ALARA, have two objectives:

- Reducing the necessity for and amount of personnel time spent in radiation areas
- Reducing radiation levels in routinely occupied plant areas in the vicinity of plant equipment that require personnel attention

Equipment and facility designs are evaluated for maintaining exposures ALARA during plant operations. Plant operations include normal maintenance and repairs, refueling operations and fuel storage, in-service inspection and calibrations, radioactive waste handling and disposal, and decommissioning.

Descriptions of general design features that maintain doses ALARA during normal power and shutdown operations are provided in Section 12.3.

Features that assist in maintaining low occupational exposures during decommissioning include:

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- Provisions for draining, flushing, and decontaminating equipment and piping
- Equipment design minimizes the radioactive material buildup and facilitates crud traps flushing
- Shielding protection during maintenance, repairs, or during decommissioning operations
- Provision for adequate space for movable shielding
- Separation of highly radioactive equipment from less radioactive equipment
- Provision for separate shielded compartments for adjacent items of radioactive equipment
- Provision for access hatches for the installation or removal of plant components
- Provision for the Reactor Water Cleanup (CUW) and Shutdown Cooling (SDC) System and the condensate demineralizer to minimize crud buildup

12.1.5.2 BWRX-300 Design Considerations

The following BWRX-300 design features minimize radioactive contamination:

1. Containment in areas where leaks and spills are likely to occur.
2. Leak detection capability to provide prompt detection SSC leakage.
3. Usage of leak detection methods (e.g., instrumentation, automated samplers) capable of early leak detection in areas difficult (inaccessible) to conduct regular inspections (such as the fuel pool), and buried, embedded, or subterranean piping) to avoid release of contamination. All BWRX-300 tanks containing radioactive fluids are within the Radwaste Building (RWB) that have cubicles and drain back into the radioactive liquid waste for processing.
4. Minimizing embedded piping, sumps, or buried equipment to facilitate decommissioning.
5. Removal or replacement of equipment or components during facility operation or decommissioning.
6. Minimizes the generation of radioactive contamination and waste during operation and decommissioning by reducing the volume of components and structures that become contaminated during plant operation.

12.1.5.3 ALARA Equipment Design Considerations

BWRX-300 engineering design procedures require the integration of international standards (including USNRC Regulatory Guide 8.8), and the radiation received for the component or system application.

12.1.5.3.1 Equipment Design Consideration to Limit Time Spent in Radiation Areas

BWR fleet OPEX is factored into current designs. Equipment instrumentation and controls are accessible during normal and abnormal operating conditions. Equipment such as the CUW and the Fuel Pool Cooling and Cleanup System (FPC) are remotely operated, including backwashing and precoat operations.

Equipment design facilitates maintenance. Equipment such as the Isolation Condenser System (ICS) heat exchanger is designed with an excess of tubes in order to permit tube plugging, when necessary. Heat exchanger drains exist on the shell side. Some valves have stem cartridge-type packing that is easily replaced. Refueling tools come equipped with drainage and have smooth surfaces that reduces contamination. Vessel and piping insulation is easily removed.

The materials selected are chosen for operating conditions. Valves use stem packing to reduce leakage and maintenance.

Equipment Design Consideration to Limit Component Radiation Levels

Equipment and piping reduce the accumulation of radioactive materials in the equipment. The piping, where possible, is constructed of seamless pipe as a means to reduce radioactivity accumulation on seams. The filter demineralizers in the CUW and FPC systems are backwashed and flushed prior to maintenance.

Equipment is designed for limiting leaks or controlling fluid leaks include piping the released fluid to the sumps and using drip pans with drains piped to the floor drains.

The primary coolant system materials selected consist mainly of austenitic stainless steel, carbon steel and low alloy steel components.

The system design includes a CUW system on the reactor coolant. This system is designed to limit the radioactive isotopes in the coolant by processing the coolant in the plant Condensate Filters and Demineralizers System (CFD) for cleanup. The feedwater piping downstream of the CFD is stainless steel with low cobalt content. Zinc is injected into the system to aid in reducing dose rates from Co-60 deposition.

12.1.5.4 Facility Layout General Design Considerations for Maintaining Radiation Exposures ALARA

Facility general design considerations minimizing personnel time spent in radiation areas include:

- Locating equipment, instruments, and sampling stations, that require routine maintenance, calibration, operation, or inspection, for ease of access and minimum required occupancy time in radiation areas
- Laying out plant areas allowing remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment
- Providing, where practicable, transportation of equipment or components requiring service to a lower radiation area
- Minimizing radiation levels in plant access areas and vicinity of equipment

Facility general design considerations directed toward minimizing radiation levels in plant access areas and in the vicinity of equipment requiring personnel attention include:

- Separating radiation sources and occupied areas where practicable (e.g., pipes or ducts containing potentially high radioactive fluids not passing through occupied areas)
- Providing adequate shielding between radiation sources and access and service areas
- Locating equipment, instruments, and sampling sites in the lowest practicable radiation zone
- Providing central control panels permitting remote operation of Safety Class (SC) instrumentation and controls from the lowest radiation zone practicable
- Separating highly radioactive equipment from less radioactive equipment, instruments, and controls
- Providing adequate space for movable shielding for sources within the service area
- Providing and facilitating contamination control

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- Providing service area decontamination
- Providing space for pumps and valves outside highly radioactive areas for maintenance and inspection
- Providing remotely operated centrifugal discharge and/or back-flushable filter systems for highly radioactive RW and cleanup systems
- Providing labyrinth entrances to radioactive pump, equipment, and valve rooms
- Providing adequate space in labyrinth entrances for easy access
- Maintaining ventilation airflow patterns from areas of lower radioactivity to areas of higher radioactivity

12.1.1 References

- 12.1-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 12.1-2 Government of Canada,, SOR/2000-203, Radiation Protection Regulations.
- 12.1-3 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."
- 12.1-4 IAEA Safety Standards Series No. NS-G-1.13, "Radiation Protection Aspects of Design for Nuclear Power Plants," International Atomic Energy Agency.
- 12.1-5 ICRP Publication 101(b), "The Optimization of Radiological Protection – Broadening the Process," International Commission on Radiological Protection.
- 12.1-6 ICRP Publication 103, "The 2007 Recommendations of the International Commission on Radiological Protection," International Commission on Radiological Protection.

12.2 Sources of Radiation

This section contains a qualitative description of radiation sources present in the BWRX-300 plant. These sources are identified based upon GEH BWR design evolution and fleet operations.

At this stage of the plant design, source strengths have not been calculated. The characterization of plant sources is limited to source descriptions. These sources are quantified in the Pre-Operational Safety Analysis Report (POSAR).

12.2.1 Contained and Immobile Sources of Radioactive Material

12.2.1.1 Containment Radiation Sources

This section describes the significant radiation sources found inside the BWRX-300 containment. The calculated sources provided in the POSAR consist of those elements that contain significant quantities of radioactive elements. The radiation source does not include incidental contamination, such as sources in valves (deposition from corrosion), fission product species on component surfaces, or gamma sources emitted from pipes containing radioactive fluids. The Fine Motion Control Rod Drive (FMCRD) system provides the only other notable source of radiation in containment. The only significant source of post-operation radiation in containment is the reactor core and irradiated reactor internals.

Reactor Vessel Core Sources

A reactor vessel model and pertinent data necessary to calculate neutron and gamma fluxes inside and outside of the reactor core during normal operation is defined. Ex-core particle fluxes from the reactor core during operation calculations require a detailed analysis of neutron particle transport and using a deterministic solution to the Boltzmann equation or probabilistic modeling techniques. The primary source for the neutron and gamma fluxes outside the core is the fission process. Gammas are also created by the decay of fission products, and secondary gammas resulting from neutron absorption and scattering in structural materials, both inside and outside of the core. Nuclide cross-section libraries contain gamma production data for all of these sources. As a result, it is only necessary to define the neutron fission source in the core, and then perform a coupled neutron-gamma transport calculation.

After shutdown, the neutron fluxes are negligible, and N-16 activity quickly decays to zero. The most significant radiation source is the gammas resulting from fission product decay in the reactor core. Neutron and gamma fluxes inside and outside the reactor core during normal operation are provided in the POSAR that include neutron and gamma flux results, reactor vessel model definition, and all pertinent data used in the analysis for the reactor during operation and shutdown conditions.

Physical Data

A POSAR table presents the physical data required to form a model that provides sufficient regions to adequately represent the reactor. The in-core region is divided into a determined number of axial nodes, with one radial node per fuel bundle. A unique neutron fission source is determined for each of these nodes using the nodal cycle average power and exposure data. Water densities are determined at each of the determined planes for peripheral bundles and in-core bundles. A POSAR table provides nominal dimensions and material volume fractions for each boundary and region in the reactor model with core average data presented for the core. To describe the reactor core, thermal power, power density, core dimensions, core average material volume fractions, and cycle average reactor power distributions and exposures are provided. Reactor power distributions are given for both radial and axial distributions and represent cycle averages for an equilibrium cycle.

Core Boundary and Vessel Neutron Fluxes

A POSAR table presents multigroup neutron fluxes at the representative location of the core boundary and at the vessel. The multigroup neutron fluxes and the fast neutron flux ($E > 1$ MeV) at the peak elevation of the core boundary, vessel inside surface, and a determined thickness of the vessel are presented the POSAR. The uncertainty of the fast neutron flux at the vessel is estimated within a determined percentage. Normalized axial variations for the fast flux at the vessel inside surface are shown in another POSAR table.

Gamma Ray Source Energy Spectra

The average gamma ray source energy spectra in both core and non-core regions is presented in a POSAR table. The energy spectrum in the core, bypass water, shroud, downcomer, and Reactor Pressure Vessel (RPV) is presented. This represents the average gamma ray energy released by energy group per unit volume of the region. The energy spectra in MeV per seconds per cm^3 is used with the power distributions to obtain the source in any part of the core.

The gamma ray energy spectrum includes the fission gamma rays, the fission product gamma rays, and the gamma rays resulting from inelastic neutron scattering and neutron capture. The total gamma ray energy released in the core is estimated accurate within a conservative percentage.

Post-Operation Gamma Sources

A gamma ray energy spectrum in MeV/sec per MW thermal in spent fuel as a function of time after operation is provided in a POSAR table. The data is prepared from the irradiation and decay calculation of a representative BWRX-300 fuel bundle average exposure. To obtain shutdown sources in the core, the gamma ray energy spectra are combined with the core thermal power and power distributions. Shutdown sources in a single fuel bundle is obtained by using the gamma ray energy spectra and the thermal power of the element during operation. Similar information suitable for the spent fuel pool fuel bundles with an average exposure is also provided in the POSAR.

Neutron and Gamma Ray Fluxes Outside the Vessel

The maximum axial neutron and gamma ray fluxes outside the vessel is provided in the POSAR. The maximum axial flux occurs near the core midplane elevation where the maximum power density is located for the peripheral bundle. This elevation is determined using the data for the maximum axial neutron and gamma ray fluxes outside the vessel. The fluxes at this elevation represent the fluxes at the peak azimuth angle. The gamma ray calculations include gamma ray sources from regions inside the vessel and the vessel itself.

Control Rod Drive System Sources

The Control Rod Drive (CRD) source term data is provided in the POSAR.

Historically, removal and replacement of CRDs is a significant contributor to occupational exposures. Improved maintainability associated with the FMCRDs design reduces personnel exposures compared to conventional BWR CRDs. The CRD system is comprised of three major elements: (a) the FMCRD mechanisms; (b) the Hydraulic Control Unit assemblies; and (c) the CRD hydraulic subsystem. The BWRX-300 CRD hydraulic subsystem supplies high-pressure demineralized water that is regulated and distributed to provide charging of the Hydraulic Control Unit scram accumulators and purge water flow to the FMCRDs. The CRD system is described in Section 4.5.

Reactor Startup Source

The Cf-252 reactor startup source is shipped to the site in a special shielded cask. The source is transferred under water while in the cask and loaded into a stainless steel source holder. This is loaded into the reactor while remaining under water. The source and source holder are removed from the reactor during the first refueling outage and moved to a designated location in the fuel pool. Operations and Radiation Protection personnel determine placement and duration of residence for the Cf-252 source and holder in the fuel pool. The activity requirements for the reactor startup source are provided in the POSAR

12.2.1.2 Reactor Building and Plant Services Area Radiation Sources

The Reactor Building (RB) radiation source terms consist of significant quantities of radioactive elements from the fission source, but do not include sources due to incidental contamination such as sources in valves and pipes due to deposition of corrosion or fission products species on the surfaces of the components. Water radioactive source concentrations in the fuel pool are assumed as 1% of normal reactor water concentrations. The Plant Services Area includes the RB auxiliary bay where new and spent fuel enter and leave the RB. Source terms in the RB are identified in the POSAR.

Fuel Pool Cooling and Cleanup System

The FPC system is designed to service the fuel pool, cask pit, and reactor cavity pool on a rotating basis. The activity in this system is the result of the accumulation of residual activity in each of these pools. The filters are backwashed and the backwash routed to the liquid RW systems for processing. Clean water connections are provided for this system to flush lines prior to switching between pools to prevent ancillary contamination between pools.

Fuel Pool

The gamma source energy spectra from spent fuel in the fuel pool is significant; however, the spent fuel is submerged in water and shielding around the fuel pool is provided ALARA. Radiation source terms from the fuel pool is provided in the POSAR.

Reactor Building and Containment Post-Accident Radiation Source

The BWRX-300 design limits potential radiation exposure from accidents to personnel and to the public in the plant using passive safety features, containment, and treatment of potential normal operation and accident sources. The following describes those features of the BWRX-300 germane to post-accident radiation sources in containment and the RB.

1. Containment is a nitrogen-inerted, Steel-Plate Composite Containment Vessel (SCCV) pressure boundary capable of containing all design basis accident sources with minimal leakage to the environment or other plant areas. The containment is provided with redundant passive cooling systems to ensure that this primary boundary does not exceed design criteria. Radioactive sources from the pressure vessel are adequately contained during design basis accidents and some beyond design basis accidents.
2. Surrounding the cylindrical containment, the BWRX-300 employs a RB that provides a secondary holdup volume to trap containment penetration and valve leakage. All major connections from containment (e.g., main steam, feedwater) include an isolation valve in the RB. Pipe ruptures within containment are mitigated by the RPV isolation valves for large and intermediate pipe ruptures, and the coolant released that becomes airborne is captured in the containment airspace. Any releases from potential ruptures outside of containment but inside the RB are isolated by the RPV isolation valves.

12.2.1.3 Turbine Building Source Terms

Significant radioactive source terms in the BWRX-300 Turbine Building (TB) consist of those elements that contain significant quantities of radioactive materials. This does not include sources due to incidental contamination such as valves with corrosion deposition or fission products species on component surfaces. The following systems that may contain radioactive material sources are located in the TB.

Reactor Water Cleanup System (CUW) Radiation Sources

The CUW system is located in the TB in a shielded area containing other radioactive equipment and systems. Radioactive sources contained in the CUW result from component contamination by reactor water through the system and accumulation of radioisotopes removed from the water. Components for this system include a Regenerative Heat Exchanger (RHX), pressure reduction station, and valves. The RHX is fitted with decontamination ports. The CUW system does not have filters and demineralizer beds; however, the CFD provides all of the cleanup of the reactor coolant.

Heating, Ventilation and Cooling System

The Heating Ventilation and Cooling System (HVS) employs High Efficiency Particulate Air (HEPA) filters for use in the event of airborne contamination of the RB or controlled purge of containment. The HEPA filters, which can accumulate radioactive sources, are shielded to ensure the radiation zones are maintained ALARA in areas adjacent to the filter housings.

Main Steam and Feedwater Lines

All radioactive material in the main steam system results from radioactive sources carried over from the reactor core during plant operations. In most components carrying live steam, N-16 is the dominant source of radioactivity. Otherwise, under conditions where sufficient decay time has removed the N-16 source, noble radioactive gases become the dominant source term.

Normal Operating Sources

N-16 in the steam flow from the pressure vessel is the primary TB source of radioactivity. The N-16 source results in significant gamma shine from the main steam lines and steam bearing components. Other sources of radiation in the TB are the Offgas System (OGS) and CUW. The OGS consists of the recombiner, OGS condenser, and offgas charcoal tanks (located in the RWB), and the CUW includes an RHX, pressure reduction station, and various valves.

Post-Accident Sources

The TB contains no major post-accident sources of releasable radioactivity (discounting N-16 because of the short half-life) and potential releases are limited to liquid releases of low activity water. Two other sources exist that contain radioactive species, but in a form not amenable for release. The potential for accident releases from these two sources, the OGS, and the CUW system, is reduced due to heavy shielding and compartmentalizing of the components.

12.2.1.4 Radwaste Building Sources

This subsection provides brief descriptions of the significant radioactive source terms found in the BWRX-300 RWB. These source terms consist of those elements that contain significant quantities of radioactive materials but do not include sources due to incidental contamination, such as sources in valve corrosion deposition or fission product species on the surfaces of the components.

Normal Operating Sources

Major RWB components that contain radiation sources include liquid RW collection tanks, spent resin tanks, sludge tanks, liquid waste sample tanks, filtration equipment, and high integrity containers with solid waste. These contained sources will be quantified as the design progresses. These sources are based on the waste stream concentrations fed to the RW systems and represent sources for shielding calculations.

Post-Accident Sources

Potential releases in the RWB are contained by isolating the RWB atmosphere and sealing any water releases in the building. The RWB is seismically designed, and the concrete tank areas are provided with sealant and steel liners to prevent any releases from high-activity areas.

12.2.2 Sources of Airborne Radioactive Material

12.2.2.1 Reactor Building

The RB HVS is discussed in Subsection 9A.5.1. Subsection 12.2.1.3 discusses the radiation control aspects of the HVS.

Normal Operation

The main source of airborne activity in the RB is primary coolant leakage. The highest potential activity sources are in containment but pose no threat to plant personnel during normal operation.

The containment cooling system conditions and circulates air through the contaminated areas of the building. Flow is directed from the corridors (point of highest pressure) to the equipment alcove rooms, the rooms themselves, and finally to the external wall pipe chases and from the pipe chases back to the HVS.

Access into the containment is not permitted during normal operation. The ventilation system inside merely circulates the air, without filtering it. The only airflow out of containment into accessible areas is minor potential leakage through penetrations. During maintenance, the containment atmosphere is purged using the containment inerting system and replaced with clean air before access is permitted.

Reactor Building Airborne Sources During Refueling

BWR operating experience shows that airborne radioactivity is generated from the steam dryer and separator if their surfaces are allowed to dry. Other potential airborne sources occur during vessel head venting and fuel movement. The airborne radioactive material sources resulting from reactor vessel head removal are minimized by venting to the containment cooling system or to the main condenser prior to removal. The contribution to the airborne radioactivity due to the reactor vessel internals is minimized because they are wet and submerged.

Airborne radioactivity during refueling is similar or lower due to improved design features and OPEX incorporation. Experience shows that airborne radioactivity results from water in the reactor cavity exceeding 38°C (100°F) and flaking of cobalt dioxide (CoO₂) from the steam dryer and separator if their surfaces are allowed to dry. Other potential airborne sources resulting from reactor vessel head and internals removal are determined from OPEX. OPEX shows that Iodine-131, Co-60, Mn-54, Nb-95, Zr-95, Ru-103, and Ce-144 are the major radioisotopes found with Ce-141, Cs-137, Co-58, and Cr-51 at lower concentrations.

To minimize airborne radioactivity, the following actions are specified:

- Maintain steam dryer and separator surfaces wet or covered
- Cool fuel pool through large heat capacity heat exchangers

- Fuel pool ventilation system designed to sweep the pool surface and prevent pool releases from mixing with the area atmosphere

Refueling Area of Reactor Building

The refueling area is serviced by the containment cooling system, including radiation control aspects of the system is discussed in Chapter 9A, Subsection 9A.5.6.

The airborne activity source in the fuel pool area of the RB is in the spent fuel storage pool and equipment areas. The ventilation system is designed to sweep air from the fuel pool surface removing the major portion of potential airborne contamination. In addition, evaporation from the fuel pool is minimized by pool cooling.

The assumptions and parameters used to determine the airborne activity levels in the spent fuel storage pool and equipment areas, and the airborne concentrations are provided in the POSAR.

12.2.2.2 Turbine Building Airborne Sources

The main potential source of airborne radioactivity within the TB is leakage from valves on large steam pipes. The design provides for collection of this leakage and its transport back to the condenser. Therefore, noble gas airborne concentrations are negligible throughout the TB except for inside the Steam Jet Air Ejector (SJAE) cubicles. These areas are not normally occupied during operation, and the exhaust from these cubicles is exhausted to the environment after filtration eliminating the possibility of contamination of adjoining areas. Another airborne activity source is equipment leakage in the TB atmosphere.

12.2.2.3 Radwaste Building Airborne Sources

The RWB HVS is discussed in Chapter 9A, Subsection 9A.5.3.

RWB corridors and routine access operating areas do not have significant airborne radioactivity levels. Equipment cubicles are infrequently accessed and may contain low levels of airborne radioactivity, but design provisions minimize the release of radioactivity.

RWB tanks are filled from the top and as the water splashes into the tanks, dissolved and entrained radioactivity may become airborne. This activity is not released into the atmosphere in the rooms because the tank vents are connected directly to the building ventilation system. Pumps and valves for radioactive systems in the RWB are in separate compartments that are not normally accessed by operating personnel. The RWB ventilation design provides airflow from areas of low potential for airborne contamination to areas of increasing potential. This ensures that any leakage from RW pumps and valves is not directed into normally occupied areas of the building but exhausted from the building.

12.2.2.4 Control Building Airborne Sources

The Control Building (CB) HVS is discussed in Chapter 9, Subsection 9A.5.2. If the CB air intake radiation monitor setpoint is reached, the intake is isolated, and the system initiates the recirculation mode. The SCR is available if the MCR becomes uninhabitable. There are no contaminated areas in the CB.

12.2.2.5 Plant Services Area

The Plant Services Area (PLSA) contains the Reactor Auxiliary Bay where new and irradiated fuel is moved in and out of the reactor building.

12.2.3 Airborne Sources for Within the Plant

The design keeps all radioactive material contained. Leaks from process systems, refueling, and decontamination may lead to airborne radioactivity. Equipment cubicles, corridors, and areas

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routinely occupied by operating personnel do not contain significant airborne radioactivity sources and comply with the limits specified in SOR/2000-203 (Reference 12.2-2). Radioactive equipment that potentially leaks is installed in separate shielded compartments that are not routinely occupied.

In general, airflow within the building ventilation systems is from areas of low potential for airborne contamination to areas of increasing potential. Routinely occupied areas are maintained at low airborne radioactivity levels. OPEX data from operating BWRs corroborate the general lack of airborne activity in corridors and routinely occupied operating areas. Air samples and surface contamination swipe samples are performed to verify the absence of airborne and surface contamination.

Process leakage results in potential release of noble gases and other volatile fission products via ventilation systems. Leakage of fluids from the process system results in the release of radionuclides into plant buildings. In general, the noble radiogases remain airborne and are released to the atmosphere with little delay via ventilation exhaust ducts. Other radionuclides partition between air and water may plate-out on metal surfaces, concrete, and paint. Radioiodines are found in ventilation air as methyl iodide and as inorganic iodine (particulate, elemental, and hypoiodous acid forms).

12.3 Design Features for Radiation Protection

The design features incorporated in the BWRX-300 are based on GEH BWR OPEX. The BWRX-300 ALARA design requirements incorporate the guidance of REGDOC-2.7.1 Radiation Protection (Reference 12.3-1).

12.3.1 Facility and Equipment Design Features

The BWRX-300 plant design features ensure plant personnel radiation exposures are ALARA. General design considerations and methods to maintain in-plant radiation exposures ALARA are:

- Minimizing the necessity for and amount of personnel time spent in radiation areas
- Minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment that require personnel attention

The BWRX-300 is designed to ensure the radiation exposure to any nuclear energy worker is less than the limits specified in the SOR/2000-203 (Reference 12.3-2).

12.3.1.1 Radiation Source Term Reduction

Radiation source term reduction design considerations:

1. N-16 skyshine dose contribution is determined from N-16 activity in the vessel nozzle outlet steam based upon the hydrogen water chemistry.
2. Noble gas airborne concentrations are negligible throughout the TB except for inside the SJAE cubicles. The process air from these cubicles is exhausted to the environment after HEPA filtration and noble gas decay in activated carbon beds in the gaseous RW system reducing contamination in adjoining areas.
3. Minimizing airborne radioactivity includes:
 - a. Storing underwater the steam dryer, separator, and chimney during refueling operations
 - b. Cooling the RB pools with large heat capacity heat exchangers
 - c. Sweeping the spent fuel pool surface and preventing evaporative pool losses and gases from mixing with the area atmosphere using the FPC
4. Radioactive material is kept in containers to the extent possible.
5. Equipment cubicles, corridors, and areas routinely occupied by operating personnel do not contain significant airborne radioactivity sources. Radioactive equipment that potentially leaks is installed in separate shielded compartments.
6. Material selection minimizes the creation of activated corrosion products.
7. Zinc injection minimizes dose rate from Co-60 deposition.

12.3.2 ALARA Design Guidelines for Mechanical System and Components

This section describes the incorporation of the ALARA principle in the BWRX-300 systems and components design complying with REGDOC-2.7.1, Section 4.1 ALARA (Reference 12.3-1).

12.3.2.1 General Mechanical Design Guidelines

1. Shielding, remotely operated valves, and sample transporting casks reduce radiation exposure.

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2. Equipment is designed for normal and abnormal operating conditions while maintaining radiation exposures ALARA.
3. Passive equipment is used where possible, thus minimizing worker exposure.
4. Process equipment that accumulates radiation sources from filtering process streams, such as the condensate and feedwater and the fuel pool cooling systems, are remotely operated, including backwashing operations.
5. Equipment facilitates maintenance.
6. Refueling tools have smooth surfaces and are easily drained to reduce contamination.
7. Vessel and piping insulation is easily removed and restored.
8. Equipment and piping minimize crud buildup and provide easy decontamination and maintenance.
9. Remotely operated mechanical devices are used to inspect the RPV body and nozzle welds.
10. In-Service Inspection (ISI) laydown space accommodates insulation removal.
11. These services are in close proximity to equipment facilitating maintenance and inspection:
 - a. Electrical outlets
 - b. Welding outlets
 - c. Condensate grade and/or demineralized water
 - d. Compressed air
 - e. Communications
12. For areas where the dose rate is 50 $\mu\text{Sv/hr}$ (5 mrem/hr) or higher where worker access is limited due to the radiation zone (as described in Section 12.6), or where equipment is removed for maintenance, the equipment design provides rapid removal using "plug in" or "quick disconnect" piping connections, electrical leads, instrumentation connections, and support linkages.
13. Passageways are provided for access ease where equipment is removed and replaced for maintenance by plant personnel wearing full Personnel Protective Equipment (PPE).

12.3.2.1.1 Mechanical Components

Tanks

1. Overflow pipes are connected to liquid radioactive waste drain system for processing. Vents run to the building ventilation exhaust system.
2. To facilitate decontamination, corrosion-resistant materials and linings prevent corrosion products buildup on the walls.
3. Walls or curbs and drains to the liquid radioactive waste system are provided around the tank for leakage collection. Room collection capacity is sized to contain the tank volume.
4. Spent resin, phase separator, and backwash tanks are located in enclosures free of flammable or combustible substances.
5. Bottom valves on slurry tanks are flush-mounted.

Heat Exchangers in Radioactive Service

1. Tube and shell heat exchangers are designed with excess tubes accommodating tube plugging.
2. Where feasible, leakage from highly contaminated fluids is prevented from entering clean fluids by maintaining a higher pressure in the clean fluid sided.
3. Heat exchanger design allows for fluid drainage avoiding pooling that may lead to radioactive crud deposition.
4. Heat exchanger drains are hard piped to floor drains.
5. Cleaning connections are provided for condensate, demineralized water, and/or chemical solutions.

Piping

1. Piping selection provides a service life equivalent to the design life of the plant, with corrosion and environmental condition allowances.
2. Contaminated piping systems are welded to the extent possible.
3. Piping in highly radioactive systems such as the spent fuel pool cooling system have butt-welded connections, rather than socket welds, reducing crud traps.
4. Radioactive piping is sized to produce turbulent flow to maintain solids suspension without plugging.
5. In systems where contamination settling can occur, the drains are placed on the side of the piping.
6. Systems that contain multiple pumps are placed in separate alcoves with piping routed into shielded pipe chases.
7. Radioactive piping minimizes plant personnel radiation exposure as follows:
 - a. Radioactive pipe routing in corridors is minimized
 - b. Shielded pipe trenches and pipe chases are used for routing high-activity pipes in low-level radiation areas
 - c. Separating radioactive and non-radioactive pipe routes for maintenance
 - d. Piping is arranged based upon SC, grouping of components, system operability, and maintenance access
 - e. Non-radioactive line systems are not run inside biologically shielded compartments except where needed to supply services inside the compartment
 - f. Piping configurations minimize the number of "dead legs" and low points to avoid radioactive crud and fluids accumulation
8. Systems are designed with vents and flushing lines reducing crud buildup.
9. Instrument lines, except those for the reactor vessel, are designed for backflushing and maintaining a clean fill in the sensing lines. The reactor vessel sensing lines are flushed with condensate from the instrument rack when the reactor is depressurized.
10. Piping conveying highly contaminated fluids is routed through shielded pipe chases and shielded cubicles. When these options are not feasible, the radioactive pressurized

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process piping is sleeved embedded in concrete walls and floors. Embedded pipes connected to floor drains are butt-welded stainless steel.

11. Contaminated piping design facilitates cleaning and removal during decommissioning.
12. "Clean" services, such as compressed air and demineralized water are routed through shielded pipe chases.
13. Where piping penetrates walls the use of embedded piping is minimized facilitating system dismantling and plant decommissioning.
14. When radioactive and non-radioactive pipes are collocated, drain provisions are provided for removing radioactive fluids, and the valves are remotely controlled.
15. Piping penetrations through shield walls minimize the shine on surrounding areas.
16. Underground piping is installed for clean service only. If not possible, it is routed with a guard pipe, trench or tunnel for visual inspection and monitored leakage. Threaded and flanged connections are kept to a minimum. Other joints are welded or permanently bonded depending on the piping material.

Ion Exchange and Activated Carbon Beds and Filters

1. Filters in purification and cleaning systems are operated and cleaned remotely.
2. For purification systems, preparation, and resin changeout are performed remotely.
3. A mesh trapping resins or activated carbon particles is installed downstream of vessels.
4. Sampling points are installed at the inlet and outlet of, ion exchange vessels and activated carbon beds.
5. Cartridge-type filters or duplex filters with on-line exchangers are sized for one fuel cycle without changeout.

Pumps and Valves in Radioactive Service

1. Pumps located in radiation areas are designed with features to minimize maintenance time: quick-change cartridge-type seals, back pullout features that permit removal of the pump impeller, or mechanical seals without disassembly of attached piping.
2. Where two or more pumps conveying highly radioactive fluids are located adjacently, shielding is provided between them maintaining exposure levels ALARA. Pumps adjacent to other highly radioactive equipment are shielded to reduce personnel exposure.
3. Pump control instrumentation for process equipment is located outside high radiation areas in separate alcoves. Motor or pneumatic-operated valves or valve extension stems are used allowing operation.
4. Radioactive system pumps and valves are remotely controlled.
5. Pump casing drains are provided on radioactive systems and drained to the liquid radioactive waste system for processing.
6. Valves have cartridge-type packing that is easily replaced, or diaphragm valves or quarter-turn ball valves are used that do not require similar packing.
7. Instrumentation and valves in high radiation areas are remotely operated.
8. Straight-through valves are selected over those that exhibit flow discontinuities or internal crevices minimizing crud trapping.

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9. Valves are placed in horizontal runs to facilitate dismantling.
10. For systems in high radiation areas, gasket materials are selected that inhibit degradation. Degradation mechanisms (i.e., stress corrosion, thermal aging, embrittlement, fatigue, thermal fatigue etc.) are considered during the material selection process.
11. Valves containing radioactive fluids are separated from "clean" services to reduce the radiation exposure from adjacent valves and piping.
12. Safety and relief valves and depressurization valves are designed with flange connections to allow whole valve removal.
13. Isolation valves are located so that fluid contained in dead branch-pipes is minimized.
14. Back seat valves are used reducing leaks through the packing when the valve is full open.
15. Pipe insulation around valves is easily removed.

Radioactive Drains

1. For systems containing highly radioactive fluids, frequently used drains are hard piped to the nearest drain. An air gap exists between the system drain and the floor or equipment drain hub. For drains used less frequently, a drain connection with a valve is provided so plant personnel can connect a hose and direct the outflow to the nearest drain.
2. Sump vents are located near the room exhaust vent register to control airborne radioactivity released from sump discharges.
3. Sumps are stainless steel-lined reducing corrosion and providing easily decontaminated surfaces.
4. Pipes carrying fluid to radioactive drains and clean drains are separate and independent.
5. The high-activity level resin slurry drains are separated from other drains and the contents routed to liquid radioactive waste for subsequent treatment.
6. Drain lines are gravity-sloped.
7. Drain lines containing highly radioactive fluids are routed through pipe chases or shielded cubicles.
8. One piece floor drains minimizing possibility of liquid penetrating at embedment boundaries. Non-porous alternatives to grout such as epoxy are used for the installation of floor drains.
9. Drain lines contain traps that prevent the flow of radioactive gas from compartment to compartment.
10. The drains from rooms that house spent resin tanks, phase separator tanks, or demineralizer vessels are equipped with normally closed isolation valves.
11. Sumps are located at the lowest level of the buildings.

Sludge and Resin Systems

System design and implementation prevent accumulating and retaining radioactive sludge and resins in transfer pipes by:

1. Pipe runs are reduced.
2. Low points and blind sections are avoided.
3. Appropriate pipe slope to support gravity flow is used.

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4. Pipe flow restrictions are avoided.
5. Short 90° elbows are avoided and replaced by large-radius curved sections at least five times the diameter, or two 45° elbows spaced apart.
6. Whenever T connections are required, the main flow runs through the straight section. If the secondary flow is into the T, then the connection is made above the axis of the pipe. If the secondary flow is out of the T, then the connection is on the lateral connector side.

12.3.2.1.2 Materials

1. The reactor coolant system materials selected prevent corrosion product formation. Reactor internal components, except for the zircaloy in the reactor core, are stress corrosion-resistant stainless steels or other high alloy steels.
2. Carbon steel used in systems processing or storing reactor coolant or steam are low in nickel content and contain small amounts of cobalt impurity.
3. Main condenser tubes and closed loop cooling heat exchangers cooled by the plant cooling water tubes and their tube sheets are low-cobalt stainless steel or titanium alloys and low-cobalt materials.
4. Ni-Cr-Fe alloys such as Inconel 600 and Inconel X750 that have high nickel content, are used in some reactor vessel internal components. These materials are used in applications where special requirements (possessing specific thermal expansion characteristics along with adequate corrosion resistance) are necessary, and where no suitable alternative low-nickel material is available. Cobalt content in the Inconel X750 used in the fuel assemblies is limited to 0.05%.
5. The BWRX-300 limits the use of cobalt-bearing materials on moving components that are historically identified as major sources of reactor coolant contamination. Stellite is used for hard facing of components that must be extremely wear resistant. Use of high cobalt alloys such as Stellite is restricted to those applications where no satisfactory alternative material is available. Alternative materials (Colmonoy® and NOREM®) are used for some hard facings.
6. Control Blade Materials: All control rod materials are less than 0.03 weight percent cobalt. The average niobium content for the handle and absorber section, less boron carbide and hafnium, is less than 0.1 weight percent.
7. Non-stainless steel equipment has an adequate finish and is protected with a non-corrosive material aiding in decontamination for some applications.
8. For systems in contact with reactor coolant (pumps, valves, etc.), component materials are as degradation-mechanism resistant as possible. Degradation mechanisms (i.e., stress corrosion, thermal aging, embrittlement, fatigue, thermal fatigue etc.) are considered during the material selection process.
9. System components are made of materials that are qualified to withstand pressure, temperature, and radiation minimizing maintenance.
10. Pipes are designed for a 60-year life, based on the environmental conditions and corrosion preventing maintenance, unless documented otherwise (e.g., abrasive slurry lines). Pipes with less than a 60-year life are replaceable (i.e., not embedded in concrete).
11. Long-life materials that withstand the operating pressure, temperature and radiation are used for component joints and seals minimizing maintenance.

12. Heat exchangers are constructed of stainless steel or titanium tubes minimizing failure and reducing maintenance requirements.

12.3.2.2 Mechanical Systems, Equipment and Components Arrangements

12.3.2.2.1 Confinement of Systems

1. Radioactive systems are isolated from non-radioactive systems by confining them in different enclosures. Adequate shielding is provided between both areas.
2. Clean systems (non-radioactive) are in areas with no radioactivity or with low background radioactivity levels.
3. Piping and pumps or systems that have components that are major sources of radiation are in separate cubicles to reduce exposure during maintenance.
4. The various redundancies of a radioactive system that are provided for the same function are in separate areas and shielded from each other.
5. As a consequence of normal steam and water leakage into containment, airborne radioactive contamination exists during normal operation. Containment atmosphere purging to the environment is via the containment purge system, that is routed and processed through HEPA filters then to the TB plant stack.

12.3.2.2.2 Confinement of Equipment

1. Radioactive system equipment with different radiation levels are in different rooms.
2. System equipment that accumulates or contains radioactive fluids (tanks, heat exchangers) are isolated from other equipment (pumps, valves) and located in different rooms.
3. System equipment that concentrates radionuclides (particulate filters, and demineralizers) are isolated from the rest of the equipment in the system and located in different rooms or shielded.

12.3.2.2.3 Location of Components

1. Radioactive piping is accessible and arranged allowing inspection and maintenance with minimum exposure to personnel.
2. The location of equipment is optimized to minimize the length of the radioactive fluid ducts between them.
3. Functionally different radioactive system components such as control valves and pumps that require frequent maintenance or inspection, are in different rooms, to avoid personnel exposure.
4. Radioactive equipment is located as far as practicable from personnel dwelling and passage areas.
5. Labyrinths are used in radioactive equipment room accesses so that personnel are not exposed directly in corridors or staircases. Double labyrinths are used to shield high intensity radiation sources. Sealed cubicles with hatches or shield doors are used to shield very high radiation sources.
6. Intense radiation source equipment rooms are arranged so that workers are not exposed to the source while in the room.
7. Rooms house piping associated with the functional system.

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8. Radioactive pipes are run inside trenches or shielded pipe boxes whenever the run is through a low-level radiation area.
9. Radioactive pipes are grouped on the opposite side of the entrance to the room and away from components that require frequent maintenance.
10. Radioactive pipes are sloped unless there are drains or purges at low points.
11. Equipment startup, handling devices, and actuation mechanisms are in low activity areas.
12. Sampling equipment, instrumentation, and stations requiring routine attention (maintenance, calibration, actuation, reading or inspection) are in places that are easily and quickly accessible (next to the entrance to the enclosure) and in low activity areas.

12.3.2.2.3 Auxiliary Spaces

1. Remotely operated, back-flushable filter systems are provided for radioactive RW and cleanup systems.
2. The following auxiliary spaces prevent contamination dispersion:
 - a. Posted areas for storage of contaminated parts and equipment before they become decontaminated
 - b. Areas where equipment is decontaminated before repair. (e.g., the PLSA maintenance rooms that may be used for disassembling pumps, motors, etc.)
3. Decontamination areas have condensate quality water available to carry out preliminary decontamination of equipment segregated for repair.

12.3.2.3 Minimization of Crud Buildup

The crud buildup minimization facilitates decontamination and cleaning equipment, components, and contaminated areas, and minimizes airborne contamination dispersion.

Flushing provisions are made for equipment for minimizing crud buildup.

12.3.3 In-Service Maintenance and Inspection Provisions

12.3.3.1 Layout of Components, Space and Access Provisions

The following provisions prevent or minimize radiation exposure:

1. Adequate means are provided to monitor and control access to medium and high-activity areas (These are radiation zones C to I, inclusive as described in 12.6).
2. Equipment and component laydown areas have adequate space for dismantling, repair, removal, and insulation restoration for inspection or maintenance. Adequate access is also provided for transporting failed and replacement components and maintenance tools.
3. The welds in the reactor cooling circuit and in the pipes connected to it requiring ISI are readily accessible.
4. Sufficient space is provided to store and use temporary shielding.

12.3.3.2 Staircase and Platform Requirements

The following design requirements minimize time spent in radiation areas:

1. Stairs are provided and stepladders or ladders are avoided.
2. Stairs are sized for wearing protective gear and carrying necessary tools.

3. Platforms and staircases used to access components requiring frequent inspection or maintenance are permanent structures sized with sufficient space and capacity to withstand the component size and weight and additional mobile shielding.

12.3.3.3 Hoisting and Transport Mechanisms

The following provisions limit the operations and exposure duration:

1. Space is available for installing and hoisting transport mechanisms such as cranes, hoists, monorails, hand rigging, and trolleys used during periodic inspection and maintenance. Monorail beams are permanently installed over heavy equipment (equipment weighing one metric ton or more).
2. Hoisting/transport mechanisms are provided to handle heavy equipment, component parts, slabs, and heavy removable structural elements.
3. Hoisting/transport mechanisms are provided to handle parts and components in decontamination areas, in contaminated storage areas and active workshops
4. Remotely operated electric hoists are used for tasks/equipment associated with general area dose rates in excess of 1 mSv/hr (100 mrem/hr) where access is limited to one hour per week (In alignment with radiation zone E as described in 12.6).
5. Embedments are provided, in walls, floors, and overhead facilities for special rigging that requires scheduled maintenance. The design load capacity is marked on each embedment.

12.3.4 Room Leak-Tightness and Drain Provisions

The following measures prevent contamination dispersion and radiation passing through holes in walls and structures:

Leak-Tightness

1. House equipment enclosures with large radioactive liquid volumes (tanks, heat exchangers, filters, etc.) are leak tight. Additionally, containment measures are provided to prevent dispersion in the event of liquid overflow (curbs, sloping floor drains).
2. Gaseous radioactive waste system is leak tight minimizing radioactive gases and potentially flammable hydrogen escape.
3. Penetrations in walls and structures separating enclosures and buildings are located and sealed to prevent fluid contamination if a break or leak develops.

Drains

1. Enclosures that contain equipment used to store radioactive liquids have drains for spills or failures.
2. Sample station effluents are returned to the process stream or to the liquid radioactive waste through a common return line.

12.3.5 General Considerations of Systems that Process Radioactive Fluids

12.3.5.1 System Isolation

1. When a contaminated or potentially contaminated system is connected to a clean (i.e., non-contaminated) system, provisions are implemented ensuring backflow cannot occur. Backflow prevention is provided by double isolation and double check valves that permit periodic sampling ensuring clean systems are not contaminated.

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2. All contaminated or potentially contaminated systems are isolated from outside environment or otherwise equipped with isolation devices to control discharges.
3. Systems discharges to the environment are equipped with radiation monitoring devices to automatically isolate the discharge if high activity is detected. The high-activity setpoints for these systems are provided in the POSAR that ensure compliance to the (Darlington New Nuclear Project (DNNP) site derived release limits.

12.3.5.2 Sealing of Radioactive Components and Leak Isolation

1. Gaseous radioactive systems are sealed.
2. Equipment designs includes leak isolation, limiting leak devices, or controlling fluid leaks by piping to the sumps or using drip pans with drains piped to the floor drains.

12.3.5.3 Corrosion Limitation Products

1. Thermal cycle system corrosion is limited by using high quality, corrosion-resistant materials, and continuously controlling other properties via water chemistry.
2. Retention and elimination of corrosion products with the use of CFD are incorporated into the thermal cycle systems.
3. Features preventing flow discontinuities that lead to retention of corrosion products (crud traps) in the walls of the equipment and components are incorporated. Bends, branches, corners, dead legs, and low points are avoided in piping and piping layout. Mitigating engineering features are added where avoidance is not possible.
4. Backing rings and consumable inserts are not used in pipes containing radioactive materials.

12.3.5.4 Activity Control and Monitoring Systems

The activity control and monitoring criteria are described by system function.

12.3.5.4.1 Process Sampling System

Sample stations in the plant provide routine reactor water quality surveillance:

1. Local sampling points and sampling stations are in low radiation areas.
2. Fume hoods are used for airborne contamination control. Sample sinks, fume hoods, and glove boxes are constructed of polished stainless steel and/or plastic easing decontamination if a spill occurs.
3. Grab spouts are located above the sink reducing contaminating surrounding areas during the sampling process.
4. Radiation shielding is used for panels, local sampling boxes, cells, and racks to maintaining ALARA.
5. Sampling stations are closed system and the grab samples taken at the sampling stations have a chemical fume hood preventing operating personnel exposure. Constant air velocity is maintained through the working face of the hoods ensuring airborne contamination does not escape to the room under operating conditions.
6. A shut-off valve is placed in the sample lines immediately downstream of the sample nozzle, and sampling lines not continuously used during normal operations are drained-sloped to prevent crud pockets or areas of stagnant fluids.

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7. If a sample cannot be returned to its original system, it is directed to the nearest appropriate drain.
8. Provisions are made for flushing pipes with demineralized water before the sample bottle is removed from the panel. Panel drains are drained to the appropriate drain system.
9. Means are provided for remote handling and shielded transport of sample bottles, when required.
10. Connections to containment airspace are provided facilitating atmospheric sampling for isotopic composition and hydrogen and oxygens concentrations in the event of an accident.

12.3.5.4.2 Process Radiation and Environmental Monitoring System

1. Gross beta-gamma activity is constantly monitored in the main steam line to detect potential increases in fuel leaks to the coolant.
2. Activity monitors are installed close to the interface barriers where clean fluids that serve to cool the contaminated process fluids might receive radioactive fluid leakage.
3. Activity monitors are installed to monitor and control system discharge. These systems are isolated upon high level detection.
4. Area radiation monitors are installed in:
 - a. Areas where doses may significantly increase either periodically or inadvertently
 - b. Areas where unauthorized or inadvertent movement of radioactive material may occur
 - c. Areas where leaks of radioactive material from process streams may occur
 - d. Areas where nuclear fuel is stored or handled, such as areas close to the refueling machine and close to the fuel pool
 - e. Areas frequently accessed for operation or maintenance requirements

To prevent radiation overexposure, to detect leaks, and to prevent high-activity discharge, the atmospheres of the rooms and stacks are monitored for particulates, noble gases, and iodine. Radiation monitor locations and descriptions for the plant are provided in Subsection 12.3.14 Monitoring of Individuals and Working Areas (Instrumentation for Radiation Levels and Airborne Radioactivity).

The Containment, TB, TB, PLSA, and RWB atmospheres are monitored for particulates, noble gases, and iodine:

- Laboratories are monitored continuously for noble gas and particulate activity levels, and periodically for iodine
- Atmosphere adjacent to personnel access hatches is monitored for particulates
- Buildings housing large equipment that store reactor coolant are continuously monitored for noble gases and particulates
- Buildings where radioactive waste is stored are continuously monitored for particulates
- Refueling machine atmosphere is continuously monitored for noble gases during refueling

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- Particulates, via an iso-kinetic sampler, and noble gas activity levels are continuously monitored in the TB stack during normal and shutdown operations, and periodic surveillance

Activity monitoring systems detect and signal off-normal operating conditions for:

- High differential pressure in the sampling filters
- High and low air flow in the activity monitoring lines
- Malfunction of a blower or monitor
- Incorrect operation of valves

12.3.6 ALARA Structures Criteria

This section delineates the main ALARA criteria provided in the BWRX-300 structural design.

12.3.6.1 Building/Structure Specific Shielding Requirements

1. Containment - The major shielding structures consist of containment walls and reactor shield wall (bioshield):
 - a. The reactor shield wall consists of shielding using the steel bricks and steel plate.
 - b. The containment outer wall is a steel concrete steel cylinder or steel bricks, that totally surrounds containment. A steel brick top slab covers containment.
 - c. The shielding for the RB maintains areas requiring regular daily access to dose rates less than 6 $\mu\text{Sv/hr}$ (0.6 mrem/hr).
 - d. All necessary shielding in the upper containment maintains dose ALARA during transfer of irradiated spent fuel assemblies.
 - e. Shielding or appropriate controls minimize dose rates if a fuel handling mishap results in dropping a fuel assembly across the reactor flange.
2. Penetrations through the containment wall are shielded reducing radiation streaming. Localized dose rates outside these penetrations are limited to less than 50 $\mu\text{Sv/hr}$ (5 mrem/hr). The penetrations through interior shield walls of the RB are shielded using a modified high-density silicone elastomer, or equivalent absorber, reducing radiation streaming. Penetrations are also located to minimize radiation streaming consequences into surrounding areas.
3. The CUW RHX is located in a shielded cubicle.
4. Demineralizer and filter units are in separate shielded cubicles. This arrangement allows maintenance of one unit while the other operates. The dose rate in the adjoining demineralizer cubicle from the operating unit is less than 250 $\mu\text{Sv/hr}$ (25 mrem/hr). Infrequent entry into the demineralizer cubicle is through shielded hatches. The bulk of the piping and valves for the demineralizers and filters are in an adjacent shielded valve gallery.

Backflushing and resin application of the demineralizers are controlled from an area where dose rates are less than 10 $\mu\text{Sv/hr}$ (1 mrem/hr).
5. Fuel Storage - The fuel storage pool ensures the dose rate around the pool area is less than 25 $\mu\text{Sv/hr}$ (2.5 mrem/hr).
6. Fuel Handling - Integral shielding installed on the refueling machine is equivalent to one foot of water. Refueling pool depth maintains less than 25 $\mu\text{Sv/hr}$ (2.5 mrem/hr) at the

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water surface (per ANSI/ANS 57.1) during transit of a single grappled fuel bundle from/to the reactor vessel and the spent fuel racks.

7. The dose rate in the Main Control Room (MCR) is limited to a maximum threshold of 2 $\mu\text{Sv/hr}$ (0.2 mrem/hr) during normal reactor operating conditions. The outer walls of the CB are designed to attenuate radiation from within the RB and from possible airborne radiation surrounding the CB following beyond design basis event. The wall shielding limits the direct-shine exposure from all potential high radiation sources.
8. Noble gas airborne concentrations are expected to be negligible throughout the TB except near the SJAEs. These components are placed in shielded cubicles that are not normally occupied during operation.
9. Facilities supporting responses to accidents and emergencies in accordance with REGDOC-2.5.2, Section 8.13.3 and REGDOC-2.10.1, Sections 5.2 and 5.7 are provided with radiological protection and monitoring equipment necessary to ensure that personnel radiation exposure during accident recovery does not exceed 50 mSv (5 rem).

12.3.6.2 General Layout and Accesses

1. Radiological protection layout follows the concept of separation into three distinct and isolable areas: uncontrolled access areas, controlled (limited) access areas, and uncontrolled areas.
2. The controlled access areas encompass all radioactive systems, equipment, components, and materials.
3. The buildings and areas that define the controlled access area are separated and/or have sufficient shielding to maintain dose rates in uncontrolled areas to less than 6 $\mu\text{Sv/hr}$ (0.6 mrem/hr.)
4. The routes between buildings of the controlled access area and their accessible rooms are short as possible, designed for easy passage, and run through low radiation areas.
5. All accessible rooms are easily reached from service corridors.
6. The accessible areas floor-ceiling height accommodates personnel transporting tools for inspections and maintenance.
7. Areas are provided for storing removable components and mobile shielding that do not hinder personnel passage and mobility.
8. Plant areas are arranged for service operations, monitoring or inspection of highly radioactive equipment that is carried out manually or remotely.
9. Easy escape routes are provided with appropriate signage posted.
10. Access to high radiation areas is posted and blocked by physical barriers.
11. Access to enclosures accommodates entry and exit of components and tools.
12. Properly sized access hatches are provided (e.g., installation or removal of plant components).
13. Equipment, instruments, and sampling stations that require routine maintenance, calibration, operation, or inspection, are located for ease of access and minimum occupancy time.
14. Highly radioactive equipment plant areas are laid out to allow remote or mechanical operation, service, monitoring, or inspection.

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15. Transportation equipment (monorails, carts, hoists, etc.) is provided so that equipment is serviced in lower radiation areas.
16. Centralized control panels are provided permitting remote operation of all SC equipment from the lowest radiation zone practicable and are provided for SCN equipment.
17. Adequate space is provided for movable shielding for sources within the service area.
18. Means to control contamination and facilitate decontamination of contaminated areas are provided.
19. Space for pumps and valves not containing radioactive fluids are outside radioactive areas.
20. Pump configuration provides sufficient space for maintenance, including laydown space for disassembled piping, insulation, and components and for maintenance support features such as monorails and work platforms.
21. Personnel exposure is minimized during inspection and maintenance by locating equipment and instrumentation as far as possible from potential sources of high radiation.
22. The BWRX-300 plant does not contain underground pipelines that are directly buried in the ground (i.e., contained in concrete trenches/tunnels or concrete duct bank) or buried in masonry wall construction.
23. Penetrations through outer walls of the building containing radiation sources are sealed to prevent miscellaneous leaks into the environment.
24. Sloped floor drains are provided in shielded cubicles and other areas where spill potential exists limiting contamination.
25. Smooth, flat surfaces are used where radioactive spills could occur and expanded metal-type floor gratings are minimized.
26. The RW tank cubicles concrete is provided with a sealant and tank cubicle stainless steel liners to prevent any water releases to the environment.
27. The RB washdown bays are designed so that after the spent fuel cask is loaded on the transporter, potential surface contamination is monitored and decontaminated.
28. The washdown bays include the following:
 - a. Walls or curbs are located around areas of potential contaminated fluid leakage
 - b. Floor surfaces are sloped to drains, and sumps sized for maximum decontamination liquid flow rate
 - c. Concrete surfaces, including floor surfaces are protected with a non-porous coating
 - d. The decontamination fluid is processed through the liquid RW system
29. Health physics facilities and features administratively control:
 - a. Plant personnel activities limiting personnel exposure to radiation and radioactive materials ALARA, and within the guidelines of REGDOC-2.7.1
 - b. Effluent releases to maintain releases ALARA, and within the limits of CSA-N288.1, and the plant Operational Limits and Conditions (OLC)
 - c. Waste shipments meet applicable shipping and receipt material requirements at the storage or burial site

12.3.6.3 Walls, Structures and Shielding Elements

1. The shielding requirements:
 - a. Limit radiation exposure to the public, workers, and the environment that are ALARA and comply with REGDOC-2.7.1, Section 4.4.1 (Reference 12.3-1) requirements
 - b. Limit radiation exposure to workers, in the event of an accident, to levels that are ALARA and consider the limits in ICRP Publication 103 (Reference 12.3-3)
 - c. Limit the radiation exposure of critical components assuring component performance and design life are not impaired
2. Shielding reduces radiation levels from N-16 sources in occupied areas during power operation.
3. Shine between joints of different shielding elements, especially removable blocks, are avoided preventing radiation passage.
4. Penetrations through shield walls are avoided reducing the number of streaming paths. Whenever penetrations are required through shield walls, they are located minimizing the effect on surrounding areas. Penetrations are located so that the radiation source cannot pass through the penetration. When this is not possible, an added order of radiation source reduction is provided.
5. The annular area between pipe and penetration sleeves and electrical penetrations are filled with shielding material reducing the streaming. Examples of the shielding materials used in these applications include modified high-density silicone elastomer (or equivalent), with a density comparable to concrete, and boron-loaded refractory-type material for applications requiring neutron as well as gamma shielding.
6. Whenever a penetration cannot be avoided and a simpler and more effective solution does not prevent the radiation passing through the penetration, a shielded enclosure is installed around the penetration as it exits the shield wall.
7. The maximum number of elements is fed through penetrations and openings minimizing the number of openings.
8. The number of openings between highly radioactive areas and accessible areas are minimized.
9. Equipment is located minimizing shielding requirements. Shielded labyrinth entrances are selected for radioactive pump, equipment, and valve rooms instead of shield doors and hatches. Adequate space is provided in labyrinth entrances for access.
10. All systems containing radioactivity are identified and shielded based on access and exposure level requirements for surrounding areas.
11. Provisions are made for shielding major radiation sources during ISI to reduce inspection personnel exposure. For example, steel platforms are provided for ISI of the RPV welds and associated piping.
12. Concrete used for shielding purposes is designed considering guidance in Regulatory Guide 1.69 (Reference 12.3-4). Where special circumstances dictate, steel, encapsulated lead, water, a modified high-density silicone elastomer, or a boron-laced refractory material is used.

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13. Shielding thicknesses are selected reducing the aggregate dose rate from significant radiation sources in surrounding areas to values below the upper limit of the radiation zone specified in the zone maps.

12.3.6.4 Pools Containing Radioactive Liquids

1. The liner welds for all pools are equipped with a leak detection system.
2. The leak chase is grouped to different pool areas and leakage is directed to area drains. This allows both leak detection and determining where leaks originate. After construction is finished, each isolated pool is leak tested.

12.3.6.5 Doors and Shield Walls

1. Doors are installed to manage access to high radiation level areas. Additional measures such as radioactive signs or locks may be used.
2. The doors in shielding labyrinths are located at the entrance to the labyrinth.

12.3.6.6 Coatings, Surface Finishes and Painting

1. Epoxy-type wall and floor coverings are used providing smooth surfaces easing the decontamination. Smooth surfaces are applied to the walls and floors of the RB, TB, RWB, and TB Annex rooms containing equipment with liquid radioactive sources, floor drain areas, and washdown bays.
2. A non-porous surface paint and finish for easing decontamination is applied to room walls, floors, staircases, and platforms.
3. Protective coatings are qualified and capable of surviving a Design Basis Accident (DBA) without adversely affecting safety mitigating SSCs per guidance in Regulatory Guide 1.54 (Reference 12.3-5).
4. A contrasting undercoat color is included in the coating system for wear indication.

12.3.7 ALARA Criteria for Electrical Instrumentation and Control Design

The ALARA criteria are applied to the electrical and instrumentation design.

12.3.7.1 Component Arrangement

1. Motor control centers and system control panels are in low radiation level areas.
2. Motor-operated valves are controlled from low radiation level areas.
3. The cable trays and equipment layout does not hinder personnel passage.
4. The electrical motors connectors located in high radiation zones has quick disconnects simplifying repair and maintenance.
5. The RW piping gallery between the TB and the RW contains only SCN electrical cables that are separated from the RW piping by a minimum shielding of 20 cm (7.9 in).

12.3.7.2 Lighting, Telephone and Loudspeaker System

1. Lighting levels are suitable for surveillance and maintenance operations in radiation areas.
2. Multiple lighting fixtures prevent the immediate need to replace faulty bulbs in shielded cubicles.
3. Lighting fixtures are in easily accessible locations reducing exposure time during bulb replacement.

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4. Light bulbs within containment, main steam tunnel, and refueling level of the RB containing mercury are not used due to mercury's neutron activation potential.
5. Radioactive areas rooms and corridors have emergency lighting and illuminated exit signs to indicate escape routes.
6. Alarms and the personal address system use uninterruptible power supplies.
7. The lighting equipment, communications, and loud-speakers location allows for maintenance with minimal radiation exposure.

12.3.7.3 Instrumentation

1. Instrumentation for equipment operation, control, and sampling is located in low-level areas (shielded rooms for valves, corridors, or control rooms) that are accessible during normal and off-normal operating conditions. Shielded valve galleries are provided for the CUW, FPC, and RW (e.g., spent resin tank) systems.
2. Sensing lines are routed from primary system taps avoiding placing transmitters or readout devices in high radiation areas. For example, reactor water level sensing instruments are located outside containment.
3. Instrument sensors requiring inspection, repair, or maintenance accessibility are located facilitating operations.
4. Instruments located in high radiation areas allow removal to low radiation areas for maintenance.
5. Transmitters and display instruments are in radiation zones A or B to the extent possible. The radiation levels in these zones are defined in Subsection 12.6.1.
6. Sampling instrumentation system design minimizes the required amount of radioactive fluid.
7. Radiation monitors located in high radiation areas have displays located in an adjacent low radiation areas and local visible and audible alarms.

12.3.7.4 Materials

Components and cables used in radioactive areas are qualified for the calculated radiation conditions.

12.3.8 System and Major Component Design Considerations

12.3.8.1 Main Condenser

1. Main condenser hotwell provides a holdup for Main Steam Isolation Valve leakage.
2. During normal operation, the main condenser shells operate with a vacuum. Radioactive circulating water leakage to the atmosphere does not occur. Any air leakage is into the main condenser shell side.
3. Condensate is retained in the main condenser permitting radioactive decay before entering the condensate system.
4. Necessary shielding and controlled access for the main condenser is provided.

12.3.8.2 Nuclear Boiler System

1. Entry into the RB steam tunnel is through a controlled and shielded personnel access door that has a concrete labyrinth to attenuate radiation streaming from the adjoining steam lines. During reactor operation, the steam tunnel is controlled access.

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2. Selected valves within the steam tunnel are provided with drains that are piped to drain sumps minimizing leakage into surrounding areas. Floor drains are provided within the steam tunnel minimizing contamination if a leak occurs.
3. Penetrations through the steam tunnel walls are minimized, reducing streaming paths. Penetrations through the steam tunnel walls are located exiting controlled access areas or in areas not aligned with the steam lines. A modified high-density silicone elastomer (or equivalent) is employed whenever possible for these penetrations reducing the available streaming area presented.
4. The steam tunnel shields the plant complex from N-16 gamma shine in the main steam lines. The steam tunnel walls provide shielding limiting the direct-shine exposure from the main steam lines at any point that are inhabited in areas adjacent to the steam tunnel to less than 50 $\mu\text{Sv/hr}$ (5 mrem/hr) during normal operations (controlled access area).

12.3.8.3 Fuel Pool Cooling System

1. The FPC system services the fuel pool, cask pit, and reactor cavity pool on a rotating basis. This system operates continuously removing generated heat, reducing pool evaporation, and reducing contamination.
2. The FPC system includes two independent filters and demineralizers that remove radioactive contamination from the fuel pool, cask pit, and reactor cavity pool. These units are the highest radiation level components in the system. Each unit is located in a concrete-shielded cubicle accessible through a shielded hatch.
3. Remotely backflushing the filters and demineralizers is possible. FPC system filters are backwashed into a receiving tank and routed to the liquid radioactive waste system.
4. Clean water connection flushing lines are provided for the FPC system prior to switching between pools preventing contamination between pools.
5. All processing FPC system valves (inlet, outlet, recycle, vent, and drain) on the filters and demineralizers are located outside the shielded cubicles in a separate shielded cubicle or area together with associated piping, headers, and instrumentation.
6. FPC system piping containing resin is sloped downward to the receiving system or tank.
7. All shielded FPC system components are consolidated in the same section of the RB. Personnel access to shielded system components is controlled. Shielding for the components reduces the radiation level to less than 10 mSv/hr (1000 mrem/hr) in adjacent areas where normal access is permitted.
8. FPC system operation is accomplished from the MCR, and local control panels located where radiation levels are less than 50 $\mu\text{Sv/hr}$ (5.0 mrem/hr), and controlled limited access is permitted.
9. The FPC system provides for the collecting, monitoring, and draining pool liner leaks from the fuel pool, cask pit, and reactor cavity pools to the Liquid Waste Management System.
10. The fuel pool is equipped with drainage paths for:
 - a. Preventing stagnant water buildup behind the liner plate
 - b. Preventing the uncontrolled loss of contaminated pool water
 - c. Providing liner leak detection and measurement

11. The fuel pool, cask pit, and reactor cavity pool is also equipped with stainless steel liners and leak detection drains. All leak detection drains permit free gravity drainage to a local building drain sump.

12.3.8.4 Condensate Filters and Demineralizers System

The Condensate Purification System (CPS) shields dose rates in uncontrolled areas less than 6 $\mu\text{Sv/hr}$ (0.6 mrem/hr). The CPS removes condensate system corrosion products and impurities resulting from condenser tube leakage, some radioactive material, activated corrosion products, and fission products carried over from the reactor.

12.3.9 Radwaste Building

Liquid and solid radioactive wastes generated during normal plant operations are transferred to the RWB for collection, holdup, and processing by Liquid Waste Management System (LWM) and the Solid Waste Management System (SWM).

1. LWM and SWM process subsystems include shielding and controls limiting accessible general area radiation levels to less than 10 mSv/hr (1000 mrem/hr). Transient radiation levels during filter media or waste container transfer operations may exceed these levels, but remote operation limits the average worker radiation dose to less than 150 $\mu\text{Sv/hr}$ (15 mrem/hr).
2. Design features minimizing occupational exposure include:
 - a. Remote pipe and equipment flushing
 - b. Remote viewing and handling equipment
 - c. Centralized sampling station minimizing exposure time
 - d. LWM and SWM tanks vent to RWB Heating, Ventilation, and Air Conditioning (HVAC) for processing
 - e. Radiation monitors with alarms provided both inside and outside RWB building liquid or solid RW processing rooms
3. Processing systems (pumps, valves, tanks, and skid-mounted process systems) are remotely operated from a central RWB control panel.
4. Remote operation for routine radioactive waste management system functions is performed from local RW control panel. This includes remote actuated valves for controlling process flow between process stations and permanently installed equipment.
5. The RWB process systems area accommodates modular shield walls that limits access and reduces radiation levels from waste processing equipment.
6. Dry active waste sorting, processing, and packaging operations are also performed in the RWB or TB Annex. Portable radiation detectors, portable shielding, and remote handling tools reduces radiation levels and occupational exposure.
7. Routine maintenance uses skid-mounted process systems for RW processing, reducing maintenance requirements for permanently installed systems.

12.3.9.1 Liquid Waste Management System

The LWM collects liquid wastes from floor drains, and other sources within the facility. Radioactive liquids generated during operation are segregated, collected, stored, and processed in the LWM.

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1. Radioactive contaminants are removed, and the bulk of the liquid is purified and either returned to the Condensate Storage Tank (CST) or reprocessed. The BWRX-300 is designed as a zero liquid release plant under normal conditions.
2. LWM is divided into subsystems where liquid wastes are segregated and processed according to the impurity type and chemical content. Segregating liquid waste minimizes the total waste produced through efficient processing.
3. The calculated individual dose from the annual release of radioactive liquid effluents meets the DNNP action levels.
4. The liquid effluent release controls based upon the DNNP, action levels, and discharge limits are provided in the POSAR.
5. Liquid waste removed and concentrated in filter media, ion exchange resins, and other forms is further processed into the SWM.
6. Radioactive waste tanks effluents discharge near the tank room ventilation exhaust register. Baffle plates are used in tank vents preventing liquid and entrained solids splashing out. No hard connections to the ventilation system are used to avoid liquid or moisture propagation through the ventilation system damaging the HEPA filters.
7. The RW system has significant holdup capacity in waste collection tanks and sample tanks that allows for reprocessing and minimizes effluent releases to the environment.

12.3.9.2 Solid Waste Management System

1. Radioactive solid waste is segregated for wet and dry packaging minimize total waste stream and offsite shipment and storage.
2. SWM activities are controlled remotely at a local control panel. Remote controlled operations include movement of casks and liners, filter handling, resin transport, and movement or reconfiguration of RW processing skids.

12.3.9.3 Offgas System

1. The release of radioactive material into the atmosphere is minimized by the OGS. The OGS holds gas allowing for decay of the offgas stream that contain krypton, xenon, iodine, nitrogen, and oxygen isotopes.
2. The release of gaseous radioactive effluents to the environs maintains exposure in unrestricted areas ALARA according to action levels.
3. The gaseous effluent release controls based upon the DNNP action levels, action levels, and discharge limits are provided in the POSAR.
4. OGS equipment is selected, arranged, and shielded maintaining occupational exposure ALARA.
5. OGS contains a recombiner, condenser, and activated carbon beds.
6. OGS provides sufficient holdup until the required fraction of the radionuclides has decayed, and the daughter products are retained by the charcoal.

12.3.10 Reactor Building Spent Fuel and Components

The RB structural design provides flow resistance in contaminated areas that hold up radioactive releases from containment.

Spent Fuel Storage Design Bases: The fuel storage pool has capacity for five refueling cycles and a complete core offload as well as some highly radioactive waste (e.g., filters) assuring safety under normal and postulated accident conditions. These safety measures include:

- Periodic inspection and testing of components important to safety
- Radiation Protection shielding
- Containment, confinement, and filtering systems
- Residual heat removal capability that is reliable and testable
- Ensuring fuel storage coolant inventory under accident conditions

The design allows for:

- Controlling airborne release
- Preventing coolant inventory drainage below the acceptable depth through the use of drains, gates, and weirs
- Coolant flow to spent fuel racks
- Fuel pool liner leakage detection and confinement

12.3.10.1 Reactor Water Cleanup System

1. CUW system that is in contact with the reactor coolant system is constructed of corrosion-resistant materials. For system piping, smooth bends are used instead of welded fittings. At certain locations within the system, stainless steel pipes with electro-polished finish are used to reduce corrosion product buildup.
2. CUW components include an RHX, a pressure reduction station, and associated valves located inside the TB.
3. The shielded cubicle CUW valves and piping reduces radiation levels, but entry is not allowed during normal operation. Most CUW valves and piping are in a shielded valve gallery adjacent to the cubicle and are remotely operated.
4. Chemical cleaning and decontamination connections use condensate to flush piping and equipment for decontamination prior to RHX maintenance pressure reduction station, and associated valves.
5. HVAC system limits the spread of contaminants from these shielded CUW cubicles by maintaining negative pressure.
6. Personnel access to the CUW cubicles for component maintenance is controlled to minimize personnel exposure.

12.3.10.2 Control Rod Drive System

The FMCRD design allows removal of the motor unit, position indicator probe, separation indicator probe and lower component for maintenance during plant outages without disturbing the upper assembly of the drive.

12.3.10.3 Containment

1. Containment and other safety features limit accident radiological effects and release are within acceptable limits.

2. Containment air that is inerted with nitrogen during normal operation is purged (specifically purging N-16) before access is allowed during shutdown conditions. Containment access is not permitted during normal operation above 25% power.
3. Prior to removing the reactor vessel head, airborne radioactive material sources are removed by either the containment purge exhaust system or the main condenser.

12.3.10.4 Containment Leak Detection and Isolation System

For process lines that penetrate the containment, leakage detection and isolation is provided. The leakage detection instrumentation initiates alarms at established limits and isolates the affected system(s).

12.3.10.5 Plant Cooling Water System

Plant cooling water is supplied to for the following fire protection measures:

1. Fire suppression devices are provided for activated carbon beds and radioactive filter that ignite due to high inlet temperatures.
2. Fire protection barriers and fire detection protects decontamination areas and RW storage areas.
3. Fire protection hoses are in low radiation areas.

12.3.10.6 Secondary Control Room

1. SCR shielding design protects personnel from radiological conditions during normal operation. Shielding permits access and occupancy of the SCR following an event where the MCR becomes uninhabitable.
2. Habitability contingencies are provided for continued occupancy under accident, station blackout conditions, and severe accidents.

12.3.11 Control Building

1. The MCR shielding design is based upon protecting personnel from radiation resulting from the most limiting design basis accident. Shielding permits access and occupancy of the MCR to ensure plant personnel exposure following an accident does not exceed dose limits.
2. The MCR is shielded against normal sources of radiation. Habitability contingencies are provided for continued occupancy under design basis accident, station blackout conditions and Design Extension Conditions without core melt. Severe accidents require operators relocate to the secondary control room.
3. Safe occupancy of the MCR during abnormal conditions (AOOs, DBAs) is provided in the design.

12.3.12 Decommissioning ALARA Considerations

The following plant design features ensure the plant is operated and maintained ALARA. These features aid in facilitating ALARA exposures during decommissioning:

1. Shielding design protects workers during maintenance or repairs and decommissioning.
2. The RB, TB, RWB, and TB Annex design consists of entry doors from the outside and numerous cubicles with equipment hatches inside the buildings for large equipment removal to facilitate decommissioning.

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3. The RW process systems located in the RWB are skid-mounted and facilitate decommissioning and ease of access, by allowing truck access, and skid loading/unloading.

12.3.13 Ventilation

12.3.13.1 General HVAC System ALARA Design Controls

1. The following design objectives apply to building ventilation systems:
 - a. Systems ensure releases to the environment are ALARA.
 - b. Radionuclide air concentrations in accessible areas to personnel for normal plant surveillance and maintenance is below 10% of the derived air concentrations during normal power operation.
 - c. HVAC room concentration setpoints are determined ensuring airborne releases are ALARA and comply with the action levels in the POSAR.
2. The plant stack and the major streams feeding the stack are monitored by the PRM so that action is taken avoiding releases in excess of regulatory limits.
3. During normal operation purging, the containment exhausts air through the HVAC HEPA filter unit prior to stack discharge.
4. During an isolation event, if offsite or backup power is not available, outside air is supplied to MCR by Emergency Filter Unit (EFU) operation.
5. The EFUs are in closed areas preventing the spread of radioactive contamination and providing adequate space for maintenance activities. Before charcoal removal, radioactivity is allowed to decay to minimal levels.
6. The control room EFU ventilation systems filter unit operates during accident conditions providing pressurization and outside air.
7. The shielding wall thickness between the HVAC filter cubicles is sized so that the dose contribution in any cubicle does not exceed 250 $\mu\text{Sv/hr}$ (25 mrem/hr) under normal operation, ensuring filter maintenance exposure ALARA.

12.3.13.2 Heating, Ventilation and Cooling Systems

1. HVAC system limits airborne contamination by providing airflow patterns from areas of low contamination to higher contaminated areas. This is accomplished by creating negative pressure areas in contaminated cubicles maintaining air flow into each cubicle from the corridor area. The higher contaminated areas exhausts flow through the HEPA filters prior to the plant stack.
2. HVAC service in contaminated areas circulates air through the contaminated areas of the buildings in a once-through fashion. Flow is directed from corridors (point of highest pressure) to the equipment alcove rooms, to the area rooms, to air ducts, and finally to the exhaust ductwork for processing through HEPA filters.
3. The ventilation system in the fuel pool and equipment areas sweeps air from the fuel pool surface, removing the major portion of potential airborne contamination at the source.
4. The passage of ventilation ducts through walls and structures ensures the shielding efficiency is not reduced. HVAC duct penetrations used for exhaust are routed above personnel head height to minimize streaming effects to worker exposure. HVAC duct penetrations are in line of sight between radiation sources and occupied areas.

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5. Major HVAC equipment (e.g., fans, coolers) is located in dedicated low radiation areas to minimize maintenance worker exposures.
6. Airflow from areas of lower contamination potential to areas of greater contamination potential are maintained using a negative pressure in the more contaminated area.
7. Buildings containing radioactive materials are equipped with radiological monitoring connections (sample ports) in the exhaust ductwork targeting radiological system leakages.
8. In areas where HVAC air emergency filtration is required, joints, corners, sharp bends, obstructions, and dampers where radioactive contamination accumulates are minimized.
9. Fume hoods or similar devices are installed in laboratories, sampling rooms, and component decontamination rooms.
10. HEPA and charcoal filters are installed in potentially contaminated exhaust pathways.
11. HVAC filters that are highly contaminated are shielded from one another and from personnel access areas to minimize exposure.
12. Isolation dampers are installed at fans, whenever flow needs segregation for repair and maintenance.
13. Remote actuation of redundant filter trains is provided for high-activity areas within the building.
14. HEPA filters are equipped with differential pressure detection and remote indication.
15. The MCR and SCR envelopes maintain overpressure.
16. HEPA filters are located close to the radioactive source.
17. Respiratory protection is provided in cases where high airborne contamination levels contaminants are not removed efficiently by the HVS.

12.3.14 Monitoring of Individuals and Working Areas (Instrumentation for Radiation Levels and Airborne Radioactivity)

The following Process Radiation and Environmental Monitoring System (PREMS) radiation monitoring subsystems are provided for monitoring area radiation and airborne radioactivity within the plant:

- Containment Monitoring Subsystem (CMon)
- Process Sampling Subsystem (PS)
- Process Radiation Monitoring Subsystem (PRM)
- Area Radiation Monitoring Subsystem (ARM)

Airborne radioactivity in effluent releases and ventilation air exhausts is continuously sampled and monitored by the PREMS for noble gases, air particulates and halogens. Airborne contamination is sampled and monitored at each stack, in the offgas releases, and in the ventilation exhaust from the RB, RWB and TB. Samples are periodically collected and analyzed for radioactivity. In addition to this instrumentation, portable air samplers are used to check for airborne radioactivity in work areas prior to entry where radiation levels have the potential to exceed the allowable limits.

12.3.14.1 Containment Monitoring Subsystem

The CMon provides instrumentation for monitoring the following conditions inside containment:

- Hydrogen and oxygen concentration
- Pressure
- Water level
- Temperature
- Relative humidity
- Area radiation
- Gross gamma radiation for fission products

Signals are provided to the appropriate Distributed Control and Information System according to safety category. The Distributed Control and Information System processes each signal for display in the MCR and SCR, actuation of audible and visual alarms, and initiation of response functions such as reactor scram, containment valve isolation, feedwater isolation and PAM.

Monitoring is provided during normal reactor operation, AOOs, and post-accident conditions.

12.3.14.2 Process Sampling Subsystem

PS collects liquid and gaseous samples for analysis. This analytical information is used to monitor plant and equipment performance. This process subsequently informs plant operations if changes to operating parameters are necessary in order to meet acceptance criteria. This subsystem functions during all plant operational modes. Continuously sampled flows are routed from select locations in the process streams to the sampling stations. The sample flows enter the sample stations where pressure, temperature, and flow adjustments are made as necessary. Grab sampling facilities and special monitoring are provided for select areas (e.g., liquid radioactive waste). Continuous samples are diverted to continuous monitoring equipment that transmits data to the plant computer. Alarms indicate off-normal conditions.

12.3.14.3 Process Radiation Monitoring Subsystem

Airborne radioactivity in effluent and ventilation air exhausts are continuously sampled and monitored for noble gases, air particulates, and halogens by the PRM. PRM samples and monitors airborne contamination at the stack, in OGS streams, and in the ventilation exhaust from the RB, RWB, and TB. Detection of high radiation levels are alarmed in the MCR. Sampling in the exhaust lines is continuously analyzed. PRM facilitates periodic sampling for analysis. In addition to this instrumentation, portable air samplers are used to check for airborne radioactivity in work areas prior to entry where radiation levels could exceed allowable limits. Detectors in the PRM subsystem provide measurement indication of airborne radiological conditions from occupied areas.

12.3.14.4 Area Radiation Monitoring Subsystem

The ARM continuously measures, indicates and records gamma radiation levels at strategic locations throughout the plant except within containment. Alarms are activated in the MCR and SCR, as well as in local areas to warn personnel to take appropriate protective measures.

ARM detectors are located upstream of the CB and SCR AHUs. These detectors indicate and alarm in the MCR and SCR and are used by plant personnel in conjunction with portable monitors to evaluate airborne radiological conditions in occupied areas.

12.3.14.4.1 Area Radiation Monitoring Subsystems Description

Every ARM channel consists of a gamma-sensitive detector and a digital area radiation processor. All channels are provided with local visual and audible alarms and local readouts. Additional readouts and alarms are provided by local auxiliary units. The output signals from the detectors are digitized and multiplexed for transmission to digital radiation monitors for measurement and display. The radiation signals are transmitted to the process computer for recording. Each radiation monitoring channel has adjustable trip alarm circuits, one for high radiation and the other for downscale indication (loss of sensor input). Also, each area radiation monitor has a built-in self-check for gross failures and activates an alarm on a power failure or an inoperative monitor. Auxiliary units with local audible alarms are provided in selected local areas that immediately warn occupants. Each area radiation monitor is powered from an uninterruptible power source that is continuously available during loss of offsite power.

12.3.14.4.2 Area Radiation Monitoring Subsystems Detector Locations and Sensitivity

The appropriate locations and monitoring ranges of each area radiation monitor is finalized in detailed design phase.

12.3.14.4.3 Area Radiation Monitoring Subsystems Design Parameters and Requirements

Two high-range radiation channels are provided on the refueling floor and fuel pool area to monitor radiation from a design extension condition Fuel Handling Accident. The conservative Out of Core Criticality analysis provided in Subsection 15.5.9.3 demonstrates that a criticality alarm system is not necessary at the BWRX-300 DNNP. An Out of Core Criticality accident in the refueling area is practically eliminated due to physical impossibility and supporting operating procedures. Practical elimination of this event is consistent with BWRX-300 Safety Strategy, described in Chapter 3, Subsection 3.1.8.

Defense Line 1 design provisions and supporting operating procedures provide design and administrative controls that effectively manage factors that influence system reactivity and the criticality. Design factors that influence reactivity are enrichment, mass, moderation, geometry, reflection, interaction and spacing. Plant procedures prohibit the handling and storage of more fuel assemblies than have been determined to be safely subcritical at any one time and under the most adverse moderation conditions feasible by un-borated water. The amount of water moderator available for internal flooding is limited by the storage location structure design making any credible possibility of an Out of Core Criticality physically impossible. As a result, the need for a criticality alarm system is practically eliminated consistent with BWRX-300 Safety Strategy as described in Chapter 3, Subsection 3.1.8.

Both process radiation monitors and area radiation monitors are in the fuel storage and associated handling areas.

Process radiation monitor alarms and isolation are provided for fuel pool area ventilation. Area radiation monitors with visual and audible alarms is also provided.

Final alarm setpoints established in the field are based on sensor location, background radiation levels, expected radiation levels, and low occupational radiation exposures. The radiation alarm setpoint for each channel is set slightly above the background radiation level for the monitor location.

12.3.15 References

- 12.3-1 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."
- 12.3-2 Government of Canada SOR/2000-203, "Radiation Protection Regulations."
- 12.3-3 ICRP Publication 103, "The 2007 Recommendations of the International Commission on Radiological Protection," International Commission on Radiological Protection.
- 12.3-4 USNRC Regulatory Guide 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants."
- 12.3-5 USNRC Regulatory Guide 1.54, "Service Level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants."

12.4 Shielding

The permanent shielding analysis is completed during detailed design phase. The evaluation identifies those areas experiencing elevated dose rates during power and shutdown conditions and are not mitigated by structure or component design or location. The anticipated dose rates and frequency of access are included in analysis. Using these results, system interaction and structural/component loading analyses, permanent shielding is installed in target areas/components. Permanent shielding is preferred to temporary shielding.

12.4.1 General Design Guides

The primary objective of radiation shielding is protecting operating personnel, and the general public from radiation emanating from the reactor, the power conversion systems, the RW process systems, and the auxiliary systems, while maintaining appropriate access for operation and maintenance in accordance with REGDOC-2.7.1, Sections 4.1 and 5.3 (Reference 12.4-1). Radiation shielding keeps radiation doses to equipment below levels where disabling radiation damage occurs.

The shielding requirements in the plant perform the following functions:

1. Limit, plant personnel, contractors and visitors, exposure ALARA and within guidance of ICRP Publication 103 (Reference 12.4-2) and CNSC SOR/2000-203 (Reference 12.4-3) limits
2. Limit personnel radiation exposure in the unlikely event of an accident to levels that are ALARA, conforming to the limits specified in REGDOC-2.5.2, Section 4.2.1 (Reference 12.4-4 and REGDOC-2.4.1, Section 4.3.2, (Reference 12.4-5) and ensuring the plant is maintained in a safe condition during and after an accident
3. Limit critical components radiation exposure within specified radiation tolerances assuring component performance and design life are not impaired

12.4.2 Design Description

The following guidance is used in the BWRX-300 shielding design:

1. All systems containing radioactivity are identified and shielded based on access and exposure level requirements of surrounding areas. The radiation zone maps described in 12.6 indicate design radiation levels where equipment shielding contributes to achieving the dose rate in the area.
2. The source terms used in the shielding calculations are analyzed conservatively. Transient conditions, shutdown, and normal operating conditions are used ensuring a conservative source is used in the analysis. Shielding is based on design basis fission product concentrations in the coolant in addition to activation products. For components where N-16 is the major radiation source, a concentration based upon BWR fleet operating plant data is used.
3. Processing equipment is located as far as possible from personnel dwelling and passage areas minimizing shielding. Shielded labyrinths are used to eliminate radiation streaming through access ways from sources located in cubicles.
4. Penetrations through shield walls are shielded reducing radiation streaming through the penetrations. The approaches used to locate and shield penetrations are discussed in Subsection 12.3.6.
5. Wherever possible, radioactive piping is run in a manner minimizing plant personnel radiation exposure that includes:

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- a. Minimizing radioactive pipe routing in corridors
 - b. Avoiding routing high-activity pipes through low radiation zones
 - c. Using shielded pipe trenches and pipe chases where routing of high-activity pipes in low-level areas cannot be avoided
 - d. Separating radioactive and non-radioactive pipes for maintenance
6. Maintaining acceptable levels at valve stations by using motor-operated or diaphragm valves. Minimizing worker radiation exposure by providing draining and flushing provisions in the valve design. Operator shielding is provided for manual valve applications using shield walls or valve stem extensions.
 7. Shielding is provided to permit access and occupancy of the MCR ensuring plant personnel exposure following an accident does not exceed the values set forth in REGDOC-2.5.2, Section 4.2.1 and REGDOC-2.4.1, Section 4.3.2 and supports compliance with REGDOC-2.5.2, Section 8.10.1.
 8. Provisions are made for shielding major radiation sources during ISI to reduce personnel exposure. Steel platforms are provided for ISI of the RPV nozzle welds and associated piping.
 9. The primary material used for shielding is concrete. Where special circumstances dictate, steel, lead, water, a modified high-density silicone elastomer, or a boron-laced refractory material is used.

12.4.3 Shielding Method Design

Radiation sources in various pieces of plant equipment are cited in Section 12.2. Shutdown conditions, such as fuel transfer operation, as well as accident conditions, such as a Loss-of-Coolant Accident (LOCA) or a Fuel Handling Accident, are considered in designing shielding for the plant.

The mathematical models used to represent a radiation source and associated equipment and shielding are established to ensure conservative calculation results. Depending on the versatility of the applicable computer program, various degrees of complexity for the actual physical situation are incorporated.

12.4.4 References

- 12.4-1 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."
- 12.4-2 ICRP Publication 103, "The 2007 Recommendations of the International Commission on Radiological Protection," International Commission on Radiological Protection.
- 12.4-3 Government of Canada SOR/2000-203, "Radiation Protection Regulations."
- 12.4-4 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 12.4-5 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."

12.5 Dose Constraints and Dose Assessment

This section describes the Radiation Protection philosophy, objectives, and requirements used throughout the plant design. Radiation Protection ensures:

1. Compliance with regulatory dose limits
2. Radiation exposure during normal plant operation is ALARA with social and economic factors taken into account
3. Cumulative plant worker doses average less than one person-Sv/year over the operating life of a single unit plant

12.5.1 Interfaces for Protecting People and the Environment

12.5.1.1 Radiation Protection Objectives

During normal operation and AOOs, the Radiation Protection Program objectives are:

1. Radiation exposures are kept within regulatory limits
2. Radiation exposures are kept ALARA with social and economic factors taken into account

Adherence to the first objective is demonstrated by comparing the calculated dose consequences with the dose limits specified in SOR/2000-203 (Reference 12.5-1). The second objective of keeping exposures ALARA, is a cost/benefit design decision for reducing dose exposures. This is accomplished by following the:

SOR/2000-203 (Reference 12.5-1) and REGDOC-2.7.1 (Reference 12.5-2).

12.5.1.2 Normal Operation Dose Limits and Action Levels

SOR/2000-203 (Reference 12.5-1) states in part: "every licensee shall implement a Radiation Protection Program and shall, as part of that program, (a) keep the amount of exposure to radon progeny and the effective dose and the equivalent dose received by and committed to persons as low as is reasonably achievable, social, and economic factors being taken into account."

Nuclear Energy Worker

The CNSC effective dose limits for a nuclear energy worker is an average of 20 mSv effective dose per year over a five-year period (100 mSv over five consecutive years), with no single year exceeding 50 mSv effective dose.

Members of the Public

Per CNSC Radiation Protection Regulations, the effective dose limit for a member of the public is 1 mSv per year from all sources of radiation other than natural background and medical exposures.

Action Levels

Action levels for the Radiation Protection Program are developed considering the guidance from REGDOC-2.7.1 (Reference 12.5-2). Typical action levels include external or internal doses above the planned ambient external or airborne hazards that are greater than anticipated in the design, and contamination found at levels significantly greater than expected with the plant or released to the public domain. These action levels are developed and available as part of the licence to operate.

12.5.1.3 Limiting Exposures

The BWRX-300 design minimizes radiation exposure and is consistent with CNSC REGDOC-2.7.1 (Reference 12.5-2).

12.5.1.4 Waste Management

Management of radioactive solid, liquid, and gaseous waste produced annually and during the operating life of the station is minimized according to ALARA principles and practices. Radioactive waste streams are controlled to limit worker and public doses and comply with regulatory limits. See Chapter 11 for a description of RW management systems.

The BWRX-300 design ensures public exposure from all radionuclides in effluent streams is less than 1 mSv/year (100 mrem/year). The following design and operational practices ensure the plant operates within the action levels:

1. Controlled and monitored liquid and gaseous waste releases
2. Liquid and gaseous waste activity is measured and recorded prior to release
3. The waste management systems have a target dose of 50 μ Sv per year averaged over the plant lifetime at the exclusion area boundary (see REGDOC-2.7.1) (Reference 12.5-2)

12.5.1.5 Radiation Monitoring

Radiation monitoring includes:

1. Detecting failed fuel in the core
2. Measuring external radiation in plant locations where plant workers and equipment are present
3. Monitoring, tracking, and recording personnel dosimetry
4. Monitoring, alarming, and isolating airborne contaminants
5. Providing offsite monitoring for radioactive material in the environment
6. Providing a radiation laboratory for analyzing environmental and biological samples, and workers monitoring
7. Providing a calibration facility for all plant radiation monitors

In the event of a release, the monitoring system provides:

1. Data to characterize radiological conditions around the plant and assess the need for offsite action
2. Data on radiological plant conditions facilitating safe evacuation, if necessary.

12.5.1.6 Radiation Exposure Control Program

The exposure risk for each radioactive plant system or component is estimated as part of a Radiation Exposure Control Program. The Radiation Exposure Control Program estimates the exposures involved in the operation, maintenance, inspection, repair, replacement, or decommissioning of potentially contaminated equipment.

Exposure considerations for plant workers include:

1. Breakdown of system work activities
2. Number of operators required for each activity
3. Time taken for each activity
4. External radiation field and internal airborne hazards

BWR operating experience and BWR operating data is used to inform decisions about estimated exposures.

12.5.2 References

- 12.5-1 Government of Canada SOR/2000-203, "Radiation Protection Regulations."
- 12.5-2 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."

12.6 Radiation Zones and Access Requirements

Radiation zones are established in all areas as a function of access requirements and area radiation sources. These zones are shown on Figures 12.6-1 through 12.6-17.

All areas of the plant have at least two assigned radiation zone classifications. The first classification is based on normal operation radiation fields and the second classification is based on the normal shutdown radiation fields. Additional classifications are assigned for selected off-normal events, as appropriate.

Updated radiation zones, or radiation zone maps are provided in the POSAR based on the general arrangement drawings, that show the normal operation and normal shutdown radiation zone classifications.

All radiation areas or enclosures have labels to show their radiation zone class. Access to radiation zones is controlled with locked doors in the case of high radiation areas (radiation levels greater than 1 mSv/h (100 mrem/hr)) with clearly marked entrances and chain barriers and marked entrances.

Technology infrastructure systems supporting permanent and temporary functions include:

- Physical access controls for radiologically controlled and significant areas
- Radiation work permit change stations
- Supplemental ARM with portable devices
- Equipment radiation monitoring
- Integrated security access and radiologically controlled area access
- Leak detection systems

12.6.1 Radiation Zones

The following classifications for radiation zones are used:

12.6.1.1 Class A Radiation Zones

Class A radiation zones are non-radiological controlled zones and have free access and unlimited stay time. The areas outside the controlled area, including electrical equipment rooms, MCR, SCR and administrative areas, are Class A radiation zones. The radiation dose rate in these areas are allowed a maximum of 6 μ Sv/h (0.6 mrem/h).

12.6.1.2 Class B Radiation Zones

Class B radiation zones have restricted access and unlimited occupation time. This zone corresponds to the corridors, staircases, control panel areas, and areas around rooms with radioactive material. These areas do not have radioactive equipment or components. The shielded walls guarantee worker safety. The radiation dose rate in these areas reach 10 μ Sv/h (1 mrem/h), and there is no radioactive or environmental contamination. Access to these areas is through designated entry control points, and rooms with a higher radiation zone class are accessed from these control points.

12.6.1.3 Class C Radiation Zones

Class C radiation zones have restricted access and occupation time limited to 20 h/wk. These zones are areas of frequent access, such as sampling stations and valve operating rooms. The dose rate in contact with the existing equipment is not greater than 50 μ Sv/h (5 mrem/h), and there is a possibility of radioactive and/or environmental radiation.

12.6.1.4 Class D Radiation Zones

Class D radiation zones have restricted access and occupation time is limited to 4 h/wk. These areas are infrequently accessed and are rooms for valves or equipment where the dose rate on contact is no greater than 250 $\mu\text{Sv/h}$ (25 mrem/h). There is a possibility of radioactive and/or environmental radiation.

12.6.1.5 Class E Radiation Zones

Class E radiation zones have restricted access and occupation time is limited to 1 h/wk. These are occasionally accessed for equipment actuation (pumps, valves, heat exchangers and tanks). The contact dose rate with equipment is no greater than 1000 $\mu\text{Sv/h}$ (100 mrem/h), but the radiation level may be reduced without significantly altering the plant operating mode. There is a possibility of radioactive and/or environmental radiation.

12.6.1.6 Class F to J Radiation Zones

Class F to J radiation zones are high radiation zones that are areas accessible to individuals where radiation levels external to the body could result in an individual receiving a dose equivalent in excess of 1 mSv (100 mrem) in one hour at 30 centimeters from the radiation source, or 30 centimeters from any surface that the radiation penetrates.

Access to Class F and higher zones is not permitted without specific authorization. This corresponds to equipment rooms that do not require attention for actuation, inspection, and/or maintenance, such as tanks, filters, and pipes with radiation levels greater than 1 mSv/h (100 mrem/h). Their radiation level cannot be reduced without significantly altering plant operation.

Typical non-accessible Class F to J, High and Very High radiation areas include the reactor vessel area, containment during power operation, the areas of the RB with numerous or large unshielded containment penetrations, the areas of spent fuel transfer whenever it is present, the fuel pool, the steam pipe tunnel during power operation, the FPC systems and the resin wash water collection tanks.

The relationship between radiation zone designations and accessibility requirements is presented in Table 12.6-1. The dose rate applicable for a particular zone is based on OPEX and represents design dose rates in a particular zone. They are not interpreted as the expected dose rates that would apply in all portions of that zone, for all types of work within that zone, or all instances of zone entry. BWR plants have been operating for several decades, and OPEX with similar design basis shows that only a small fraction of the permissible dose is received from radiation sources controlled by equipment layout, or the structural shielding provided. The BWRX-300 radiation zone approach accomplishes the ALARA objectives.

Access to areas in the plant is controlled and regulated by the given area zone. Areas with dose rates where an individual could receive a dose in excess of 1 mSv (100 mrem) in a period of one hour are locked and posted with "High Radiation Area" signs. Areas where an individual could receive a dose in excess of 5 Sv (500 rem) within a period of one hour at 1 meter from a radiation source, or 1 meter from any surface that the radiation penetrates are posted with "Very High Radiation Area" signs.

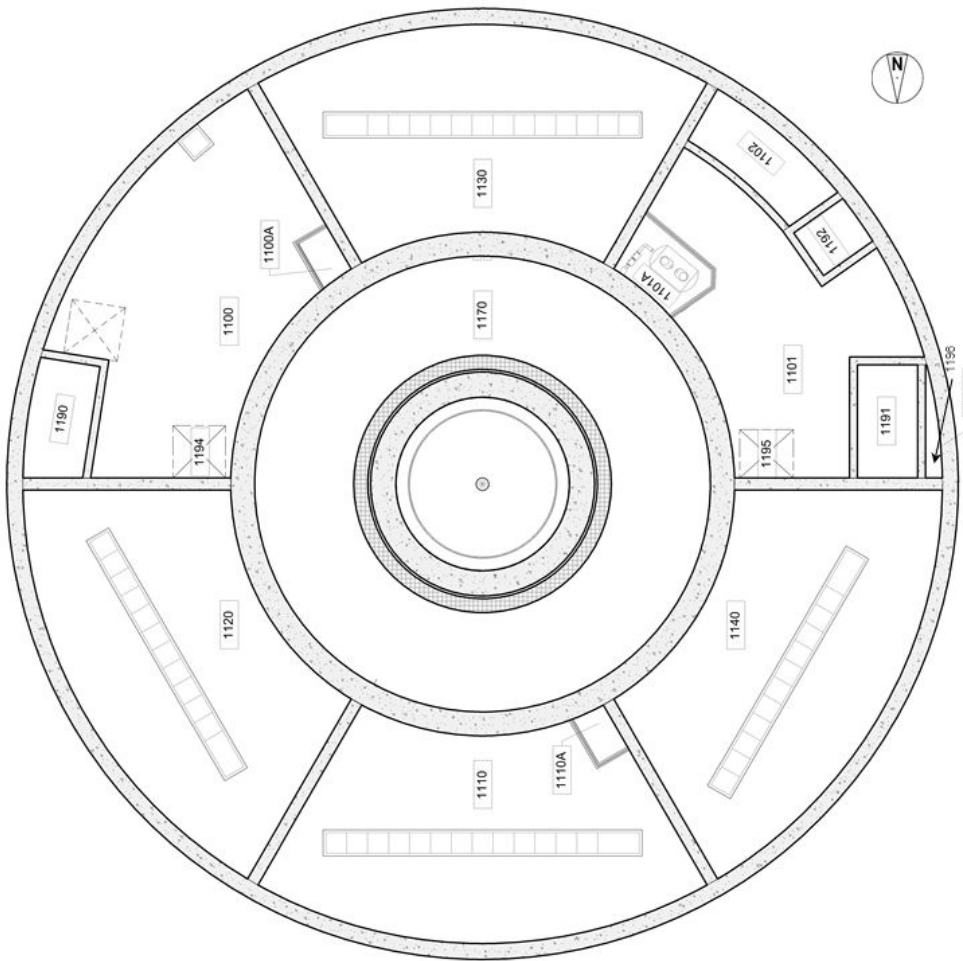
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Table 12.6-1: BWRX-300 Radiation Zone Classifications

Zone	Dose Rate	Occupancy limit	Description
A	$\leq 6 \mu\text{Sv/h}$ (0.6 mrem/h)	40 hr/wk	Uncontrolled and unlimited access
B	$\leq 10 \mu\text{Sv/h}$ (1 mrem/h)	40 hr/wk	Controlled and unlimited access
C	$\leq 50 \mu\text{Sv/h}$ (5 mrem/h)	20 hr/wk	Controlled and limited access
D	$\leq 250 \mu\text{Sv/h}$ (25 mrem/h)	4 hr/wk	Controlled and limited access
E	$\leq 1 \text{ mSv/h}$ (100 mrem/h)	1 hr/wk	Controlled and limited access
F ¹	$\leq 10 \text{ mSv/h}$ (0.1 rem/h)	5 hr/year	Limited and controlled access with special authorization permit required
G ¹	$\leq 100 \text{ mSv/h}$ (10 rem/h)	30 min/year	(Same as Zone F above)
H ¹	$\leq 1 \text{ Sv/h}$ (100 rem/h)	No access	(Same as Zone F above)
I ¹	$\leq 5 \text{ Sv/h}$ (500 rem/h)	No access	(Same as Zone F above)
J ²	$> 5 \text{ Sv/h}$ (500 rem/h)	No access	Limited and controlled access with special authorization permit required

(1) High radiation sources present. Limit and controlled access with special authorization permits required.

(2) Very High radiation sources present. Very limited and controlled access with special authorization permits required.



Radiation Zones Level -34.0			
ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1100	ENTRY -34.0	B	A
1101	SERVICE -34.0	B	A
1102	ERO STORAGE -34.0	A	A
1110	FIMCD GROUP 1 CONTROLS	D	C
1120	FIMCD GROUP 2 CONTROLS	C	B
1130	FIMCD GROUP 3 CONTROLS	C	B
1140	FIMCD GROUP 4 CONTROLS	C	B
1100A	SUMP 1	F	E
1101A	SUMP 2	D	C
1110A	SUMP 3	D	C
1170	PRIMARY CONTAINMENT	J	G
1190	STAIRWELLWELL A	B	A
1191	STAIRWELLWELL B	B	A
1192	ELEVATOR	B	A
1194	COMMODITY CHASE A*	Unoccupied/Controlled Access	Unoccupied/Controlled Access
1195	COMMODITY CHASE B*	Unoccupied/Controlled Access	Unoccupied/Controlled Access
1196	UTILITY CHASE*	Unoccupied/Controlled Access	Unoccupied/Controlled Access

* Access to the chases requires an explicit authorization permit

A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS ¹
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS ¹
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS ¹
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS ¹
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

Figure 12.6-1: Reactor Building Level -34.0 Meters Radiation Zones

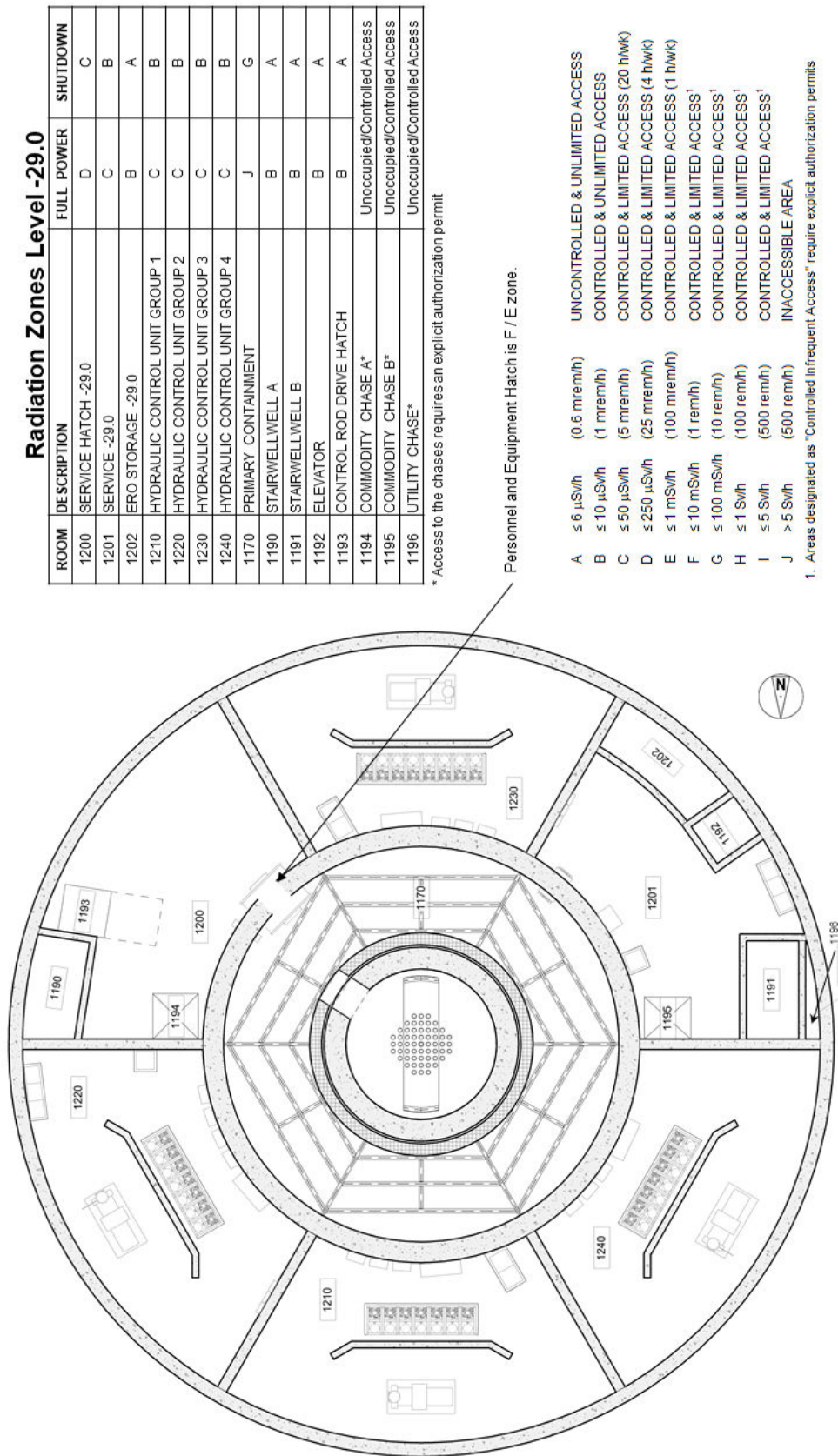
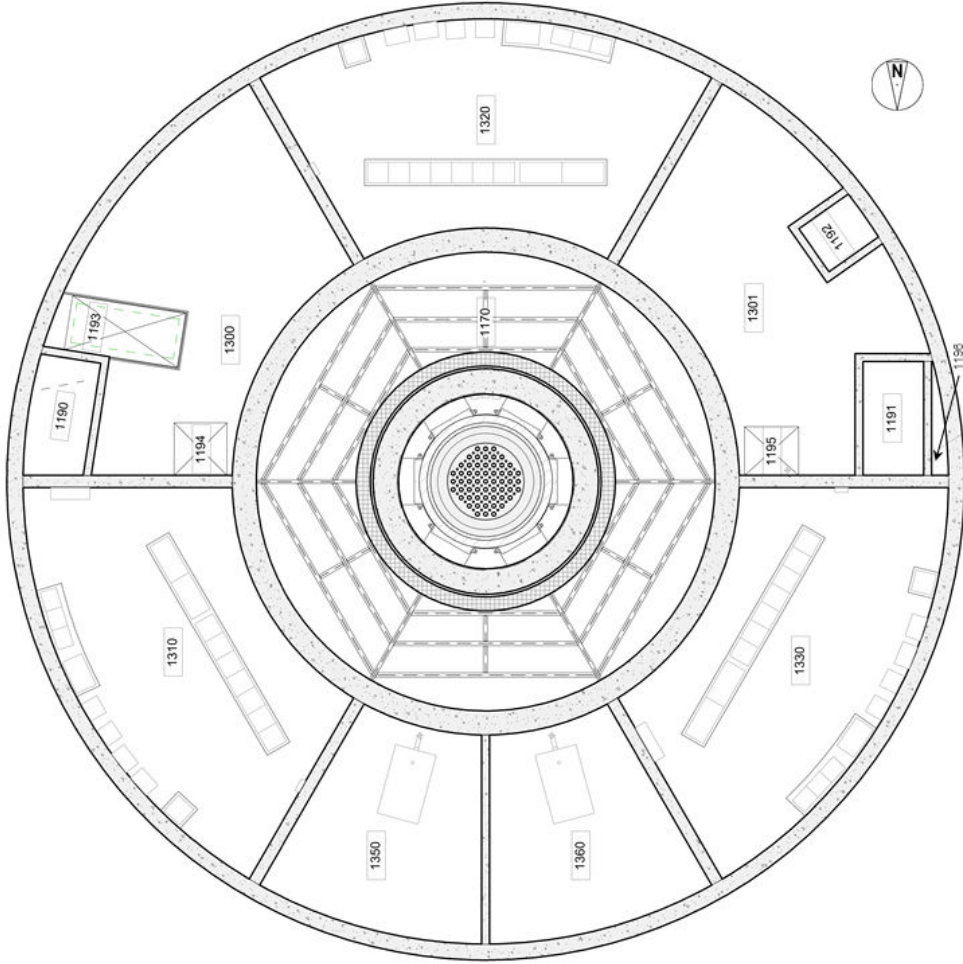


Figure 12.6-2: Reactor Building Level -29.0 Meters Radiation Zones



Radiation Zones Level -21.0

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1300	ENTRY -21.0	B	A
1301	SERVICE -21.0	B	A
1310	DIV 1 DCIS & ELECTRICAL	B	A
1320	DIV 2 DCIS & ELECTRICAL	B	A
1330	DIV 3 DCIS & ELECTRICAL	B	A
1350	SHUTDOWN COOLING PUMP A	D	C
1360	SHUTDOWN COOLING PUMP B	D	C
1170	PRIMARY CONTAINMENT	J	G
1190	STAIRWELL A	B	A
1191	STAIRWELL B	B	A
1192	ELEVATOR	B	A
1193	CONTROL ROD DRIVE HATCH	B	A
1194	COMMODITY CHASE A*	Unoccupied/Controlled Access	
1195	COMMODITY CHASE B*	Unoccupied/Controlled Access	
1196	UTILITY CHASE*	Unoccupied/Controlled Access	

* Access to the chases requires an explicit authorization permit

- A

$\leq 6 \mu\text{Sv/h}$

(0.6 mrem/h)

UNCONTROLLED & UNLIMITED ACCESS
- B

$\leq 10 \mu\text{Sv/h}$

(1 mrem/h)

CONTROLLED & UNLIMITED ACCESS
- C

$\leq 50 \mu\text{Sv/h}$

(5 mrem/h)

CONTROLLED & LIMITED ACCESS (20 h/wk)
- D

$\leq 250 \mu\text{Sv/h}$

(25 mrem/h)

CONTROLLED & LIMITED ACCESS (4 h/wk)
- E

$\leq 1 \text{ mSv/h}$

(100 mrem/h)

CONTROLLED & LIMITED ACCESS (1 h/wk)
- F

$\leq 10 \text{ mSv/h}$

(1 rem/h)

CONTROLLED & LIMITED ACCESS¹
- G

$\leq 100 \text{ mSv/h}$

(10 rem/h)

CONTROLLED & LIMITED ACCESS¹
- H

$\leq 1 \text{ Sv/h}$

(100 rem/h)

CONTROLLED & LIMITED ACCESS¹
- I

$\leq 5 \text{ Sv/h}$

(500 rem/h)

CONTROLLED & LIMITED ACCESS¹
- J

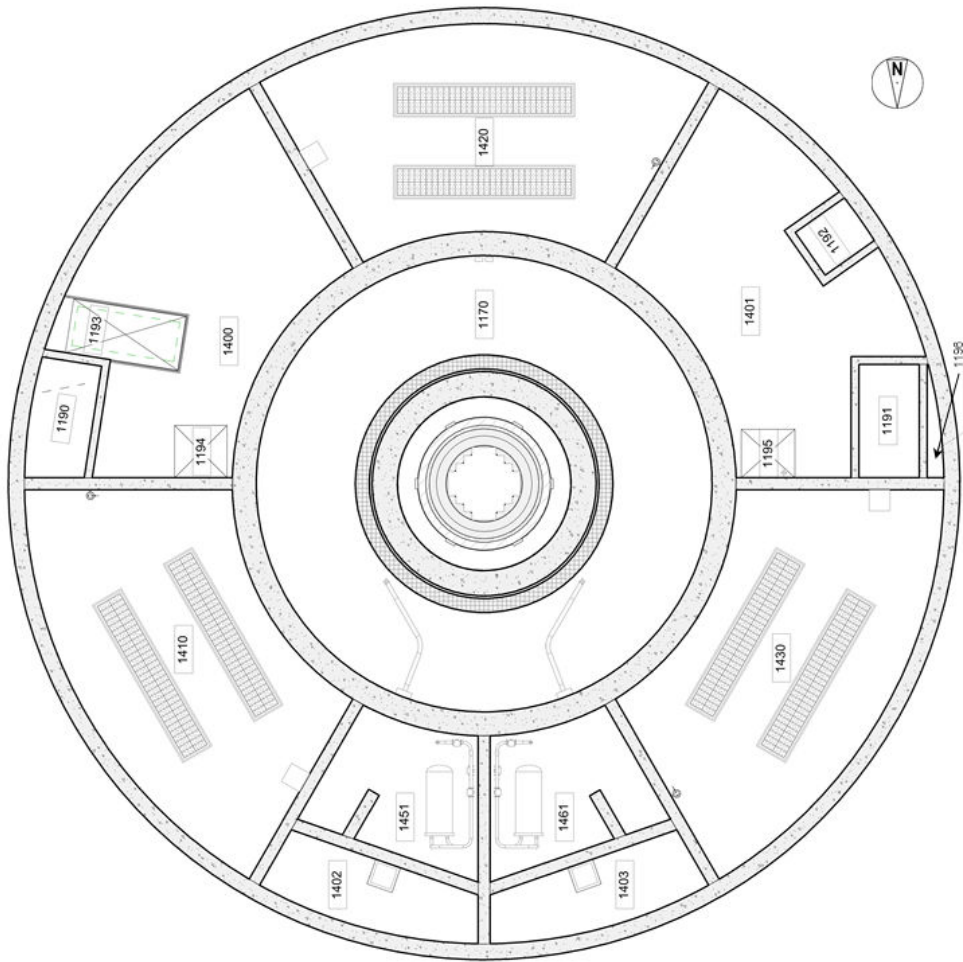
$> 5 \text{ Sv/h}$

(500 rem/h)

INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

Figure 12.6-3: Reactor Building Level -21.0 Meters Radiation Zones



ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1400	ENTRY -14.5	B	A
1401	SERVICE -14.5	B	A
1402	SHUTDOWN COOLING ENTRY A	B	A
1403	SHUTDOWN COOLING ENTRY B	B	A
1410	DIVISION 1 BATTERY ROOM	B	A
1420	DIVISION 2 BATTERY ROOM	B	A
1430	DIVISION 3 BATTERY ROOM	B	A
1451	SHUTDOWN COOLING A	D	C
1461	SHUTDOWN COOLING B	D	C
1170	PRIMARY CONTAINMENT	J	G
1190	STAIRWELL A	B	A
1191	STAIRWELL B	B	A
1192	ELEVATOR A	B	A
1193	CONTROL ROD DRIVE HATCH	B	A
1194	COMMODITY CHASE A*	Unoccupied/Controlled Access	
1195	COMMODITY CHASE B*	Unoccupied/Controlled Access	
1196	UTILITY CHASE*	Unoccupied/Controlled Access	

* Access to the chases requires an explicit authorization permit

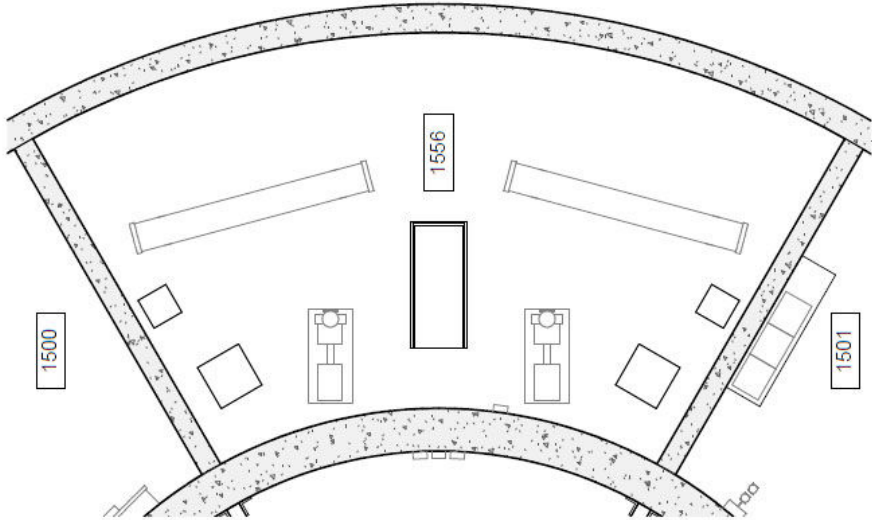
- A $\leq 6 \mu\text{Sv/h}$ (0.6 mrem/h) UNCONTROLLED & UNLIMITED ACCESS
- B $\leq 10 \mu\text{Sv/h}$ (1 mrem/h) CONTROLLED & UNLIMITED ACCESS
- C $\leq 50 \mu\text{Sv/h}$ (5 mrem/h) CONTROLLED & LIMITED ACCESS (20 h/wk)
- D $\leq 250 \mu\text{Sv/h}$ (25 mrem/h) CONTROLLED & LIMITED ACCESS (4 h/wk)
- E $\leq 1 \text{ mSv/h}$ (100 mrem/h) CONTROLLED & LIMITED ACCESS (1 h/wk)
- F $\leq 10 \text{ mSv/h}$ (1 rem/h) CONTROLLED & LIMITED ACCESS¹
- G $\leq 100 \text{ mSv/h}$ (10 rem/h) CONTROLLED & LIMITED ACCESS¹
- H $\leq 1 \text{ Sv/h}$ (100 rem/h) CONTROLLED & LIMITED ACCESS¹
- I $\leq 5 \text{ Sv/h}$ (500 rem/h) CONTROLLED & LIMITED ACCESS¹
- J $> 5 \text{ Sv/h}$ (500 rem/h) INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

Figure 12.6-4: Reactor Building Level -14.5 Meters Radiation Zones

Radiation Zones Level -4.8

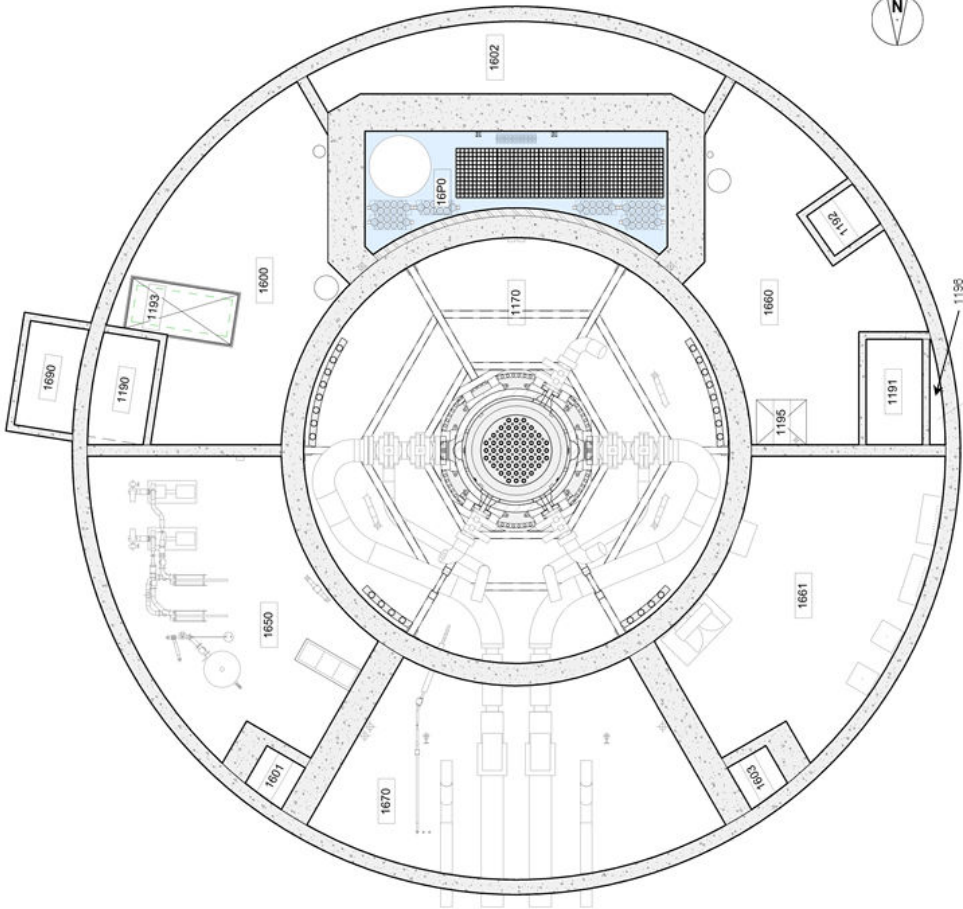
ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1556	FUEL POOL EQUIP MEZZ	E	E



A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS ¹
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS ¹
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS ¹
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS ¹
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

Figure 12.6-6: Fuel Pool Mezzanine - -4.8 Meters Radiation Zone



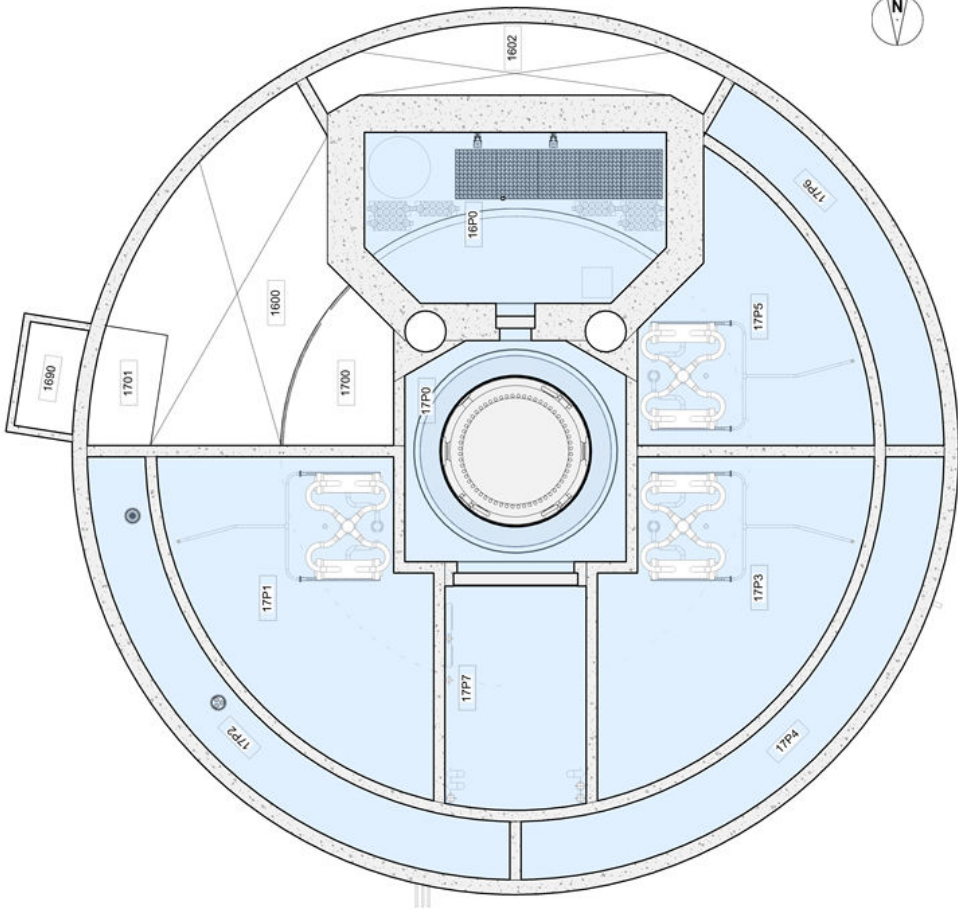
Radiation Zones Level 0			
ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1600	TRUCK BAY	C	B
1601	LABYRINTH A	F	C
1602	HALLWAY	B	B
1603	LABYRINTH B	F	C
1650	IOC COOLING	D	C
1660	SERVICES 0 A	C	B
1661	SERVICES 0 B	D	C
1670	MSFW PIPING	H	C
1690	FUEL POOL	I	I
1690	STAIRWELL C	B	A
1170	PRIMARY CONTAINMENT	J	G
1190	STAIRWELL A	B	A
1191	STAIRWELL B	B	A
1192	ELEVATOR A	B	A
1193	CONTROL ROD DRIVE HATCH	B	A
1195	COMMODITY CHASE B*	Unoccupied/No Radiation Zone	
1196	UTILITY CHASE*	Unoccupied/No Radiation Zone	

* Access to the chases requires an explicit authorization permit

A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS ¹
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS ¹
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS ¹
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS ¹
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

Figure 12.6-7: Reactor Building Level 0.0 Meters Radiation Zones



Radiation Zones Level 4.9			
ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1600	TRUCK BAY	C	B
1602	HALLWAY	B	B
1700	TRUCK BAY MEZZANINE 1	B	A
1701	TRUCK BAY MEZZANINE 2	B	B
16P0	FUEL POOL**	I	I
17P0	REACTOR CAVITY POOL**	J	F
17P1	ISOLATION CONDENSER A	C	C
17P2	ISOLATION CONDENSER POOL A	B	B
17P3	ISOLATION CONDENSER B	C	C
17P4	ISOLATION CONDENSER POOL B	B	B
17P5	ISOLATION CONDENSER C	C	C
17P6	ISOLATION CONDENSER POOL C	B	B
17P7	EQUIPMENT POOL**	C	D
1690	STAIRWELL C	B	A

** Zone J during spent fuel transfers

- A $\leq 6 \mu\text{Sv/h}$ (0.6 mrem/h) UNCONTROLLED & UNLIMITED ACCESS
- B $\leq 10 \mu\text{Sv/h}$ (1 mrem/h) CONTROLLED & UNLIMITED ACCESS
- C $\leq 50 \mu\text{Sv/h}$ (5 mrem/h) CONTROLLED & LIMITED ACCESS (20 h/wk)
- D $\leq 250 \mu\text{Sv/h}$ (25 mrem/h) CONTROLLED & LIMITED ACCESS (4 h/wk)
- E $\leq 1 \text{ mSv/h}$ (100 mrem/h) CONTROLLED & LIMITED ACCESS (1 h/wk)
- F $\leq 10 \text{ mSv/h}$ (1 rem/h) CONTROLLED & LIMITED ACCESS¹
- G $\leq 100 \text{ mSv/h}$ (10 rem/h) CONTROLLED & LIMITED ACCESS¹
- H $\leq 1 \text{ Sv/h}$ (100 rem/h) CONTROLLED & LIMITED ACCESS¹
- I $\leq 5 \text{ Sv/h}$ (500 rem/h) CONTROLLED & LIMITED ACCESS¹
- J $> 5 \text{ Sv/h}$ (500 rem/h) INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

Figure 12.6-8: Reactor Building Level 4.9 Meters Radiation Zones

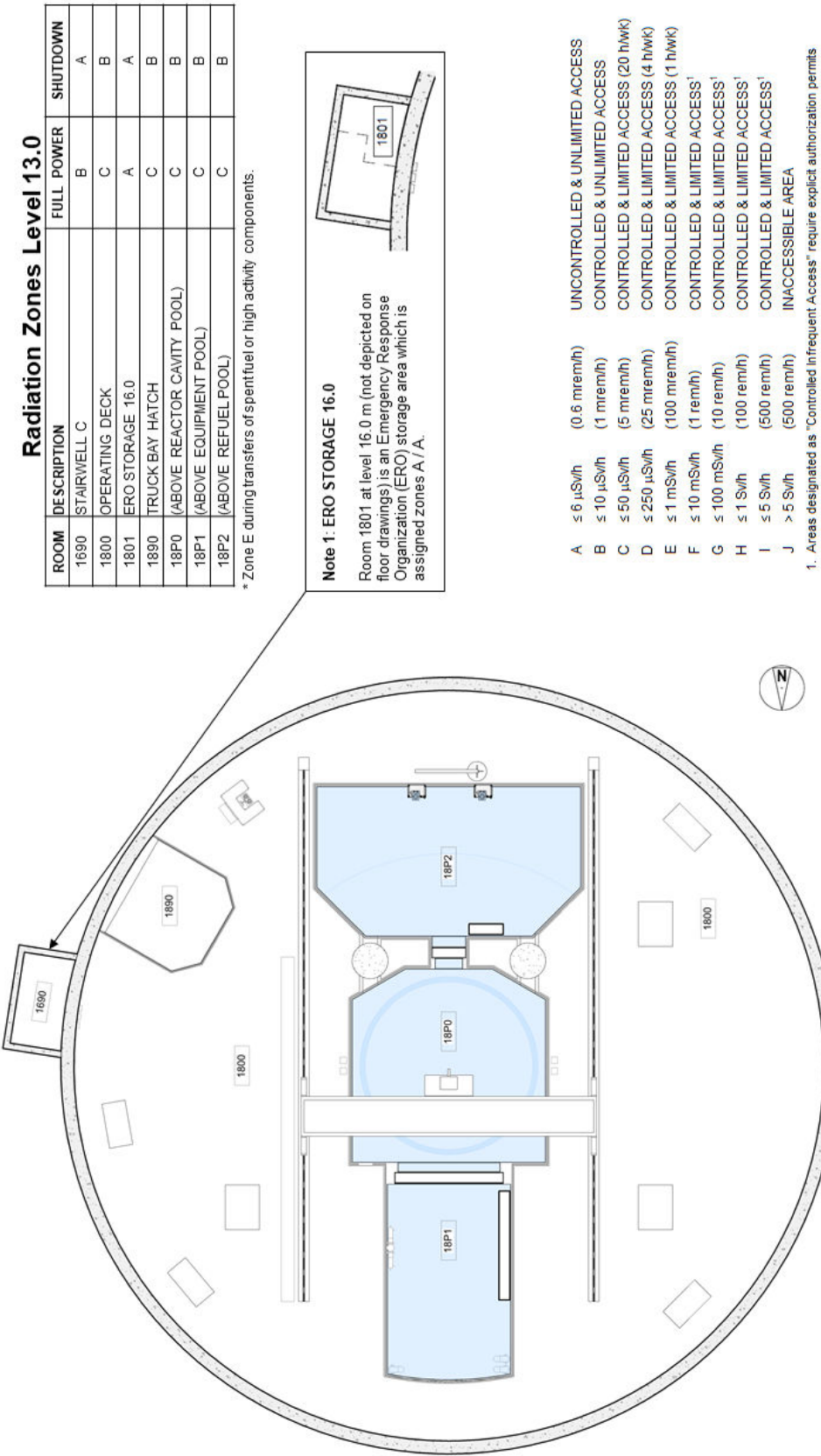


Figure 12.6-9: Reactor Building Level 13.0 Meters Radiation Zones

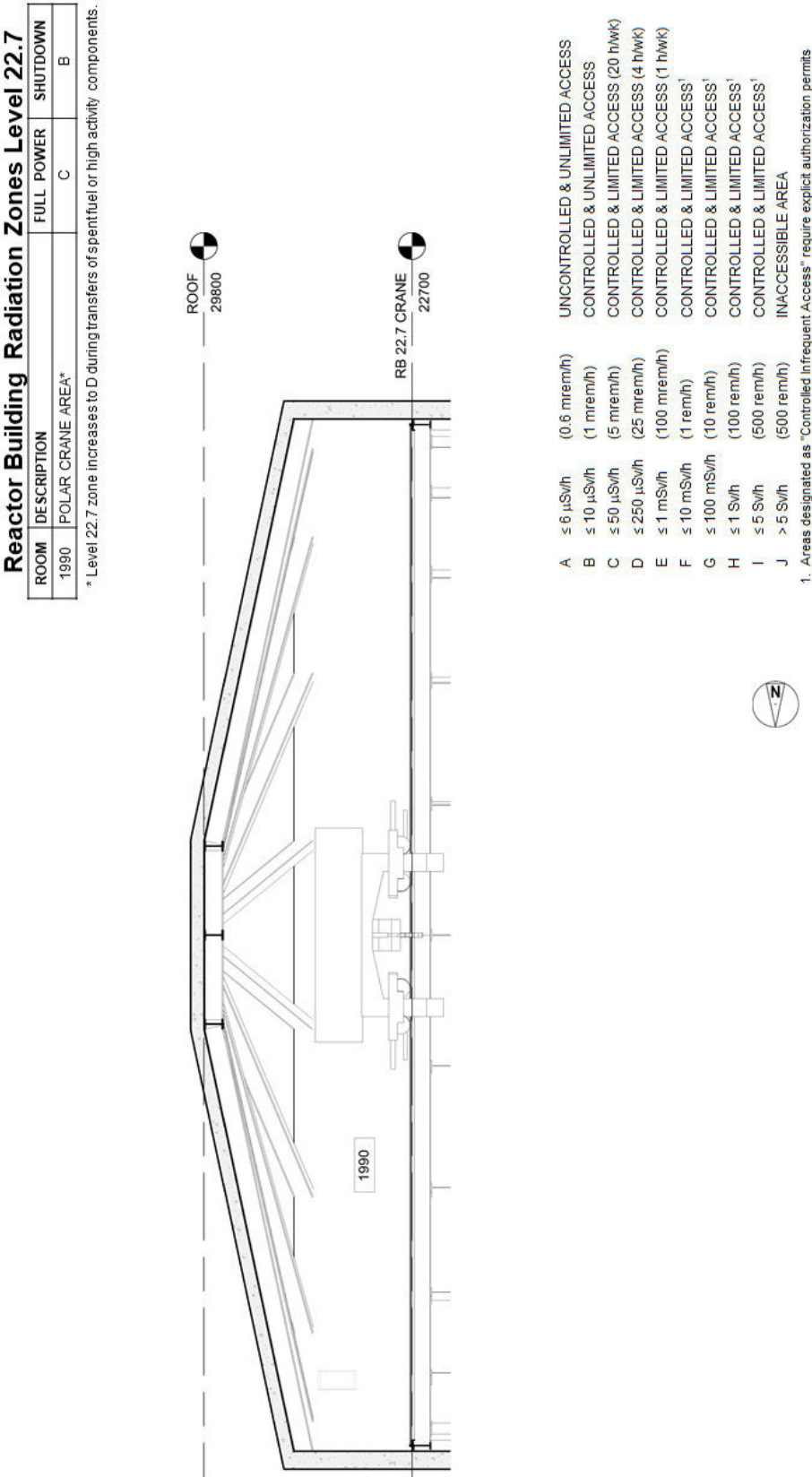


Figure 12.6-10: Reactor Building Level 20.8 Meters Radiation Zones

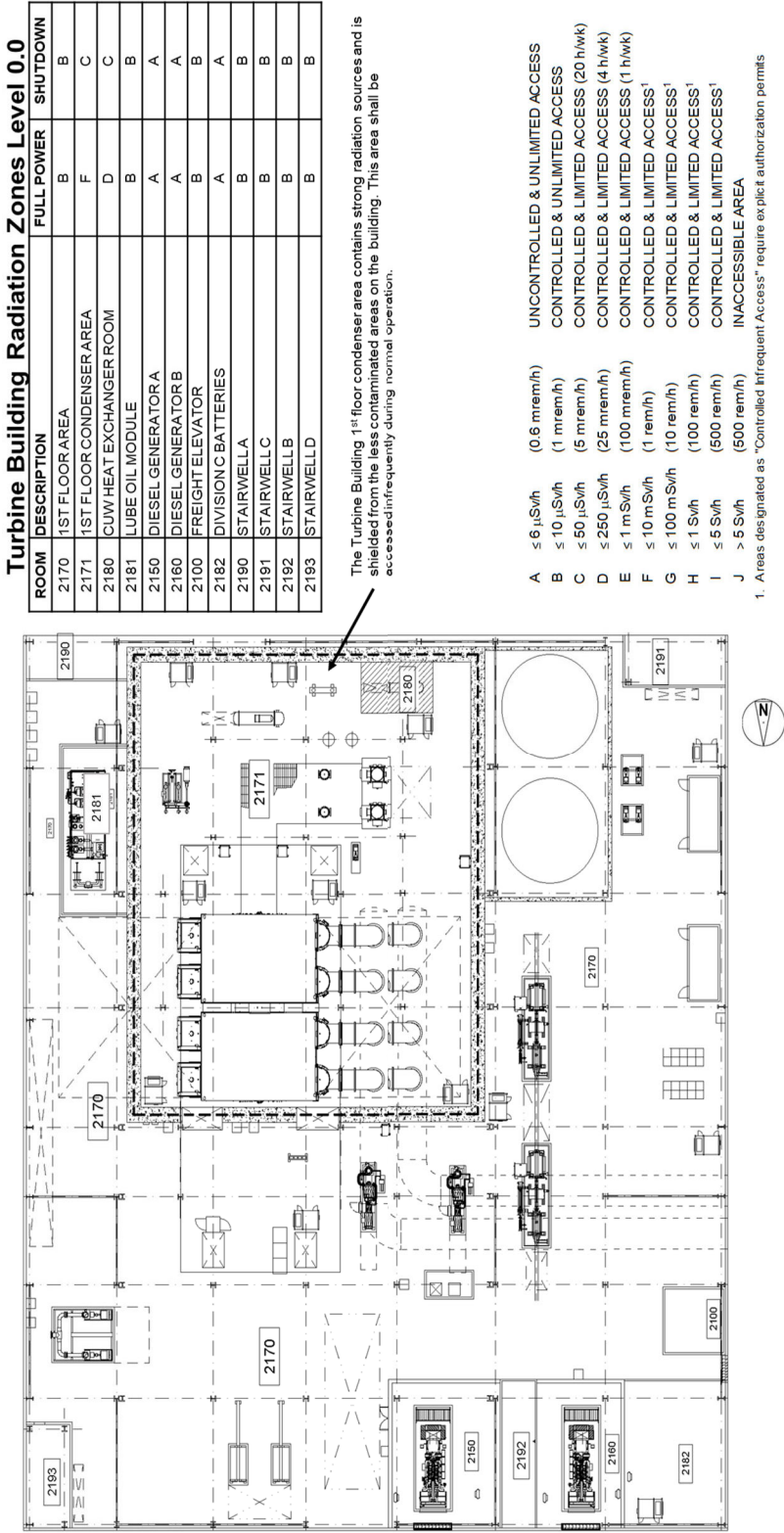


Figure 12.6-11: Turbine Building Level 0.0 Meters Radiation Zone

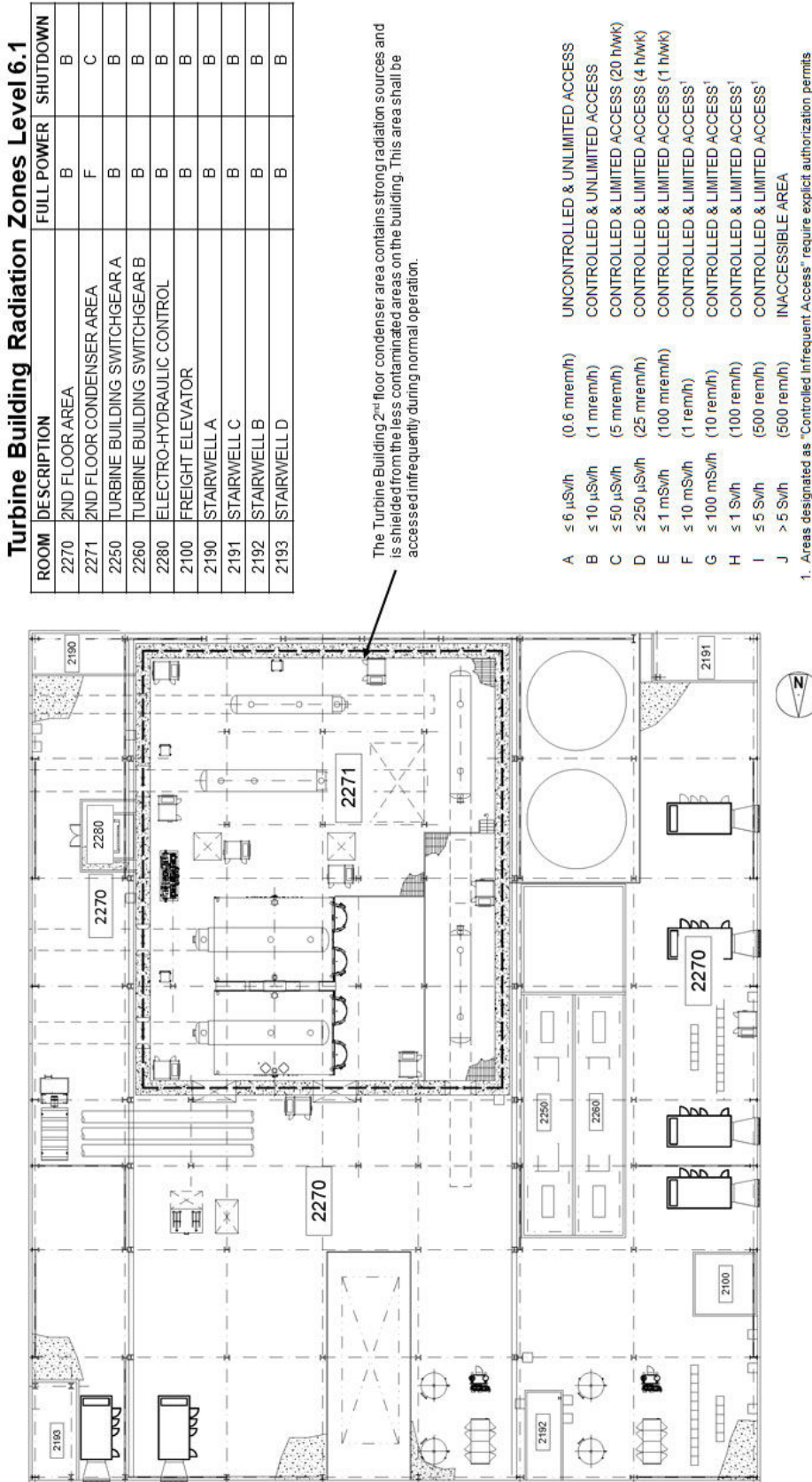


Figure 12.6-12: Turbine Building Level 6.1 Meters Radiation Zone

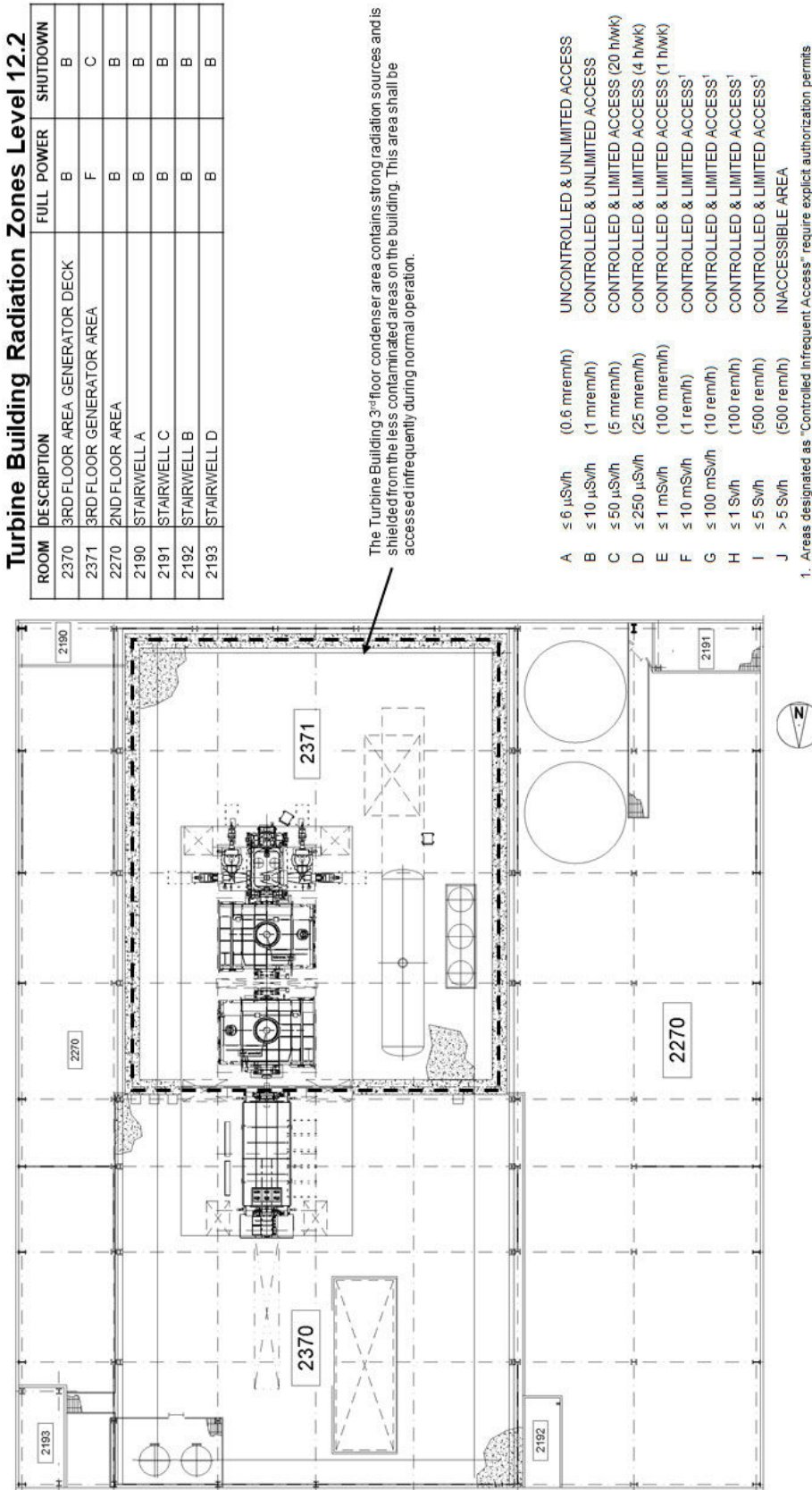
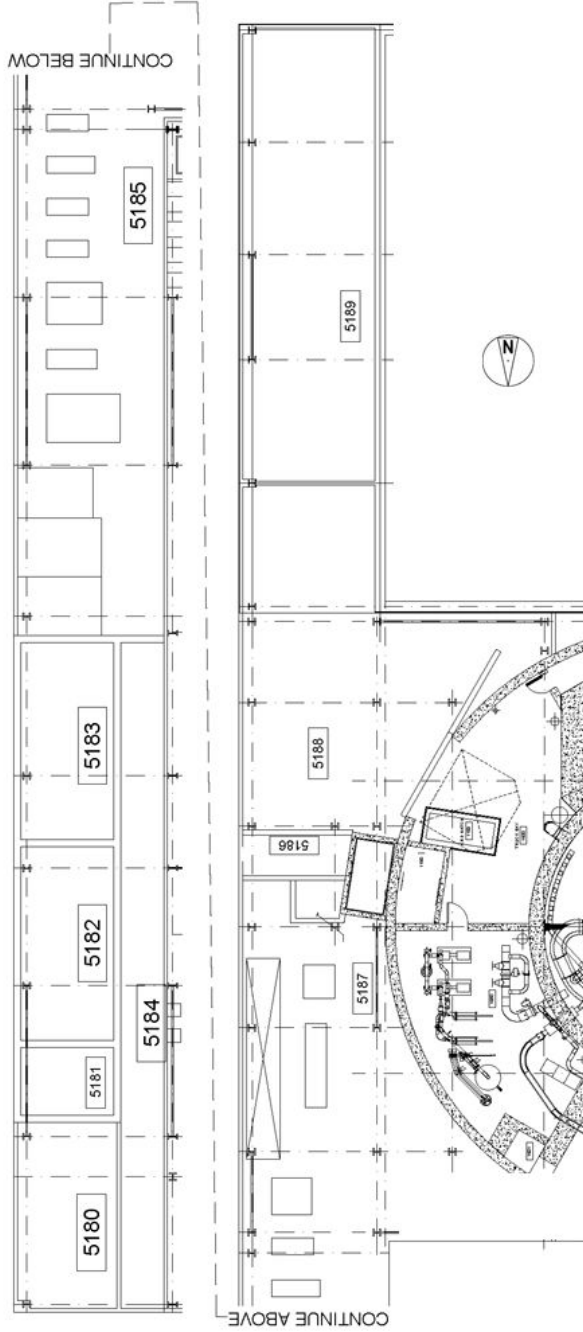


Figure 12.6-13: Turbine Building Level 12.2 Meters Radiation Zone

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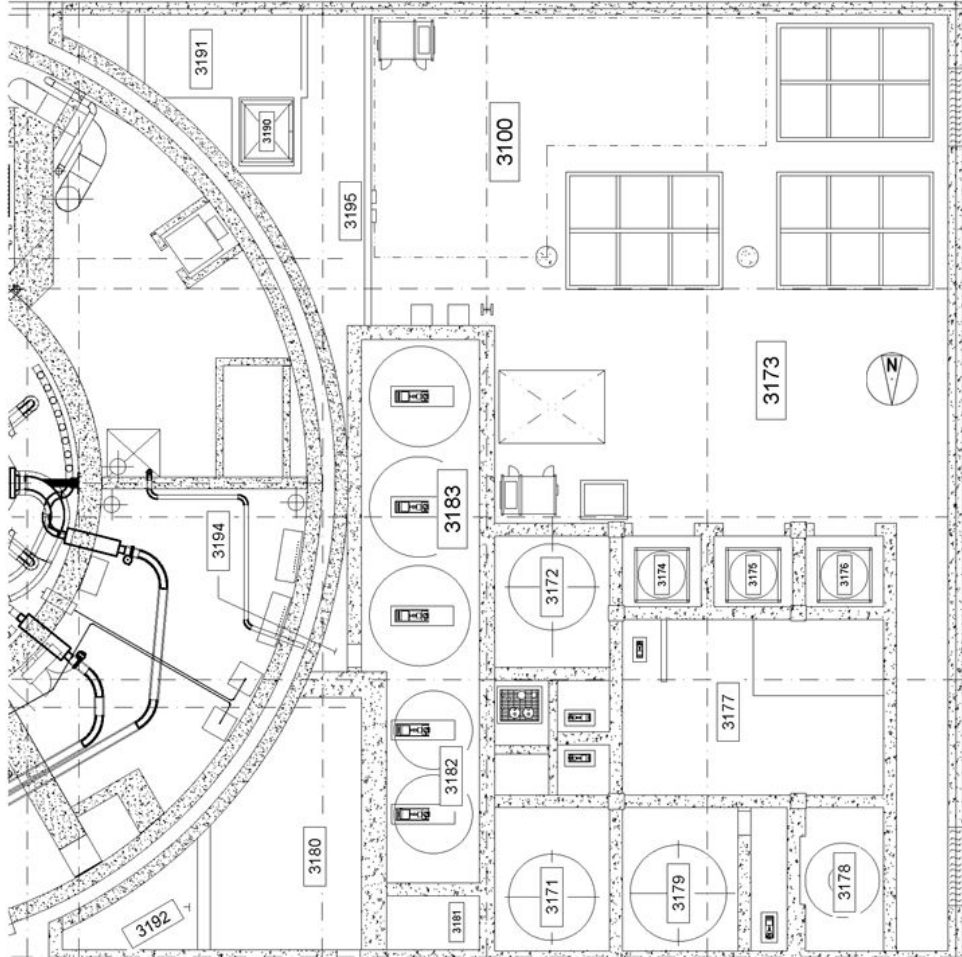
Plant Services Building Radiation Zones Level 0.0

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
5180	OFFICE SPACE	A	A
5181	I&C CALIBRATION AREA	B	B
5182	CONTAMINATED PART/TOOL STORAGE	C	C
5183	DECONTAMINATION AREA	D	D
5184	PLANT SERVICES BUILDING HALLWAY	B	B
5185	HOT MACHINE SHOP	D	D
5186	REACTOR BUILDING EGRESS PASSAGE	B	B
5187	AUX RCA ACCESS CONTROL ROOM	B	B
5188	TRUCK SPACE (CASK REMOVAL)	C	C
5189	STORAGE AREA (NEW FUEL, FMCRD, ETC.)	C	C

A	$\leq 6 \mu\text{Sv/h}$	(0.6 mrem/h)	UNCONTROLLED & UNLIMITED ACCESS
B	$\leq 10 \mu\text{Sv/h}$	(1 mrem/h)	CONTROLLED & UNLIMITED ACCESS
C	$\leq 50 \mu\text{Sv/h}$	(5 mrem/h)	CONTROLLED & LIMITED ACCESS (20 h/wk)
D	$\leq 250 \mu\text{Sv/h}$	(25 mrem/h)	CONTROLLED & LIMITED ACCESS (4 h/wk)
E	$\leq 1 \text{ mSv/h}$	(100 mrem/h)	CONTROLLED & LIMITED ACCESS (1 h/wk)
F	$\leq 10 \text{ mSv/h}$	(1 rem/h)	CONTROLLED & LIMITED ACCESS ¹
G	$\leq 100 \text{ mSv/h}$	(10 rem/h)	CONTROLLED & LIMITED ACCESS ¹
H	$\leq 1 \text{ Sv/h}$	(100 rem/h)	CONTROLLED & LIMITED ACCESS ¹
I	$\leq 5 \text{ Sv/h}$	(500 rem/h)	CONTROLLED & LIMITED ACCESS ¹
J	$> 5 \text{ Sv/h}$	(500 rem/h)	INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

Figure 12.6-14: Plant Services Building 0.0 Meters Radiation Zone



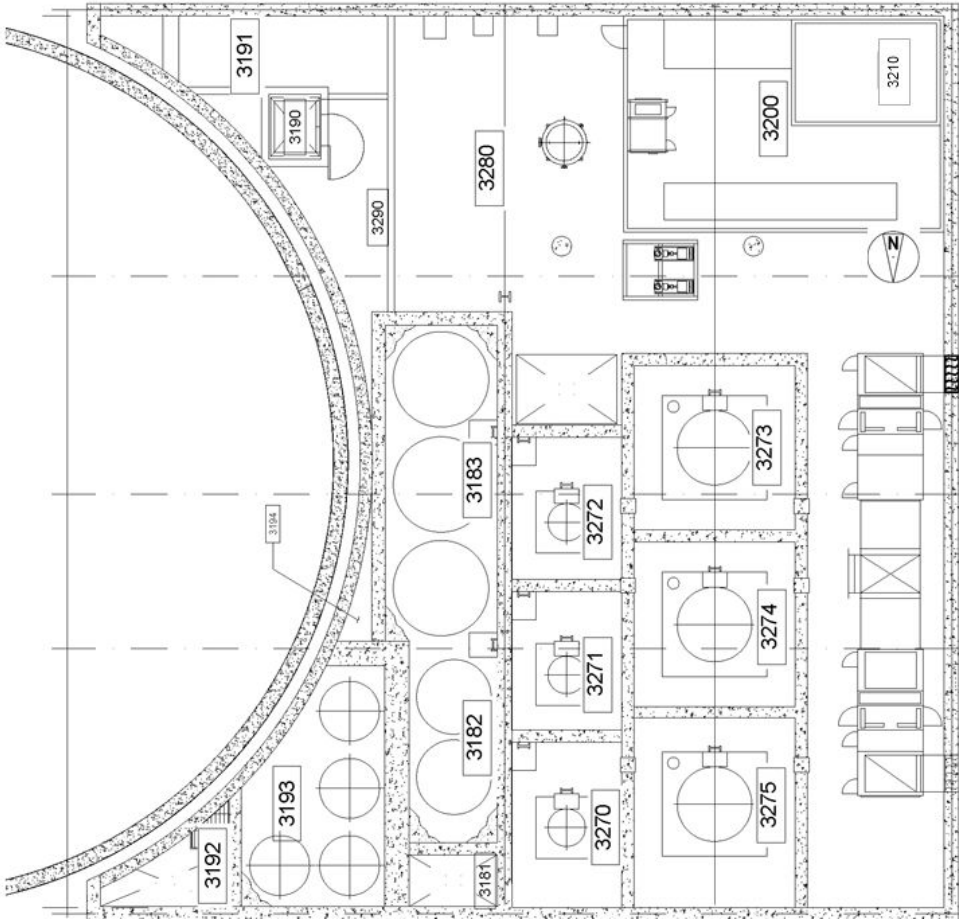
Radwaste Building Radiation Zones Level 0.0			
ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
3100	DRESS OUT AREA / EME EQUIPMENT	A	A
3171	SLUDGE 1 (K20)	H	H
3172	SLUDGE 2 (K20)	H	H
3173	FILTERING SKID AREA	E	E
3174	HIGH INTEGRITY CONTAINER 1 (K20)	G	G
3175	HIGH INTEGRITY CONTAINER 2 (K20)	G	G
3176	HIGH INTEGRITY CONTAINER 3 (K20)	G	G
3177	DEWATERING PUMP ROOM	H	H
3178	DRUM EVAPORATOR (K20)	E	E
3179	SPENT RESIN (K20)	I	I
3180	OFFGAS ABSORBER VESSEL AREA	E	E
3181	SAMPLE (K10) TANKS ENTRY AREA	D	D
3182	SAMPLE (K10) TANKS	D	D
3183	COLLECTION (K10) TANKS	D	D
3190	ELEVATOR	B	B
3191	STAIRWELL A	B	B
3192	FLOOR AREA	B	B
3194	PIPE CHASE*	Unoccupied/Controlled Access	
3195	REACTOR BUILDING EGRESS PATHWAY	B	B

* Access to the chases requires an explicit authorization permit

- A $\leq 6 \mu\text{Sv/h}$ (0.6 mrem/h) UNCONTROLLED & UNLIMITED ACCESS
- B $\leq 10 \mu\text{Sv/h}$ (1 mrem/h) CONTROLLED & UNLIMITED ACCESS
- C $\leq 50 \mu\text{Sv/h}$ (5 mrem/h) CONTROLLED & LIMITED ACCESS (20 h/wk)
- D $\leq 250 \mu\text{Sv/h}$ (25 mrem/h) CONTROLLED & LIMITED ACCESS (4 h/wk)
- E $\leq 1 \text{ mSv/h}$ (100 mrem/h) CONTROLLED & LIMITED ACCESS (1 h/wk)
- F $\leq 10 \text{ mSv/h}$ (1 rem/h) CONTROLLED & LIMITED ACCESS¹
- G $\leq 100 \text{ mSv/h}$ (10 rem/h) CONTROLLED & LIMITED ACCESS¹
- H $\leq 1 \text{ Sv/h}$ (100 rem/h) CONTROLLED & LIMITED ACCESS¹
- I $\leq 5 \text{ Sv/h}$ (500 rem/h) CONTROLLED & LIMITED ACCESS¹
- J $> 5 \text{ Sv/h}$ (500 rem/h) INACCESSIBLE AREA

1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

Figure 12.6-15: Radwaste Building Level 0.0 Meters Radiation Zone



Radwaste Building Radiation Zones Level 6.1			
ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
3181	SAMPLE (K10) TANKS ENTRY AREA	D	D
3182	SAMPLE (K10) TANKS	D	D
3183	COLLECTION (K10) TANKS	D	D
3190	ELEVATOR	B	B
3191	STAIRWELL A	B	B
3192	FLOOR AREA	B	B
3193	UPPER OFFGAS ABSORBER VESSEL AREA	E	E
3194	PIPE CHASE*	Unoccupied/Controlled Access	
3200	LABORATORY	B	B
3210	COUNTING ROOM	B	B
3270	CONDENSATE DEMINERALIZER PRE-FILTER 1	E	E
3271	CONDENSATE DEMINERALIZER PRE-FILTER 2	E	E
3272	CONDENSATE DEMINERALIZER PRE-FILTER 3	E	E
3273	CONDENSATE POLISHER 3 (DEMINERALIZER)	D	D
3274	CONDENSATE POLISHER 2 (DEMINERALIZER)	D	D
3275	CONDENSATE POLISHER 1 (DEMINERALIZER)	D	D
3280	RADWASTE BUILDING 2ND FLOOR AREA	B	B
3290	REACTOR BUILDING EGRESS PATHWAY	B	B

* Access to the chases requires an explicit authorization permit

- | | | | |
|---|-------------|--------------|--|
| A | ≤ 6 μSv/h | (0.6 mrem/h) | UNCONTROLLED & UNLIMITED ACCESS |
| B | ≤ 10 μSv/h | (1 mrem/h) | CONTROLLED & UNLIMITED ACCESS |
| C | ≤ 50 μSv/h | (5 mrem/h) | CONTROLLED & LIMITED ACCESS (20 h/wk) |
| D | ≤ 250 μSv/h | (25 mrem/h) | CONTROLLED & LIMITED ACCESS (4 h/wk) |
| E | ≤ 1 mSv/h | (100 mrem/h) | CONTROLLED & LIMITED ACCESS (1 h/wk) |
| F | ≤ 10 mSv/h | (1 rem/h) | CONTROLLED & LIMITED ACCESS ¹ |
| G | ≤ 100 mSv/h | (10 rem/h) | CONTROLLED & LIMITED ACCESS ¹ |
| H | ≤ 1 Sv/h | (100 rem/h) | CONTROLLED & LIMITED ACCESS ¹ |
| I | ≤ 5 Sv/h | (500 rem/h) | CONTROLLED & LIMITED ACCESS ¹ |
| J | > 5 Sv/h | (500 rem/h) | INACCESSIBLE AREA |
1. Areas designated as "Controlled Infrequent Access" require explicit authorization permits

Figure 12.6-16: Radwaste Building Level 6.1 Meters Radiation Zone

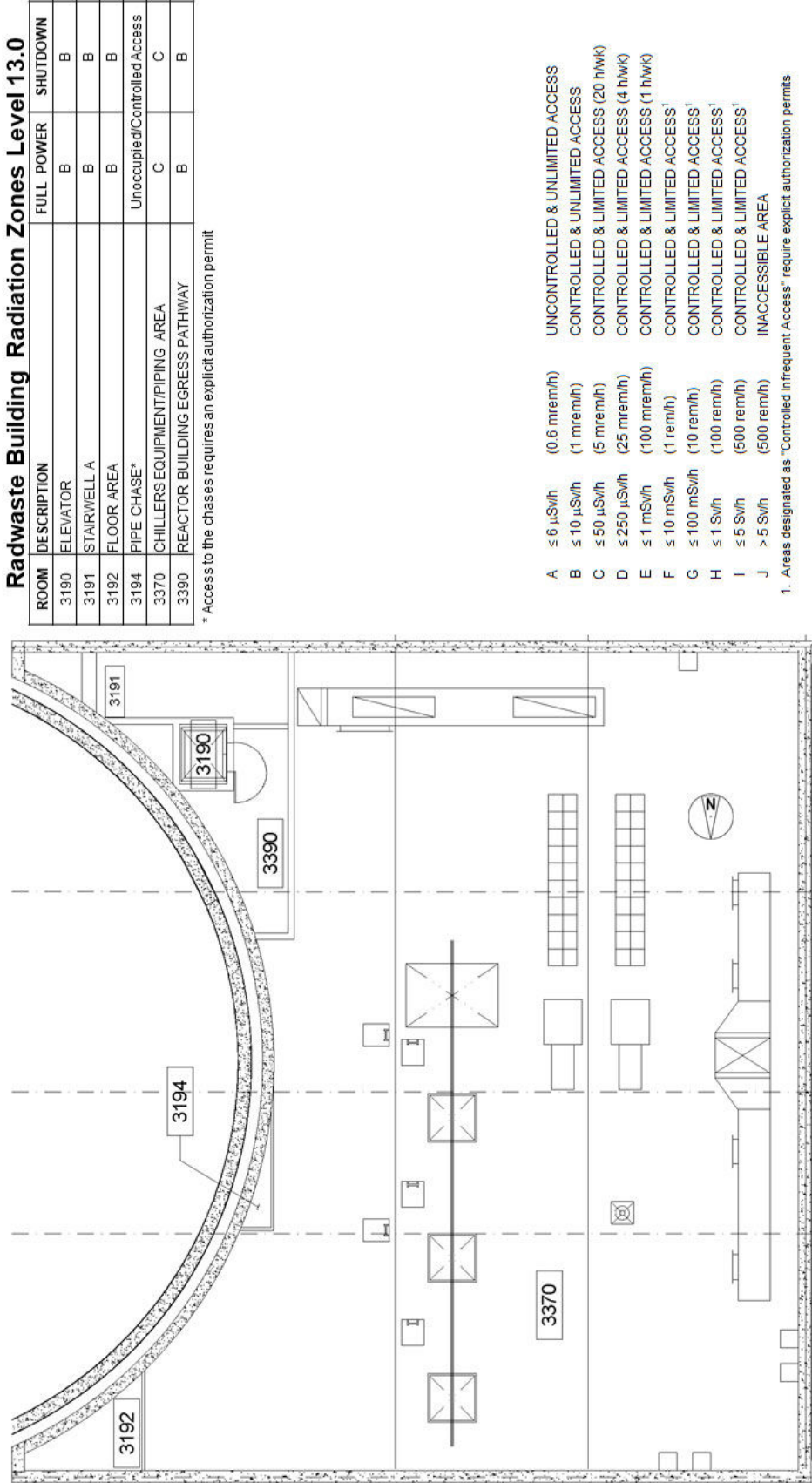


Figure 12.6-17: Radwaste Building Level 13.0 Meters Radiation Zone

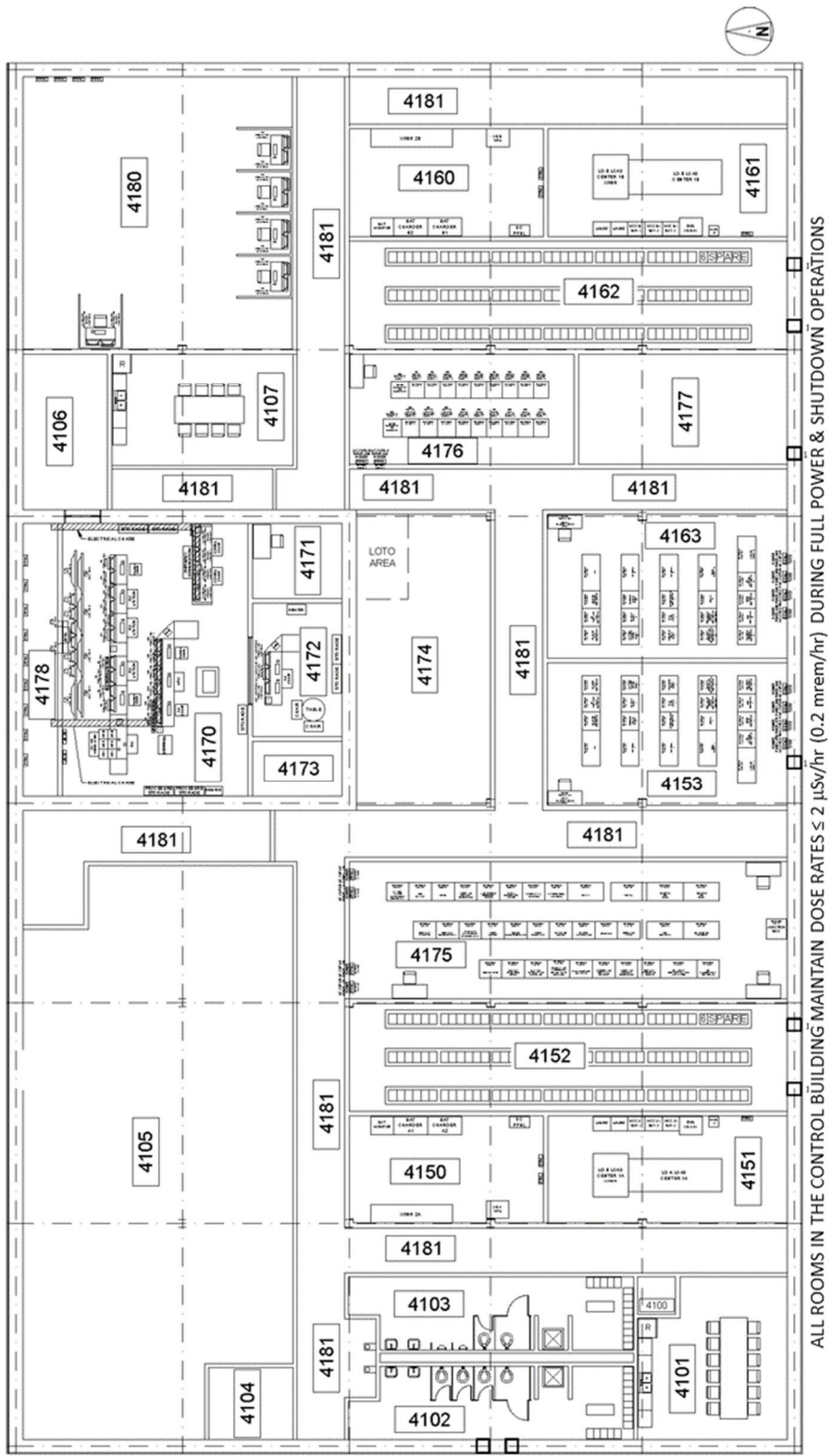


Figure 12.6-18: Control Building Level 0.0 Meters Radiation Zone

12.7 Radiation Protection Program

The Radiation Protection Program is described in the Licence to Construct Application, Section 4.7, Radiation Protection, and is expected to be revised with the Licence to Operate application.

The Radiation Protection Program is consistent with the guidance set forth in CNSC REGDOC-2.7.1, "Radiation Protection" (Reference 12.7-1). The Radiation Protection Program consists of a series of procedures and programs that keep radiation exposure to workers and the public as low as reasonably achievable.

The Radiation Protection Program includes the following elements:

- Radiation protection organization, oversight, and administration
- Training and qualifications
- Process and procedures
- Radiation zones classification
- Program measures to limit exposure and dose
- Equipment and instrumentation
- Contamination control
- Emergency planning

12.7.1 References

12.7-1 CNSC Regulatory Document REGDOC-2.7.1, "Radiation Protection."



HITACHI

GE Hitachi Nuclear Energy

NEDO-33963

Revision 0

September 30, 2022

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**Ontario Power Generation Inc.
Darlington New Nuclear Project
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 13
Conduct of Operations**

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
0	All	Initial Release

ACRONYM LIST

Acronym	Explanation
AM	Aging Management
AOO	Anticipated Operational Occurrence
BWR	Boiling Water Reactor
CNSC	Canadian Nuclear Safety Commission
CSA	CSA Group
DBA	Design Basis Accident
DNNP	Darlington New Nuclear Project
EME	Emergency Mitigating Equipment
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
GEH	GE Hitachi Nuclear Energy
HFE	Human Factors Engineering
IAEA	International Atomic Energy Agency
OPEX	Operating Experience
OLC	Operational Limits and Conditions
OPG	Ontario Power Generation
PSA	Probabilistic Safety Assessment
QA	Quality Assurance
SAMG	Severe Accident Management Guideline
SSC	Structures, Systems, and Components

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None.

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Operating Organization

The prime responsibility for safety is assigned to OPG's operating organization. This responsibility includes covering all activities related to operation directly and indirectly and the supervision of activities of all other related groups, such as design, supply, manufacture and construction, employers, and contractors, as well as the operating organization itself. This responsibility is discharged in accordance with the management system.

This section contains information addressing the following:

1. Design principles used to develop the organizational structure (e.g., layers of hierarchy, length of decision-making chains, scope of managerial control, policy for use of contracted resources)
2. Description of relationships between organizations having significant interaction with information on how any potential effect on nuclear safety management (each relationship) is recognized and addressed
3. Organizational approach taken to ensure capabilities necessary to provide nuclear safety and ensure the integrity of the safety case, including how sufficient in-house core capability is retained to:
 - a. Manage the licenced facility and activities
 - b. Prevent over-reliance on contractors and degradation of in-house capabilities
 - c. Maintain subject matter expertise for all topics necessary for nuclear safety, including "informed (intelligent) customer" roles when expertise is contracted out
 - d. Be an "informed (intelligent) customer" for items or services procured
 - e. Ensure the organization maintains sufficient numbers of qualified workers and identifies nuclear safety-related positions and underpinning roles
 - f. Control organizational changes and maintain the organizational charts as evergreen documents
 - g. Set strategies to ensure the right resources are available at the right time with the right skills and experience to meet core capabilities at all stages of the facility lifecycle and provide for review of implementation and ongoing reviews
 - h. Describe how organizational aspects that lead to vulnerabilities are identified and mitigated (e.g., reliance on scarce or singular areas of expertise)
4. Organization control and how activities will not be subject to undue influence by other organizations
5. Description of how the resource strategy is proactively managed when project work is being implemented to ensure that the resource profiles and organizational arrangements remain fit for the purpose
6. Description of how contracted work is conducted to required levels of safety and quality from an organizational perspective.
 - a. Considerations to address:
 - i. Effective supply chain strategy for delivery of safety case requirements

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- ii. “Informed (Intelligent) customer” capability for all work that may affect nuclear safety carried out by contractors or suppliers or their supply chain
- iii. Issuing specifications to contractors or suppliers that adequately describe the items or services, meet the safety case requirements, and identify the required level of Quality Assurance (QA)
- iv. Evaluation and confirmation before placing a contract with nuclear safety significance that the contractors and suppliers have the organizational, technical and project management capability, capacity, and culture to deliver the items or services

OPG manages organizational changes through a reviewed and approved change process.

13.1.1 Organizational Structure

The BWRX-300 is a simple and safe design above traditional Boiling Water Reactors (BWR), requiring a smaller organization while still meeting the requirements of International Atomic Energy Agency (IAEA) Safety Guide NS-G-2.4, “The Operating Organization for Nuclear Power Plants” (Reference 13.6-1). The organizational structure framework is expected to be defined and included with the Licence to Operate application submission with all details, including roles and responsibilities, finalized and in place prior to receipt of the Licence to Operate.

Upper tier management staffing levels are expected to be similar to those at existing OPG facilities. For a single BWRX-300, it is expected that some roles are combined from traditional operating models and could change if additional units are built on the same site.

Staffing levels required to operate the BWRX-300 are expected to be defined based on the safety analysis (with consideration and integration of Human Factors Engineering (HFE)), Maintenance Program, and outage programs as they become better defined. Staff performing operations and maintenance are expected to be qualified as determined using training analysis, using assumptions and findings of HFE analyses. The minimum staffing level complement is expected to be determined in accordance with an analysis performed in accordance with the requirements of CNSC REGDOC-2.2.5, “Minimum Staffing Complement” (Reference 13.6-2).

The Plant Manager is expected to be accountable to OPG management, the Canadian Nuclear Safety Commission (CNSC), and the public to ensure the facility is operated and maintained with due diligence and in a manner consistent with the Power Reactor Operating Licence, and within the social licence objectives set by OPG.

The Operations and Maintenance Manager(s) ensure all aspects of the managed systems for operations and maintenance are implemented. The number of Operations and Maintenance Managers is expected to be defined as the staffing levels and programs are defined. For a single BWRX-300, we propose a single manager who was previously certified or licenced at a nuclear facility. The Maintenance Manager is not required to be previously certified or licenced at a nuclear facility.

The Shift Manager is accountable to ensure the facility is operated within its Operating Licence.

The site organization is augmented with support from the fleet organization, which includes the engineering component. The fleet organization program update framework is expected to be further defined as the design is progressed and included in the Licence to Operate application submission with all programs put in place prior to receipt of the Licence to Operate. Areas in the submission will include:

- Components and Equipment Surveillance

- Major Components
- Equipment Reliability
- Reactor Safety Program
- Aging Management (AM)
- Risk and Reliability
- Chemistry
- Welding
- Environmental Qualification
- Pressure Boundary

13.1.2 Qualifications of Plant Personnel

The Plant Manager and Operations Manager positions assigned to the BWRX-300 are expected to be filled by staff who have been previously certified or licenced at a nuclear power plant. This could include Small Modular Reactor, BWR, CANada Deuterium Uranium (CANDU), or Pressurized Water Reactor experience.

Qualifications are expected to be developed for each role in the organization according to the Systematic Approach to Training. Role documents defining specific job responsibilities are expected to be developed as appropriate based on the importance of the specific position.

13.2 Training

13.2.1 General

The BWRX-300 training program is developed using a Systematic Approach to Training process that complies with the prescribed regulatory training requirements of CNSC REGDOC-2.2.2, "Personnel Training" (Reference 13.6-3). The OPG personnel training programs ensure worker competence and qualification to perform the duties of their positions.

The OPG training system is developed and implemented to adhere to two fundamental principles:

1. Performance based training is focused on the essential knowledge, skills, and safety attributes required to meet the job requirements (derived from HFE task analysis) and nuclear safety specific needs throughout the lifecycle of the facility.
2. Systematically developed training is defined, produced, and maintained through an iterative and interactive series of steps, leading from the identification and satisfaction of a training requirement.

Training requirements are applied in a manner commensurate with risk. All training requirements apply, but associated training-related processes and procedures may vary based on the safety significance and complexity of the work being performed. The training systems/programs and requirements include:

1. Identification of the performance requirements of a specific job or duty area by conduct of a job task analysis
2. General worker training, initial job training, and continuing training based on a task analysis of the knowledge and skills required to perform each task and any attributes related to safety
3. Training designed, developed, and implemented to meet qualification requirements
4. Trainers meet and maintain documented qualification requirements
5. Formal evaluations used to confirm and document workers are qualified to perform their duties
6. Training change management process that systematically analyzes procedural, equipment, and job description changes (including operational experience feedback) that may require changes to tasks and lead to training modifications
7. Continuing training deemed necessary during the job and task analysis and training needs analyses processes
8. Periodic training program evaluations, with results incorporated into the training improvement process
9. Creation and maintenance of worker training and qualification records
10. Assurance that workers receive the level of training related to nuclear safety that corresponds to their employment and position duties; including, but not limited to radiation safety, conventional safety, fire safety, and on-site emergency arrangements

In addition, training programs are established for initial personnel certification and maintenance of regulatory certifications. Initial and continuing certification training programs are implemented in accordance with the principles of a Systematic Approach to Training.

Positions requiring regulatory certification are expected to be defined based on the technology needs and safety significance. Certification programs are expected to be developed as part of

the design process and only one certification program is required to meet the regulatory requirements.

13.2.2 Training Managed System Plan

All training of personnel is expected to be designed, developed, and delivered using a Systematic Approach to Training.

A full-scope simulator, a replica of the Main Control Room panels, is expected to be utilized to train and qualify control room Operations staff. This approach allows the operators to interface with the simulated plant system in the Main Control Room environment.

13.2.2.1 Minimum Staffing

A minimum staff complement program is established to ensure sufficient numbers of qualified workers are present to meet regulatory and facility licence requirements during all credible events in the BWRX-300 safety analysis. The minimum complement staffing numbers are expected to be defined following completion of detailed design and safety analysis, and to be part of the Licence to Operate application submission.

The basis for the minimum staff complement is determined by a systematic staffing analysis as described in Chapter 18, Subsection 18.2.5. The analysis to determine the minimum staff complement considers:

1. Actions required in the facility and their timing for the full range of the most resource-intensive conditions
2. Resource-intensive initiating events and credible failures considered in the safety analysis report and the Probabilistic Safety Assessment (PSA) (with HFE considerations)
3. Operating strategies that define how the nuclear facility personnel respond to Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and emergencies
4. Required interactions among facility personnel for the purpose of diagnosing, planning, communicating, coordinating, and controlling AOOs, DBAs, and emergencies
5. Staffing demands required for the possible concurrent use of procedures related to AOOs, DBAs, and emergencies
6. Staffing demands required to monitor indicators, displays, and alarms and to promptly and effectively operate the facility's equipment controls using procedures related to AOOs, DBAs, and emergencies
7. Staffing demands required to perform tasks in field locations using procedures related to the events considered within the scope of the analysis
8. Staffing demands required for the successful completion of any important human actions using procedures related to the events considered within the scope of the analysis
9. Restrictions on the location of workers within the nuclear facility

The minimum staff complement requirements are validated to provide assurance that there are sufficient numbers of qualified workers available to operate the facility safely and respond to the most resource-intensive conditions at all times. Validation of the minimum staff complement is in accordance with the verification and validation processes described in CNSC REGDOC-2.5.1, "General Design Considerations: Human Factors" (Reference 13.6-4).

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The minimum staffing requirements are expected to be formalized in a procedure that describes:

1. The specific number of staff to be present on-site, in the facility, and in the Main Control Room, and the composition of the minimum staff complement with reference to specific positions or qualifications
2. Modifications to minimum staff complement for different operational states and the specific number and composition of the minimum staff complement with reference to specific positions or qualifications for each operational state
3. Any specific restrictions on the location of individuals in the facility
4. Measures in place to monitor compliance with the minimum staff complement and to prevent non-compliance with the minimum staff complement
5. Specific actions to be taken to reduce the risk to the facility in the event of non-compliance with the minimum staff complement

13.3 Implementation of the Operational Safety Program

13.3.1 General Implementation

The OPG Managed System is implemented under Nuclear Management System N-CHAR-AS-0002 (Reference 13.6-5). The system is implemented by a series of program documents which in turn define the required implementing procedures and standards. The Managed System is designed to be fully compliant with CSA Group (CSA) N286, "Managed System Requirements for Nuclear Facilities" (Reference 13.6-6). As such, all implementing procedures and standards ensure that all aspects of CSA N286 are fulfilled, as well as being fully compliant with all CNSC REGDOCs. The Managed System framework associated with BWRX-300 plant operation is expected to be outlined and included with the Licence to Operate application, with all programs finalized and in place prior to receipt of the Licence to Operate.

Changes to the Managed System are made in accordance with Nuclear Management System Administration, N-PROG-AS-0001 (Reference 13.6-7). Changes to the current charter (N-CHAR-AS-0002) (Reference 13.6-5) are expected to be required to accommodate BWRX-300 technology.

13.3.2 Conduct of Operations

This section addresses important operational issues relevant to safety throughout the lifetime of the plant and how the operating organization addresses identified issues adequately.

The OPG Nuclear Management System sets the standards for health, safety, environment, security, economics, and quality during facility design, construction, commissioning, and operation based on the authority of and a safety culture driven by the OPG Nuclear Safety Policy. The OPG Nuclear Management System promotes the safety culture by committing workers to adhere to the OPG Nuclear Management System, implementing practices that contribute to the excellence in worker performance, supporting workers in carrying out their tasks safely and successfully, and monitoring to improve the culture. The organizational structure implements the programs that make up the OPG Nuclear Management System with the Chief Nuclear Officer accountable for implementation and effectiveness of the OPG Nuclear Management System. The outline of the programs and standards utilized for operating the plant is expected to be included with the Licence to Operate application submission, with all program details and standards finalized prior to receipt of the Licence to Operate.

The OPG Nuclear Management System is based on a set of principles implemented in a graded approach consistent with CNSC REGDOC-2.1.1, "Management System" (Reference 13.6-8) and CSA N286 (Reference 13.6-6) guidelines.

13.3.2.1 Safety Culture

Consistent with CNSC REGDOC-2.1.2 "Management System: Safety Culture" (Reference 13.6-9), the safety culture is established, promoted, communicated, and fostered by Senior Management through the OPG Nuclear Safety Policy and OPG Nuclear Management System. The safety culture is applicable to all activities that affect the health and safety of workers, the public, and the environment in every phase of the facility life cycle.

The OPG Nuclear Management System is maintained in accordance with the requirements of CSA N286 (Reference 13.6-6).

The OPG Nuclear Management System meets the CNSC's Safety and Control Area principles and regulatory requirements necessary to protect health, safety, and the environment.

The safety culture is implemented, monitored, and periodically assessed consistent with CNSC REGDOC-2.1.2 (Reference 13.6-9) and CSA N286 (Reference 13.6-6) guidelines through

subordinate policies, programs, processes, and procedures that implement the varied administrative, maintenance, and operational aspects of facility operation. An established program, that summarizes OPG's internal and external processes used for oversight and assessment, tracks assessment action items and monitors various metrics that may reveal safety culture aspects (e.g., Operating Experience (OPEX), performance trends, condition reports, regulatory inspections). OPG's Human Performance and Performance Improvement programs also implement OPG expectations for understanding and promoting a strong safety culture.

A more detailed discussion of the OPG Nuclear Management System is provided in Chapter 17, Sections 17.1 and 17.2.

13.3.2.2 Services and Equipment Acquisition/Receipt

The OPG supply chain process is established and controlled. OPG supply chain services are responsible for establishing and maintaining an OPG nuclear approved supplier list. Periodic audits are performed to confirm the initial and ongoing acceptability of the supplier's management system. The OPG process/program describes methods used to originate, request, evaluate, qualify, and maintain qualification of suppliers of items and services required for QA programs or other OPG nuclear quality requirements.

Suppliers are assessed on their ability to meet purchasing requirements and have the organizational, technical and project management capability, capacity, and culture to deliver the item or service.

The supplier-customer relationship is monitored to ensure purchasing requirements are met. Monitoring includes the alignment of demand and supply signals between OPG and the supplier, a supplier/customer performance assessment, involvement of the supplier in customer demand planning, reporting requirements for delays or defects and supplier involvement in obsolescence and remediation. Monitoring results are used as input in determining the frequency and extent of inspection, verification, and audit activities. The audit program is established as part of the QA Program and is discussed further in Chapter 17, Sections 17.2 – 17.4.

OPG specifies the requirements for purchased expertise and equipment, provides work oversight, and technically reviews the output before, during, and after implementation. Contractors within the supply chain are also audited on a regular basis as part of the contractual agreements.

Components are checked when initially received to ensure the components are as ordered, undamaged, and are not fraudulent, counterfeit, or suspect. The components are subjected to more detailed inspection for acceptability prior to use. After receipt, the components are stored to protect against construction activities, physical and environmental damage, and deterioration.

13.3.2.3 Fitness for Service

The fitness for service safety and control area covers activities that affect the physical condition of Structures, Systems, and Components (SSC) to ensure adequacy and ability to perform their intended functions when required. Fitness for service is addressed in established programs that include Reliability, Maintenance, AM, Chemistry Control, Periodic Inspections, and In-Service Inspections.

Programmatic requirements addressing fitness for service span the full life cycle of the facility, beginning with inclusion in facility design decision-making and consideration during each phase (e.g., design, construction, commissioning, operation) of the facility's life. Requirements evolve as the facility ages and specific process requirements may vary based on the life cycle phase (e.g., construction versus operation).

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Reliability is incorporated during facility design, consistent with the requirements of CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 13.6-10) and through the Reliability Program that complies with CNSC REGDOC-2.6.1, "Reliability Programs for Nuclear Power Plants" (Reference 13.6-11). The Reliability Program is implemented to ensure that systems function reliably in accordance with design and performance criteria. Although the Reliability Program focuses primarily on the facility operational phase, it applies to all phases of the facility life cycle. The Reliability Program includes:

- Identification and categorization of systems using a systematic process
- Identification of specific failure modes and specification of reliability targets
- Specification of minimum capability and performance level consistent with safety targets and regulatory requirements
- Provisions for information incorporation into maintenance programs
- Provisions for inspection, tests, modeling, and monitoring to assess reliability based on safety class
- Documentation of program activities, attributes, elements, results, and administration

The facility Maintenance Program establishes a maintenance strategy, based on the plant design and safety analysis, to ensure that SSC function as designed. The facility Maintenance Program is implemented by a maintenance organization, established consistent with CNSC REGDOC-2.6.2, "Maintenance Programs for Nuclear Power Plants" (Reference 13.6-12). A systematic approach is used to identify the SSC maintenance activities to be performed and the maintenance intervals.

The Maintenance Program describes the processes for planning, monitoring, scheduling, and executing maintenance work activities, including those maintenance activities performed during the construction and commissioning phases. Surveillances conducted as part of the Maintenance Program, including acceptance criteria, are addressed in Chapter 16, Sections 16.2 and 16.4, Operational Limits and Conditions.

An AM Program conforming to the requirements of CNSC REGDOC-2.5.2 (Reference 13.6-10) and CNSC REGDOC-2.6.3, "Aging Management" (Reference 13.6-13) is established to ensure the reliability and availability of the required SSC safety functions throughout the facility service life.

The effects of aging and wear are taken into consideration during the design of Safety Class SSC. The considerations include:

1. Design margin assessment that considers the known aging and wear mechanisms potential degradation in operational states, to include the effects of testing and maintenance
2. Provisions for monitoring, testing, sampling, and inspecting SSC to assess aging mechanisms and identify degradation that may occur during operation a result of aging and wear
3. Online monitoring to provide forewarning of degradation leading to failure and where failure could be safety significant

Details regarding AM design provisions are provided in Chapter 3, Subsection 3.1.12.

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Chemistry Control policies and goals are established to:

- Preserve the integrity of SSC
- Minimize the effects of chemical impurities and corrosion on SSC
- Implement As Low as Reasonably Achievable to manage radioactive material buildup
- Limit release of chemicals and radioactive material to the environment

Chemistry Control governs the development and maintenance of chemistry procedures, specifications, and methods of control. Knowledgeable and trained staff are assigned to monitor for abnormal trends so that action can be taken to ensure operations within specified limits. Performance indicators are maintained to satisfy reporting requirements.

Included in Chemistry Control are requirements for:

- Data management (to include trending, evaluation, and reporting of analysis results and investigations)
- Chemistry surveillance program
- Chemistry specifications for systems
- Procedures for chemistry parameter selection, monitoring, analysis, and trending
- Procedures for operations involving chemistry processes and evaluation of results
- Operation and reference limits for chemistry parameters and associated action levels
- Chemical Control Program
 - Training (chemical hazards, labeling and storage)
 - Procedures for the storage and handling of chemicals
 - Approval, procurement, and receipt of chemicals
 - Listing of chemicals approved for site use and those that are precluded from site use or other classification
 - Administrative controls for controlling products in the workplace

The Chemistry and Chemical Control programs, as applicable to construction and commissioning, are described in Chapter 14, Subsection 14.2.4.

Periodic and in-service inspection and testing programs are established in conformance with CNSC REGDOC-2.5.2 (Reference 13.6-10) to confirm that service-induced degradation has not increased the likelihood of a failure of a barrier against the release of radioactive material.

Periodic and in-service inspection and testing are established for:

- Nuclear pressure boundary components
- Containment components
- Containment structures
- Safety-related structures
- Balance-of-plant pressure boundary Safety Class components or based on AM requirements

13.3.2.4 Nuclear Material Packaging and Transport

Processes and procedures are expected to be established that address the safe packaging, registration, and transport of nuclear substances to and from the facility as described in OPG's Radioactive Material Transportation Program. The program ensures shipping packages are designed and maintained to ensure protection and containment of the quantities of nuclear material transported. In addition, package certification, package testing, inspection, and maintenance are addressed within the program. This program is expected to be established prior to fuel delivery to the Darlington site, an activity which will be subject to separate licensing by CNSC, as the Licence to Construct scope does not include transport, import, possession, or storage of nuclear fuel.

13.3.3 Maintenance, Surveillance, Inspection and Testing

This section provides a description and justification of arrangements that the operating organization has in place to identify, control, plan, execute, audit, and review maintenance, inspection, and testing practices that influence reliability and affect nuclear safety.

SSC credited in the safety analysis are identified and periodically tested (surveilled) at a frequency related to the results of reliability analysis and operational experience to ensure that they will function as required. SSC performance that is inconsistent with assumptions in the safety analysis are identified. Following modification to SSC, the test requirements are re-evaluated. Defense Line 3 (Safety Category 1) SSCs credited in the deterministic safety analysis are addressed in the Operational Limits and Conditions (OLC) of Chapter 16 and the Defense Line 2 and 3 SSCs credited in the deterministic safety analysis are addressed in a program required by Chapter 16. Furthermore, additional SSCs credited in the probabilistic safety analysis will be addressed in the OLC if its failure is a significant contributor to Core Damage Frequency. This defence-in-depth approach provides reasonable assurance the consequences of postulated initiating events are bounded by Chapter 15 results and safety goals are met

SSC are maintained in accordance with a maintenance strategy defining the frequency and type of maintenance to be performed, taking into consideration the supplier recommendations, safety analysis, periodic inspection requirements, OPEX, cost benefit analysis, and service conditions. Maintenance activities are performed in accordance with approved procedures and practices. Preventive measures are employed to eliminate structural, system, and component damage or the contamination of systems with foreign material. In addition, predictive maintenance is performed based on plant monitoring system information. Maintenance includes the repair or replacement of malfunctioning SSC as needed to re-establish conformance with requirements.

A Maintenance Program is to be implemented consistent with the requirements of CNSC REGDOC-2.6.2 (Reference 13.6-12) to address:

1. Measures, policies, methods, and procedures providing direction for maintaining SSC capable of maintaining their functions as described in design documents and the safety analysis
2. Processes for planning, monitoring, scheduling, and executing work activities so SSC continue to meet design intent and remain fit for service in the presence of degrading mechanisms
3. Preventive maintenance activities, maintenance processes and record retention requirements, corrective maintenance, calibration of measuring and monitoring devices, SSC monitoring (activity optimization), outage management, work planning and scheduling, work execution, maintenance procedures, post-maintenance verification and testing and Maintenance Program assessment

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4. Predictive maintenance based on plant monitoring system information
5. Surveillance program covering OLC, with surveillance frequencies based on a reliability analysis, a PSA, and previous OPEX, and that shows viability of inspection techniques to meet performance requirements while taking As Low as Reasonably Achievable into account
6. Approach taken to develop SSC surveillance program acceptance criteria
7. Assurance that the surveillance program is adequate to ensure the inclusion of all relevant aspects of the OLC
8. Timeline for the development of each program with milestones for development and implementation of each program and the processes followed
 - a. Results of each activity to be reviewed against acceptance criteria and with periodic reviews to ensure the program continues to meet objectives.

Multiple aspects of the surveillance, inspection, and testing program are addressed within OLC (Chapter 16, Section 16.4), to include:

1. Safety Class plant items that require monitoring to ensure they remain fit for their purpose and operation is within the operational limits for reliable and safe operation
2. How surveillance, maintenance and repair ensure OLC parameters remain within acceptable limits and systems/components are operable
3. Surveillance frequency basis on a reliability analysis, including where available, a PSA and a study of experience gained from previous surveillance results (in the absence of both, the surveillance is based on supplier recommendations)
4. System for ensuring testing is performed and confirmed within the timelines allowed

13.3.4 Core Management and Fuel Handling

13.3.4.1 Core Management

The programs and procedures that govern the operational activities associated with BWRX-300 core management regarding fuel reliability are based on guidelines established by GEH, utilizing decades of experience with fuel from Global Nuclear Fuel. Fuel related design aspects, including operational, transient, and accident limits, are discussed in Chapter 4, Sections 4.2 and 4.4. The core/fuel management guidelines are implemented through operational methods implemented to mitigate and reduce duty related fuel performance risks.

In general, the BWRX-300 operational methods employ an approach of limiting the duration of low power periods and limiting the rate at which power is raised following prolonged operation at low power. When raising power, a combined approach of an unrestricted power increase to an established threshold or prior conditioned power envelope, followed by raising power to a final value at a defined, controlled, slow ramp rate, is used.

The operational practices are based on BWRX-300 operational experience:

1. An established exposure dependent Linear Heat Generation Rate threshold, below which no power maneuvering restrictions are applied with power increases above the threshold limited to a defined controlled rate
2. Power envelopes (conditioned power) established by the maintenance of specific power conditions sustained for a defined period

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3. Defined power ramp rates for power increases above the more limiting of a Linear Heat Generation Rate threshold or the conditioning envelope value, performed at a defined, controlled ramp rate
4. Sequence exchange intervals established based on cycle exposure, that consider multiple factors, including fuel reliability
5. Threshold power levels established for fuel bundles or nodes with unusually long periods of low power operation (long control intervals), implemented on a case-by-case basis using industry OPEX best practices
6. Employing power envelopes best practices considering the BWR characteristics of top peaked axial power shapes at the end of fuel cycle and bottom peaked axial power shapes at the beginning of fuel cycle operation
7. Control rod exercising requirements
8. Barrier fuel risk mitigation
9. Established threshold values for fuel with high residence time in central portions of the core

Core Monitoring is a function of the plant computer system that provides three-dimensional core power monitoring to satisfy the requirements of CNSC REGDOC-2.5.2, Section 8.1 (Reference 13.6-10) to ensure the plant operates within the power distribution design basis. Core Monitoring provides confidence that the plant is operating in conformance to specified acceptable fuel design limits. Core Monitoring obtains instrumentation information from the C20 Distributed Control and Instrumentation System (refer to Chapter 7, Subsection 7.3.3.2), calculates thermal power limits, and provides estimates of power distributions. These estimates are calculated by the core simulator.

The Core Monitoring function acquires real-time reactor data from site plant data acquisition systems as necessary to define the reactor state for use by the core simulator. Core Monitoring can calculate the accumulated thermal and electrical energy produced by the plant from the beginning of an operating cycle. The Core Monitoring function is described further in Chapter 4, Subsection 4.6.8.

13.3.5 Aging Management and Long-Term Operation

This section provides a description of an integrated AM Program that will meet the requirements of CNSC REGDOC-2.6.3 (Reference 13.6-13) and CNSC REGDOC-2.5.2, Section 7.17 (Reference 13.6-10). AM processes and plans ensure the reliability and availability of required safety functions of SSC throughout the service life of the facility (Lifecycle Management Plans). Periodic inspection or in-service inspection programs, as they relate to BWRX-300 aspects, are expected to be incorporated directly into AM programs.

AM is addressed during design within the design process. The design provisions for AM are discussed in more detail in Chapter 3, Section 3.1. Consideration is given to the feedback of OPEX. A systematic approach is taken during design to understand the aging of SSC to evaluate design features for aging prevention, monitoring, and mitigation. Mechanical, thermal, chemical, electrical, physical, biological, and radiation aspects are taken into consideration. SSC determined to have shorter service lives than the nominal design life are identified with AM strategies provided in the design documentation. The components that are identified with service lives less than the nominal design life also have replacement plans defined in the plant Maintenance Program, with associated monitoring requirements and provisions to permit their removal and replacement.

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Aging effects under design basis conditions, including transient and postulated initiating event conditions, are also considered in equipment qualification programs.

The design information derived with these design aging considerations establish the baseline for the test data required to be collected and documented for AM Program monitoring and evaluation requirements.

Design documents also identify any special manufacturing or construction processes that are to be applied to prevent, mitigate, or eliminate known aging mechanisms. These provisions are necessary for specification in procurement documents.

The AM Program and processes are used to detect, assess, and manage deterioration of SSC as a result of aging effects such as irradiation, corrosion, erosion, fatigue, and other material degradation.

Descriptions of the following AM Program elements include:

- Organizational arrangements
- Data collection and record keeping
- Screening and selection process for AM
- Evaluations for AM
- Condition assessments
- SSC-specific AM plans
- Management of obsolescence
- Interfaces with other supporting programs
- Implementation of SSC-specific AM plans
- Review and improvement processes for the AM Program

An integrated AM Program ensures that availability and reliability of required safety functions throughout the facility's service life is established. The program requires AM activities to be implemented proactively throughout the life cycle of a nuclear facility in compliance with CNSC REGDOC-2.6.3 (Reference 13.6-13).

13.3.6 Control of Modifications

This section addresses the method of identification for designing, planning, executing, controlling, testing, auditing, reviewing, and documenting modifications to the plant throughout its lifetime, consistent with the guidance provided in IAEA SSG-61, "Format and Content of the Safety Analysis Report for Nuclear Power Plants," (Reference 13.6-14). The modification control process covers all safety significant changes (permanent and temporary) made to SSC, OLC, plant procedures, and process software. The design and safety analysis is incorporated into the purchasing, construction, commissioning, operating, and maintenance documentation such that the as-built configuration of the facility is aligned with the design and safety analysis. Design authority configuration requirements, including the responsibilities and authority of organizations whose functions affect the configuration of the facility, including activities such as design, maintenance, construction, licensing, and procurement, are controlled through its Configuration Management System. A series of programs, including engineering change control, design management, and software, ensures plant configuration is controlled in a manner that is analyzed to be safe. Control of modifications and configuration management during construction and commissioning phases is discussed further in Chapter 14, Section 14.1.

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The information provided includes descriptions of:

1. Modification control process for maintaining the design basis, taking into account new information, OPEX, safety analyses, resolution of safety issues, or correction of deficiencies
2. Description of how design changes are assessed, addressed, and accurately reflected in the safety analyses or record prior to implementation

The plant modification control process covers:

1. Changes made to plant systems and components, OLC, plant procedures and process software, taking into account the safety significance of the proposed modifications to allow them to be graded and referred to the CNSC when necessary
2. Changes to task performance requirements (task step alterations, expected outcomes, procedure level), personnel job role responsibilities or the operating organization
3. Records retention, and where necessary, revision documentation, procedures, instructions, and drawings to reflect the changes

13.3.6.1 Configuration Management

Configuration management is incorporated into purchasing, operating, and maintenance so that the as-built configuration of the facility aligns with the design and safety analysis in accordance with CSA N286 (Reference 13.6-6) and CSA N286.10, "Configuration Management for High Energy Facilities" (Reference 13.6-15). Configuration management is applied in a graded approach.

Configuration management during the construction and commissioning phases is described further in Chapter 14, Section 14.1.

Configuration management is not a stand-alone program. Configuration management plans are developed and integrated within the OPG Nuclear Management System (e.g., assessment, problem identification and resolution, training). From conception to the end of operations, configuration management ensures data generated during design, construction, and commissioning reflects the design basis and specified requirements are kept current in the design, as-built, and field change documentation.

The design basis and requirements for the BWRX-300, including safety analysis, are established and documented in accordance with CSA N286 (Reference 13.6-6) and are traceable to the respective SSC. Impacts of design changes are assessed, addressed, and when applicable, reflected in the safety analysis. Subsequent changes to the physical and operational configuration are maintained consistent with design requirements and configuration information throughout the operational life cycle. Where SSC requirements exceed functional design requirements (safety margin versus design margin), the process ensures safety margin is maintained for subsequent modifications. Physical assessments of SSC configuration are conducted as part of facility management.

Temporary and permanent changes are managed in accordance with the requirements of CSA N286 (Reference 13.6-6).

Configuration information, the types and sources of configuration information, and associated documentation are controlled and maintained in accordance with CSA N286 (Reference 13.6-6), with the status of changes identifiable. As-built information is turned over prior to commencement of operations (turnover) and subsequent as-built information is incorporated in a timely manner commensurate with the associated risk. The facility design basis is maintained following turnover

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and commissioning by OPG to reflect new information, OPEX, safety analyses, and the resolution of safety issues or deficiency corrections.

Configuration deviations, when identified, are managed through the problem identification and resolution processes consistent with the requirements of CSA N286 (Reference 13.6-6). Deviations are immediately controlled (if required), documented, evaluated for significance, and the underlying cause assessed if deemed systemic and accepted. Problem resolutions are reviewed for effectiveness.

Configuration management objectives and concepts are addressed in the respective training programs, with the necessary links between configuration management and the training programs established and maintained.

Configuration information records and documents are maintained consistent with the requirements of CSA N286 (Reference 13.6-6).

13.3.6.2 Engineering Change Control

OPG engineering change control is an integrated management process that ensures the physical and operational configuration and documentation continue to conform to the design and licensing basis requirements.

Facility configuration is maintained from initial fuel load to the end of operating life through established programmatic configuration and change control processes that adhere to CSA N286 (Reference 13.6-6).

The change control process makes certain that safety limits, design basis, licensing basis, and normal operating margins are controlled under engineering change control to always ensure the facility is operated well within conditions analyzed to be safe. Additionally, this process ensures all changes, from minor parts substitution to safety-related modifications, are controlled to ensure the designed facility is operated with margin.

13.3.6.3 Design Management

The Design Management Program will specify requirements for the following two areas:

1. Management of prescribed activities appropriate for execution and control of required design, design support, and documentation for nuclear facilities and organizations in accordance with CSA N286 (Reference 13.6-6)
2. Processes for creating or modifying documentation required for controlling design bases and design outputs

A minimum set of documentation identifies and describes design bases, design output, and the design process.

13.3.6.4 Software

The Software Program complies with CSA N286 (Reference 13.6-6) and CSA N286.7 "Quality Assurance of Analytical, Scientific, and Design Computer Programs" (Reference 13.6-16), and applies to software classified as Real-Time Process Computing and Scientific, Engineering, and Safety Analysis Software and Software Engineering Tools. The program identifies the processes and overall requirements for classification of software and identifies governing standards for each software classification, defining requirements for software development, maintenance, procurement, qualification, use, and retirement.

13.3.7 Program for the Feedback of Operating Experience

This section describes the program implemented for the feedback of OPEX. The OPEX Program ensures operational events and incidents occurring at the facility and other relevant facilities are captured or identified, recorded, notified, investigated internally, and used to incorporate lessons learned for the operation of the facility.

Relevant OPEX is considered for the BWRX-300 during design, construction, commissioning, operation, maintenance, and decommissioning. The design authority (GEH) establishes provisions for the incorporation of OPEX through Integrated Management Systems. The OPEX comes from a variety of sources, including direct input, GEH/Global Nuclear Fuel experience from operating the BWR and Advanced Boiling Water Reactor fleet, Institute of Nuclear Power Operations, Electric Power Research Institute, Boiling Water Reactor Owners' Group, U.S. Department of Energy, U.S. Nuclear Regulatory Commission, CANDU Owners Group, and CNSC. OPEX associated with the construction and commissioning phases is discussed in Chapter 14, Section 14.1.

Industry OPEX information is routinely made available to or distributed by GEH design and modifications personnel. The more important industry-wide issues are routinely addressed in CNSC Nuclear Incident Reports and U.S. Nuclear Regulatory Commission Generic Letters and Bulletins.

OPG has an established OPEX process for evaluating, integrating, accessing, and sharing OPEX information. The OPEX process addresses implementation of OPEX feedback during design activities and its continuance through the construction, commissioning, and operational phases of the facility life cycle, to include how events are identified, recorded, investigated, and reported; as well as how findings from the events are used to enhance safety performance.

13.3.8 Documents and Records

This section addresses the programmatic provisions for creating, receiving, classifying, controlling, storing, retrieving, updating, revising, and deleting documents, records, and reports relevant to the operation of the facility over its lifetime in accordance with OPG-PROG-0001 "Information Management" (Reference 13.6-17).

Document and records program management is the responsibility of the operating organization.

OPG records management encompasses the control of documents and records with requirements addressed in the Controlled Document Management Program. The OPG process for the control of documents includes the development, validation, and approval of safety-related documents. Documents are available for use at the location where the work is to be performed. Changes to documents are documented and tracked. The OPG process for the control of records ensures that records are readable, complete, identifiable, traceable, retrievable, preserved, and retained as necessary.

Documents are controlled consistent with their intended use and consistent with CNSC REGDOC-2.1.1 (Reference 13.6-8) and CSA N286 (Reference 13.6-6).

The program document ensures that controlled documents include:

- Unique identification
- Defined format and presentation
- Identification of status
- Review for adequacy and approval

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- Availability for use at the location where the work is performed or where the document is required for reference
- Prompt removal of obsolete documents for use

Records are:

- Readable
- Complete
- Identifiable
- Traceable to the related items and work
- Retrievable
- Preserved
- Retained as specified

Document management for the BWRX-300 is controlled under the QA Program during design. The QA Program includes document management aspects. The QA Program requirements during design of the BWRX-300 are established in NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 13.6-18). NEDO-11209-A includes requirements during design, addressing:

- Procurement document control
- Instructions, procedures, and drawings
- Document control
- Control of QA records

Document and records management is discussed with respect to the QA Program in Chapter 17, Section 17.2. Documents and records management during construction and commissioning is discussed in Chapter 14, Section 14.1.

13.3.9 Outages

This section addresses the programmatic aspects of the conduct of periodic reactor shutdowns (outages).

The current reference cycle for DNNP BWRX-300 is based on a nominal 12-month fuel cycle. Different fuel cycle durations can be supported depending on the overall fuel reload strategy to be deployed on a cycle/multi-cycle specific basis.

Outage analysis does not address forced outages, but surveillance and maintenance activities that require the plant to be shut down are minimized to the extent possible, largely by enhanced system reliability achieved through design simplicity.

Outage planning, scheduling, and maintenance activities are managed consistent with the guidance provided in CNSC REGDOC-2.6.2 (Reference 13.6-12), and reporting requirements are consistent with the requirements of CNSC REGDOC-3.1.1, "Reporting Requirements for Nuclear Power Plants" (Reference 13.6-19).

The Work Management Program provides for the implementation of processes and procedures for the planning, scheduling, and execution of maintenance activities. Work planning is conducted at both the overall plant and individual job levels. The Outage Management Program establishes the criteria followed to confirm that planned outage and emergent work is completed satisfactorily.

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In addition to procedures for routine outage maintenance activities, OPG maintains a forced outage plan for emergent conditions.

Outage plans are reviewed for nuclear safety, with work groups reviewing the plans within their area of responsibility and with specific consideration given to:

- Impact on operating units and systems
- Application of controls during infrequently performed tests and evolution to ensure the plant is maintained within the design basis
- Contingency plans for alternate measures to maintain safe shutdown
- Routine review to capture changes from the original plan impact assessment
- Outage OPEX

The cumulative effect of plant equipment taken out of service is taken into consideration to ensure there are no adverse effects on the performance of safety functions when planning and scheduling outage work. In addition, plans to remove equipment from service during an outage include measures to deal with the possible consequences of an event occurring while the equipment is out of service. Clear statements are made to identify when equipment is being taken out of service, to include the duration and impact of removing the equipment from service.

The outline of the Outage Management Program utilized for operating the plant is expected to be included with the Licence to Operate application, with all details finalized prior to receipt of the Licence to Operate.

13.4 Plant Procedures and Guidelines

This section programmatically addresses the relevant documents used by plant staff to ensure that procedures and guidelines for normal operation, AOOs, and accident conditions are followed in the intended manner. Procedure development is a technical element of the BWRX-300 HFE program. The procedure development process is described in Chapter 18, Subsection 18.3.7.

13.4.1 Administrative Procedures

This section describes the administrative procedures that outline the essential elements of the administrative programs used by the operating organization to ensure the safe management of the plant. The processes to develop, approve, revise, and implement the procedures are described along with a list of the relevant procedures.

Administrative procedures contain adequate programmatic controls to provide an effective interface between organizational elements. This includes contractors or organizations providing support to the facility operating organization.

Procedure Writer's Guides promote standardization and the application of HFE usability principles to administrative procedures. Additional details are provided in Chapter 18, Subsection 18.3.7.

Procedural compliance with all administrative procedures ensures all regulatory requirements and standards are met. Procedural steps that implement these specific requirements are flagged with "bases" statements (e.g., [B-1] meaning refer to B-1 for the overriding regulatory and legal requirement). The "Content Authority" is accountable to ensure the administrative procedure meets applicable regulatory requirements and standards. The flagging of bases requirements ensures that regulatory and legal requirements are checked during the continuous improvement (revision) process.

Procedure maintenance and control of procedure updates are performed in accordance with OPG's QA Program processes.

The plant administrative procedures provide for the following:

- Establishment of a formal review and approval process
- Control of equipment, as necessary, to maintain personnel and reactor safety, and to avoid unauthorized operation of equipment
- Control of maintenance and modifications
- Temporary changes
- Temporary procedure issuance and control
- Special orders of a temporary or self-cancelling nature
- Standing orders to shift personnel, including the authority and responsibility of the control room staff
- Manipulation of controls and assignment of shift personnel to duty stations
- Shift relief and turnover procedures
- Fitness for Duty
- Working hour limitations
- Feedback of design, construction, and applicable important industry OPEX
- Shift Manager administrative duties

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- Verification of correct performance of operational activities
- Vendor interface program that provides vendor information for SSC is incorporated into plant documentation
- Fire protection program implementation
- Process for implementing safety/security interface requirements

13.4.2 Operating Procedures

The facility is operated, monitored, and maintained within the safe operating envelope and in accordance with procedures that are consistent with the design. Operating procedures are established to provide for the safe conduct of BWRX-300 normal operations. Normal operation is operation within specified OLC, within one of the following plant operating modes (further defined in Chapter 16, Appendix 16A):

- Mode 1: Power Operation
- Mode 2: Startup
- Mode 3: Hot Shutdown
- Mode 4: Stable Shutdown
- Mode 5: Cold Shutdown
- Mode 6: Refueling

Procedure Writer's Guides promote standardization and the application of HFE usability principles to the operating procedures. Additional details are provided in Chapter 18, Subsection 18.3.7.

Normal, abnormal, unplanned, and emergency operating procedures are validated to be accurate and usable without any human error traps and verified to be consistent with the safe operating envelope.

Plant operations are performed in accordance with procedures, with use and adherence direction provided for the worker. Temporary procedures may be issued when existing permanent procedures are not applicable to the work being performed. Temporary procedures are periodically reviewed for applicability and cancelled when no longer required.

Operating procedures address:

- Normal operation
- Abnormal operation
- Emergency operation
- Refueling and outage planning
- Alarm response
- Maintenance, inspection, test, and surveillance
- Beyond design basis and severe accidents

The status of SSC is controlled with the following requirements:

- Status changes must be authorized
- Operational position of Safety Class devices is known and controlled

- Status of SSC under maintenance, inspection, or test is known
- Field equipment deficiencies are identified
- Placement and removal of tags on systems and components (e.g., caution tags, work protection tags, terminal point tags, and other similar tags) is controlled
- Plant status information is transferred during shift turnovers

13.4.3 Procedures and Guidelines for Operating the Plant During Accidents

13.4.3.1 Emergency Operating Procedures

This section describes the programmatic approach followed to develop the Emergency Operating Procedures (EOPs) in accordance with CNSC REGDOC-2.3.2, "Operating Performance – Accident Management" (Reference 13.6-20), and procedure development that supports the operator when responding to anticipated and unanticipated events. EOPs are developed in accordance with CNSC REGDOC-2.5.2, Section 4.2.4 (Reference 13.6-10).

Emergency procedures are available for non-routine and emergency conditions that require immediate action. Emergency conditions addressed include unexpected radiological and non-radiological hazards, excessive emission of radiological and non-radiological liquid or gaseous effluent, fires, and natural disasters. Emergency procedures are kept in prominent, easily accessible locations. Emergency procedures are exercised in practice drills to ensure that requirements are met.

EOPs implement the strategies and measures employed in the integrated accident management plan and ensure that escalation of an accident is avoided, the accident progression is terminated, and fission product releases are kept to a minimum. The EOPs contain a set of information, instructions, and actions designed to prevent the escalation of an accident, mitigate its consequences, and bring the reactor to a safe and stable state. The development of these procedures takes into consideration the information available to the operating staff and conditions where some of the information may be incomplete with significant uncertainties. Also taken into consideration are long time periods to initiate and complete required actions, human and organizational performance, and the possibility of prolonged times to restore power.

All EOPs are developed in accordance with a systematic procedure development plan that considers HFE principles in both the actions required by the procedure and the design of the procedure itself. Procedure development is a technical element of the BWRX-300 HFE Program, as described in Chapter 18, Subsection 18.3.7.

13.4.3.2 Guidelines for Accident Management

This section describes the programmatic approach followed to develop accident management procedures and guidelines, including EOPs, Emergency Mitigating Equipment (EME) guidelines and Severe Accident Management Guidelines (SAMGs) in accordance with CNSC REGDOC-2.3.2 (Reference 13.6-20).

Accident management includes multiple components such as equipment and instrumentation, procedures and guidelines, and organizational accountabilities, and it interfaces with many programs established for a reactor facility. An adequate accident management plan ensures the ability to respond to any credible accident in order to prevent the escalation of the accident, mitigate the consequences of the accident, and achieve a long-term stable state after the accident.

Integrated accident management planning consists of a cohesive set of plans and arrangements undertaken to ensure:

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1. Safety systems and the available SSC can be used to control the reactivity, cool the fuel, and contain the radioactive materials such that damage to the reactor vessel and harm to workers, public, and environment is prevented or mitigated
2. Personnel with responsibilities for accident management are adequately prepared to utilize the available resources, procedures, and guidelines to perform effective accident management actions and, when deemed necessary, to call for and interact with the emergency response teams

EOPs, EME guidelines, and SAMGs are developed and implemented to facilitate a licensee's capability to manage the AOOs, DBAs, and Beyond Design Basis Accidents, including Design Extension Conditions and severe accidents. These procedures are developed using a systematic approach in accordance with CNSC REGDOC-2.3.2 (Reference 13.6-20) and CNSC REGDOC-2.5.2, Sections 4.2.4, 7.9.3, 8.5, 8.10, and 9.3 (Reference 13.6-10).

The process of accident management planning will define and describe the following requirements:

- Specific goals of accident management
- Requirements of accident management
- Equipment and instrumentation
- Procedures and guidelines
- Organizational accountabilities

A timeline with milestones for the development, validation, and implementation of all operating procedures, EOPs, EME guidelines, and SAMGs for accident management is expected to be provided in the Licence to Operate application submission.

13.5 Nuclear Safety and Nuclear Security Interfaces

13.5.1 General Nuclear Safety and Security

The plans for physical protection of the facility are described in separate, confidential files. This section addresses how safety measures and nuclear security measures are designed and applied programmatically.

OPG, the operating organization, is responsible for managing the implementation of safety requirements and security requirements, with the primary objective of minimizing risk, through programs and processes established to ensure close cooperation between safety managers and security managers. The safety and security measures are designed and implemented through programs and processes in a complementary manner that do not compromise each other. Mechanisms are established within the programs to resolve any potential conflicts and to manage the safety-security interfaces.

13.5.2 Security

The following security measures, in accordance with the Nuclear Safety Regulations (SOR/2000-209, "Nuclear Security Regulations" (Reference 13.6-28)) and consistent with the relevant guidance provided in CNSC REGDOC-2.12.1 Volume I, "High Security Facilities, Volume I: Nuclear Response Force" (Reference 13.6-21); CNSC REGDOC-2.12.1 Volume II, "High Security Facilities, Volume II: Criteria for Nuclear Security Systems and Devices" (Reference 13.6-22); and CNSC REGDOC-2.12.3, "Security of Nuclear Substances: Sealed Sources and Category I, II and III Nuclear Material" (Reference 13.6-23), are established for the prevention, detection, and response to unauthorized acts, criminal or intentional, that could directly or indirectly produce harmful site consequences:

- Site physical security
- Personnel security
- Information protection
- Document security
- Cyber security

Prescribed and security-sensitive information is only provided to persons with a valid security clearance and "need to know."

Public access to the site-controlled area is restricted by fencing and signage and with OPG Nuclear Security Officer patrols.

OPG maintains a security clearance program consistent with CNSC REGDOC-2.12.2, "Security: Site Access Security Clearance" (Reference 13.6-24). Staff and contractors requiring unescorted access to the site require a security clearance commensurate with activities performed or access required.

Threat risk assessment is performed as part of the Nuclear Security Program, with results taken into consideration in plan development and facility response. A Memorandum of Understanding exists with the Durham Regional Police Services and is maintained to provide for an off-site response to OPG facilities. The agreement(s) ensure that necessary resources are available to address design basis security events. The Memorandum ensures resources are available to address design basis security events in support of existing armed Nuclear Security. OPG periodically conducts drills and exercises that include integrated response with the off-site response force. Lessons learned from drills and exercises are implemented within the Security

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Program. The Memorandum is subject to annual review and is revised to reflect existing security postures.

OPG's Cyber Security Program implements the OPG Cyber Security Policy. Information technology and industrial control systems are managed in a secure, vigilant, and resilient manner that minimizes cyber risks to information assets, renewable generation, and nuclear facilities. The Nuclear Cyber Security Program ensures secure operations of computer systems associated with the industrial control systems for OPG nuclear facilities. Cyber security is applied to plant systems, including those used to ensure safe operations and those which provide physical security of OPG nuclear facilities. The Nuclear Cyber Security Program complies with requirements of CSA N290.7, "Cyber Security for Nuclear Facilities" (Reference 13.6-25).

Nuclear Security Officers are selected, trained, and equipped in accordance with the applicable requirements of CNSC REGDOC-2.12.1 Volume I (Reference 13.6-21) and CNSC REGDOC-2.2.4, "Fitness for Duty, Volume III: Nuclear Security Officer Medical, Physical and Psychological Fitness" (Reference 13.6-26).

OPG has programs in place at existing operating nuclear facilities to facilitate compliance with applicable safeguard requirements and agreements. Measures related to site buildings and structures, operational parameters, and the flow and storage of nuclear material throughout the life cycle of the nuclear facility are described in the Environmental Impact Statement.

Details (prescribed information) of the Security Program are transmitted only by secure means consistent with OPG-STD-0030, "Protecting OPG's Information" (Reference 13.6-27) and SOR/2000-202, "General Nuclear Safety and Control Regulations," Sections 21-23, Prescribed Information (Reference 13.6-29).

Details providing security in design, which informs the Security Program, are provided in the Security Annex.

13.5.3 Physical Security

The Nuclear Security Program is implemented by Nuclear Security, N-PROG-RA-0011, "Nuclear Security" (Reference 13.6-30) using a graded approach. The bulk of the program is classified and addressed separately.

13.5.4 Cyber Security

The Cyber Security Program, OPG-PROG-0042 (Reference 13.6-31), procedures and controls ensure the following:

1. Ensure employees and contractors are in compliance with all applicable requirements of this Cyber Security Program
2. A culture of awareness is fostered to promote secure practices in the use of all digital technologies
3. Methods are established to monitor Information Technology and Operational Technology environments on an ongoing basis in order to detect and respond to threats that impact the confidentiality, integrity, and availability of OPG's assets
4. Strategies are in place to prepare for, respond to, and recover from cyber security incidents that impact OPG's reputation, energy production, and public and employee safety

13.5.5 Safeguards

The Safeguards Program, N-PROG-RA-0015 (Reference 13.6-32) is compliant with Nuclear Safety and Control Act, June 2000; its associated General Regulations, and CNSC REGDOC-2.13.1, "Safeguards and Nuclear Material Accountancy" (Reference 13.6-33); and includes the following:

- Communication protocol between the IAEA, CNSC, and Ontario Power Generation, Nuclear
- Obligations to meet applicable regulatory requirements and requirements of associated safeguards procedures
- Reporting to meet applicable regulatory requirements and requirements of safeguards agreements

See the Safeguards Annex: Safeguards and Nuclear Material Accountancy for additional details.

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13.6 References

- 13.6-1 IAEA Safety Standards Series No. NS-G-2.4, "The Operating Organization for Nuclear Power Plants," International Atomic Energy Agency.
- 13.6-2 CNSC Regulatory Document REGDOC-2.2.5, "Minimum Staff Complement," April 2019.
- 13.6-3 CNSC Regulatory Document REGDOC-2.2.2, "Personnel Training."
- 13.6-4 CNSC Regulatory Document REDOC-2.5.1, "General Design Considerations: Human Factors."
- 13.6-5 N-CHAR-AS-0002, "Nuclear Management System," Ontario Power Generation.
- 13.6-6 CSA N286, "Management System Requirements for Nuclear Facilities," CSA Group.
- 13.6-7 OPG Document, N-PROG-AS-0001, Nuclear Management System Administration.
- 13.6-8 CNSC Regulatory Document REGDOC-2.1.1, "Management System."
- 13.6-9 CNSC Regulatory Document REGDOC-2.1.2, "Management System: Safety Culture."
- 13.6-10 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 13.6-11 CNSC Regulatory Document REGDOC-2.6.1, "Reliability Programs for Nuclear Power Plants."
- 13.6-12 CNSC Regulatory Document REGDOC-2.6.2, "Maintenance Programs for Nuclear Power Plants", August 2017.
- 13.6-13 CNSC Regulatory Document REGDOC-2.6.3, "Aging Management."
- 13.6-14 IAEA Specific Safety Guide No. SSG-61, "Format and Content of the Safety Analysis Report for Nuclear Power Plants," International Atomic Energy Agency.
- 13.6-15 CSA N286.10, "Configuration Management for High Energy Facilities," CSA Group.
- 13.6-16 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
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HITACHI

GE Hitachi Nuclear Energy

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September 30, 2022

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**Ontario Power Generation Inc.
Darlington New Nuclear Project
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 15
Safety Analysis**

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
0	All	Initial Release

ACRONYM LIST

Acronym	Explanation
AAZ	Automatic Action Zone
AC	Alternating Current
ACRW	All Control Rod Withdrawal at Power
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
ATLM	Automatic Thermal Limit Monitor
BDBA	Beyond Design Basis Accident
BE	Best Estimate
BIS	Boron Injection System
BL-AOO	Baseline Abnormal Operational Occurrence
BL-DBA	Baseline Design Basis Accident
BL-DSA	Baseline Deterministic Safety Analysis
BOP	Balance of Plant
BWR	Boiling Water Reactor
CAFTA	Computer Aided Fault Tree Analysis
CANDU	CANada Deuterium Uranium
CB	Control Building
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CDF	Core Damage Frequency
CET	Containment Event Tree
CIV	Containment Isolation Valve
CN-DBA	Conservative Design Basis Accident
CN-DSA	Conservative Deterministic Safety Analysis
CNSC	Canadian Nuclear Safety Commission
CPR	Critical Power Ratio
CRD	Control Rod Drive

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Acronym	Explanation
CRDA	Control Rod Drop Accident
CRDM	Control Rod Drive Motor
CSA	CSA Group
CSAU	Code Scaling, Applicability, and Uncertainty
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
DBA	Design Basis Accident
DCF	Dose Conversion Factor
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DF	Decontamination Factor
DL	Defense Line
D-in-D	Defence-in-Depth
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
DSA	Deterministic Safety Analysis
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EHE	External Hazard Evaluation
EOC	End of Cycle
EOP	Emergency Operating Procedure
EOR	End of Rated Cycle
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
EX-DBA	Extended Design Basis Accident
EX-DEC	Extended Design Extension Condition
EX-DSA	Extended Deterministic Safety Analysis
FFHE	Functional Failure Hazard Evaluation
FHA	Fuel Handling Accident
FLEX	Diverse and Flexible Coping Strategy
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analysis

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Acronym	Explanation
FSF	Fundamental Safety Function
FV	Fussell-Vesely
FW	Feedwater
FWCIV	Feedwater Containment Isolation Valve
FWFI	Feedwater Flow Increase
FWPT	Feedwater Pump Trip
GEH	GE Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
HEP	Human Error Probabilities
HFE	Human Factors Engineering
HOHE	Human Operation Hazard Evaluation
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
ICRW	Inadvertent Control Rod Withdrawal at Power – Single Rod
ICS	Isolation Condenser System
IE	Initiating Event
IHE	Internal Hazard Evaluation
IR	Inventory Reduction
LFWH	Loss of Feedwater Heating
LOCA	Loss-of-Coolant Accident
LOPP	Loss-of-Preferred Power
LPSD	Low Power Shutdown
LPZ	Low Population Zone
LRF	Large Release Frequency
LR-TT	Load Reduction – Turbine Trip
MAAP	Modular Accident Analysis Program
MCC	Motor Control Center
MCCI	Molten Core Concrete Interaction
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room

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Acronym	Explanation
MCS	Minimal Cutset
MOC	Middle of Cycle
MRBM	Multi-Channel Rod Block Monitor
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSRIV	Main Steam Reactor Isolation Valve
NBS	Nuclear Boiler System
NBR	Nuclear Boiler Rated
NPP	Nuclear Power Plant
OBE	Operating-Basis Earthquake
ODE	Ordinary Differential Equation
OLC	Operational Limits and Conditions
OLMCPR	Operating Limit Minimum Critical Power Ratio
OOC	Out Of Core Criticality
OPEX	Operating Experience
OPG	Ontario Power Generation
PA	Postulated Accident
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PCW	Plant Cooling Water System
PI-AOO	Pressure Increase- Abnormal Operational Occurrence
PIE	Postulated Initiating Event
PIRT	Phenomena Identification and Ranking Table
POS	Plant Operating State
PPS	Plant Pneumatics System
PRA	Probability Risk Assessment
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSIG	Pounds per Square Inch Gauge
PWR	Pressurized Water Reactor
QHO	Quantitative Health Objectives

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Acronym	Explanation
RAW	Risk Achievement Worth
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RI	Reactivity Increase
RI-AOO	Reactivity Increase- Abnormal Operational Occurrence
RIV	Reactor Isolation Valve
RPC	Reactor Pressure Control
RPF	Radial Peaking Factor
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SA	Severe Accident
SAM	Severe Accident Management
SAMG	Severe Accident Management Guideline
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator
SMR	Small Modular Reactor
SPSA	Seismic Probabilistic Safety Assessment
SSC	Structures, Systems, and Components
SS-DBA	Safe-Shutdown Design Basis Analysis
STP	Simulated Thermal Power
TB	Turbine Building
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TRACG	Transient Reactor Analysis Code General Electric
TSV	Turbine Stop Valve
URD	Utility Requirements Document
USNRC	U.S. Nuclear Regulatory Commission
Δ CPR/ICPR	Delta Critical Power Ratio Over Initial Critical Power Ratio

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15.0 SAFETY ANALYSIS

This chapter includes the following safety analyses complying with the requirements of CNSC REGDOC-1.1.2, Section 4.4.1, Draft Version 2 (Reference 15.1-1):

- Hazards Analysis
- Deterministic Safety Analysis (DSA)
- Probabilistic Safety Assessment (PSA)

The safety analysis primary objective is demonstrating that the Fundamental Safety Functions (FSFs) (see Chapter 3, Subsection 3.1.5) are effective in:

- Controlling reactivity
- Removing heat from the fuel (reactor or fuel pool)
- Confining radioactive materials
 - Shielding against radiation
 - Controlling operational discharges
 - Limiting accidental releases

The BWRX-300 design basis is achieved through an iterative safety process. The design is implemented to meet defined safety objectives that are confirmed via the safety analyses. Results of the safety analyses provide feedback to the design. If indicated by the results, the design may be modified until safety objectives are met.

This chapter documents the safety objective and the safety analyses approach, description and results performed for the BWRX-300 design. The DSA and the PSA are conducted in compliance with CNSC REGDOC-2.4.1 (Reference 15.1-2), and CNSC REGDOC-2.4.2 (Reference 15.1-3), respectively. The content of Chapter 15 is structured as follows:

Section 15.1 General Considerations of the BWRX-300 Safety Analyses:

- Safety Analysis Objectives, Scope, and Approach
- Analysis of Hazards – Scope and Approach
- Analysis of Design Basis Conditions - Scope and Approach
- Analysis of Design Extension Conditions Without Core Damage Scope and Approach
- Analysis of Beyond Design Basis Accidents (BDBAs) with Core Damage (Severe Accident)- Scope and Approach

Section 15.2 Identification, Categorization, and Grouping of Postulated Initiating Events and Accident Scenarios discusses:

- Development of categorizing events according to their frequencies
- Grouping of events according to their types
- Selection of the bounding events for each group type

Section 15.3 Safety Objectives and Acceptance Criteria discusses:

- Safety objectives based upon the safety strategy

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- Associated acceptance criteria for Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and safety goals

Section 15.4 Human Actions discusses:

- General Considerations
- Human Actions considered in the DSA
- Human Actions considered in the PSA

Section 15.5 Deterministic Safety Analysis discusses:

- Safety margins in the safety analysis (Subsection 15.5.1)
- Analysis of normal operations (Subsection 15.5.2)
- Analysis of AOOs (Subsection 15.5.3)
- Analysis of DBAs (Subsection 15.5.4)
- Analysis of Design Extension Conditions (DECs) without core damage (Subsection 15.5.5)
- Analysis of DECs with core damage (Subsection 15.5.6)
- Analysis of PIEs and accident scenarios associated with the fuel pool (Subsection 15.5.7)
- Analysis of fuel handling events (Subsection 15.5.8)
- Analysis of radioactive release from a system or component outside containment Loss-of-Coolant-Accidents (LOCA) outside containment (Subsection 15.5.9)
- Analysis of internal and external hazards (Subsection 15.5.10)

Section 15.6 Probabilistic Safety Assessment discusses:

- General approach to the PSA (Subsection 15.6.1)
- Results of the Level 1 PSA (Subsection 15.6.2)
- Results of the Level 2 PSA (Subsection 15.6.3)

Section 15.7 Summary Results of the Safety Analyses discusses:

- Analysis results of normal operation (Subsection 15.7.1)
- Analysis results of AOOs and DBAs (Subsection 15.7.2)
- Analysis results of DECs without core damage (Subsection 15.7.3)
- Analysis results of DECs with core damage (Subsection 15.7.4)
- Analysis results of fuel pool events (Subsection 15.7.5)
- Analysis results of fuel handling events (Subsection 15.7.6)
- Analysis results of radioactive releases from LOCAs outside containment (Subsection 15.7.7)
- Analysis results of internal and external hazards (Subsection 15.7.8)
- Analysis results of the PSA (Subsection 15.7.9)

Appendix 15A discusses reference source term for conditions that are practically eliminated.

Appendix 15B discusses complementary design features for mitigating DEC's.

Chapter 15 documents the analysis and results from the BWRX-300 plant safety analyses.

15.1 General Considerations of the BWRX-300 Safety Analyses

The BWRX-300 safety strategy framework integrates the Defense Lines (DLs) provided by the implementation of the Defence-in-Depth (D-in-D) concept (defined in CNSC REGDOC-2.5.2, Section 4.3.1 (Reference 15.1-4) with the safety analyses. The D-in-D concept uses insights gained from operating experience and deterministic and risk-informed and performance-based analyses.

Chapter 3 defines the safety strategy framework implementation process shown on Figure 3.1-2. Design features, functions, and practices are organized into DLs that protect the integrity of the physical barriers against the radioactive releases. Safety analyses are performed that demonstrate the effectiveness of the Structures, Systems, and Components (SSCs) necessary to perform the functions assigned in various DLs that are credited to mitigate the Postulated Initiating Events (PIEs). The SSCs necessary to achieve the DL safety functions are described in Chapters 4 through 8. The PIEs analyzed are identified and selected through the systematic process of fault evaluation described in Section 15.2.

The safety analyses prove that the plant design meets the underlying safety objectives and acceptance criteria for event mitigation and confirms that the applicable regulatory safety objectives are met.

15.1.1 Safety Analysis Objectives, Scope, and Approach

The general Nuclear Safety Objectives in Chapter 3, Subsection 3.1.1 are demonstrated by the results from the safety analyses. Chapter 3, Figure 3.1-2 presents a graphical representation of the safety analysis performed according to the BWRX-300 safety strategy framework that includes:

- Hazards Analysis
- Deterministic Safety Analysis (DSA)
- Probabilistic Safety Assessment (PSA)

The Hazard Analysis consists of four types of hazard evaluations described in Subsection 15.1.2. The main objectives of the hazard evaluations are the identification of potential PIEs and confirmation that the plant design effectively responds to credible internal and external hazards.

The safety analysis objectives include:

- Demonstrating the design meets the acceptance criteria established following a graded approach for each plant state. The graded approach application may lead to acceptance criteria more restrictive for events with higher occurrence probability.
- Deriving and confirming Operational Limits and Conditions (OLCs) for normal operation
- Establishing and validating accident management procedures and guidelines

The safety analysis scope includes the plant states described in Chapter 3, Subsection 3.1.3 and illustrated in Figure 3.1-1. The plant states are consistent with CNSC REGDOC-2.5.2, Section 7.3 and includes:

- Normal operation
- AOOs

- DBAs
- BDBAs

The subset of BDBA or DECAs may occur without core damage or with core damage.

The hazards analysis consists of four types of hazard evaluations described in Subsection 15.1.3. The hazards evaluation main objective is the identifying potential PIEs and demonstrating the plant design effectively responds to credible internal and external hazards.

The BWRX-300 DSA uses a layered analysis approach that includes three types of DSA evaluations:

- Baseline – deterministic safety analysis (BL-DSA)
- Conservative – DSA (CN-DSA)
- Extended – deterministic safety analysis (EX-DSA)

This approach addresses initiating and mitigating DL function failures in a more systematic and structured manner than past approaches.

These DSA acceptance criteria are discussed in Section 15.3. The DSA results are compared against the applicable plant state acceptance criteria and dose limits specified in Subsection 15.3.1.

The PSA is performed to complement the DSA. PSA estimates the overall risk presented by the facility that is compared to the regulatory safety goals specified in Subsection 15.3.2. The PSA is presented in Section 15.6.

15.1.2 Analysis of Hazards

An initial step in performing the safety analysis is a systematic hazards evaluation. The BWRX-300 Safety Strategy process described in Chapter 3, Subsection 3.1.6.4 identifies four types of hazard evaluations for the complete range of operational modes (full power, low power, load following, shutdown and refueling) that produces a comprehensive set of PIEs:

- Functional Failure Hazard Evaluation (FFHE)
- External Hazard Evaluation (EHE) – addressed in Chapters 3, 6, Sections 3.3, 6.5, respectively
- Internal Hazard Evaluation (IHE) Chapter 3, Section 3.4, Chapter 6, Section 6.5, Chapter 9A, Section 9A.6
- Human Operation Hazard Evaluation (HOHE)

The hazard evaluations include any consequential failure that occurs because of the PIE. They also address all sources of radioactivity (e.g., spent fuel, fuel being handled) in addition to the reactor core itself.

Each hazard evaluation identifies any potential challenges to an FSF.

15.1.2.1 Functional Failure Hazard Evaluation

The FFHE identifies failures of plant systems or equipment with potential to cause a challenge to an FSF. These hazards are identified in Failure Modes and Effect Analyses (FMEAs) performed on the plant systems.

The FFHE is limited to random single failures and to CCFs. The system FMEAs are reviewed to identify failures that cause challenges to FSFs. A consolidated list of failures from all system FMEAs is generated and organized.

The functional failure hazard potential PIE sources are organized by quantitative frequency, using the frequency ranges defined in the Safety Strategy.

15.1.2.2 External Hazard Evaluation

The EHE includes natural and human-induced hazards that originate from a source that is not under control of the nuclear power plant licence holder. The EHE addresses individual hazard sources and combinations of sources:

- Natural external hazards include earthquakes, droughts, floods, high winds, tornadoes, tsunami, and extreme meteorological conditions
- Human-induced external hazards include toxic gas releases, aircraft crashes, or ship collisions

External events are site-specific and are specified in the site evaluation provided in Chapter 2.

Once the external hazards are identified, the BWRX-300 structures are designed to withstand these external hazards, and the resulting protection is described in Chapter 9B. Human-induced external hazards such as toxic gas and aircraft impacts are provided in Chapter 6, Section 6.5 for control room habitability.

The sources of external hazard or combinations are organized by quantitative frequency as potential PIEs evaluated in the fault evaluation (see Chapter 3, Section 3.3). Malevolent acts are addressed in the Security Annex.

15.1.2.3 Internal Hazard Evaluation

The IHE identifies conditions originating within the boundaries of the site and with potential to lead to an unplanned plant transient. The internal hazard condition does not directly challenge an FSF, but the effects of the hazard may cause equipment failures. These equipment failures are then evaluated in the deterministic and probabilistic safety analyses.

Internal hazards include:

- Fires (discussed in Chapter 9A, Section 9A.6)
- Explosions, missiles from rotating or pressurized equipment (discussed in Chapter 3, Section 3.4)
- Collapse of structures/falling objects (discussed in Chapter 9B)
- Pipe whip, jet effects, and flooding (discussed in Chapter 3, Section 3.6)

The IHE addresses both individual hazard sources and combinations of sources.

The sources of internal hazard or combinations are organized by quantitative frequency as potential PIEs and evaluated in the fault evaluation (see Chapter 3, Section 3.4).

15.1.2.4 Human Operation Hazard Evaluation

The HOHE identifies erroneous decisions or human action(s) that lead to an unplanned plant transient. Human operations hazards, typically involve unplanned changes to plant equipment status by equipment operators or maintenance personnel.

Many human operations hazards produce the same effects as corresponding equipment failures and the effects of these are included in the FFHE described in Subsection 15.1.3.1. The HOHE

is limited to a single erroneous act that may lead to multiple system responses. The HOHE focuses on identifying unique hazards such as an operator initiating a group command on multiple actuators that is beyond what is considered in a single failure analysis of a particular system.

The sources of HOHE or combinations are organized by quantitative frequency as potential PIEs and evaluated in the fault evaluation (see Section 15.5).

15.1.3 Analysis of Design Basis Conditions

The BWRX-300 design basis conditions are normal operations, AOOs and DBAs described below:

1. **Normal Operation** is operation within specified OLCs (see Chapter 16) and includes the full range of plant operating modes (Chapter 1, Section 1.8.). The objective of the normal operation safety analysis is to demonstrate that DL1 measures are effective in preventing abnormal operations and failures, thus meeting radiological requirements.
2. **AOOs** are deviations from normal operation that are expected to occur at least once during the operating lifetime of the reactor facility. The objective of the AOO safety analysis is to demonstrate that DL2 functions are effective for most AOO PIEs in meeting the applicable acceptance criteria.
3. **DBAs** conditions are identified as deviations from normal operations that are less frequent and more severe than AOOs. An objective of DBA safety analysis is to demonstrate that DL3 functions are effective in mitigating events and meeting the applicable acceptance criteria.

Acceptance criteria applicable to the DSA for each plant state is discussed in Subsection 15.3.1. The response to AOOs and DBAs is achieved by SSCs specifically designed to mitigate these events and are assigned DL2 and DL3 functions (Chapter 3, Subsection 3.1.6.2).

The DSA results for design basis conditions in Section 15.7 demonstrate that the requirements of CNSC REGDOC-2.4.1 are met.

15.1.4 Analysis of Design Extension Conditions Without Core Damage

DECs are a subset of BDBAs. DECs are postulated accident conditions that are less frequent than DBAs. DECs may occur with or without core damage.

DSA is performed for DECs without core damage demonstrating that releases of radioactive material are kept within acceptable limits and support the PSA determination of no core damage.

DEC analysis include:

- Multiple failures defined as complex sequences identified in the Level 1 PSA or as a PIE with a Common Cause Failure (CCF)
- AOO and DBAs with postulated failures of DL2 and DL3 functions analyzed in EX-DSA. For these events, the DBA acceptance criteria are used as screening criteria to the evaluation of core damage
- Low frequency events
- Non-reactor fault sequences (fuel pool accidents) are analyzed in Level 1 PSA

The results of the DSA for DECs without core damage are discussed in Section 15.7. The analysis of DECs with core damage are addressed in the Level 2 PSA described in Subsection 15.6.4.

15.1.5 Analysis of Beyond Design Basis Accidents with Core Damage (Severe Accident)

These are referred to as Severe Accidents (SA) and involve a catastrophic failure, core damage, and fission product release. A SA is generally considered to begin with the onset of core damage. To the extent that core damage is not practically eliminated, representative DECAs with core damage are postulated to provide inputs for the containment design and safety features ensuring containment functionality. This set of accidents is considered in the design of corresponding safety features for DECAs and represents a set of bounding cases. Accident scenarios considered for practical elimination are described in Appendix 15A.

Severe accident sequences are selected that identify representative core damage scenarios and corresponding plant damage states that are used as the basis for performing the Severe Accident Analysis (SAA). The scope of SA scenario selection corresponds to sequences involving significant core damage that could lead to a containment breach and radioactive release analyzed in the Level 2 PSA in Section 15.6. The selected SA scenarios are included in a fault evaluation.

The SAA goal is to provide input to accident management for terminating the progression of core damage, maintaining containment integrity as long as possible, and minimizing on-site and offsite radioactive material releases. Halting core damage progress prevents Reactor Pressure Vessel (RPV) failure.

The response to SAs considers the use of safety and non-safety, permanent and temporary systems and equipment that are beyond their originally intended functions.

Consistent with SSR 2/1, Paragraph 2.11 (Reference 15.3-3), practical elimination is applied to events or sequences of events leading to or involving core damage (a severe accident) where confinement of radioactive materials cannot be reasonably achieved. Event sequences that are either physically impossible or extremely unlikely to occur are considered for practical elimination.

The practical elimination demonstration is performed with accident conditions and phenomena knowledge and is substantiated by relevant evidence (see Appendix 15A Reference Source Term for Conditions That Are Practically Eliminated).

15.1.6 References

- 15.1-1 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 15.1-2 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 15.1-3 CNSC Regulatory Document REGDOC-2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants," May 2014.
- 15.1-4 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."

15.2 Identification, Categorization and Grouping of Postulated Initiating Events and Accident Scenarios

A fundamental element of the safety analyses is the identification and selection of PIEs that is achieved through systematic process of fault evaluation. The fault evaluation objective includes:

- Identification, categorization and grouping of PIEs
- Identification of the plant functions expected to be credited in the safety analysis and their assignment to a functional DL (DL2, DL3 and DL4)

The fault evaluation scope is the list of potential PIEs generated by the hazards evaluation (see Subsection 15.1.3) and includes:

- Complete range of operating modes
- All radioactivity sources (reactor core and outside core)
- Single failure PIEs, CCF PIEs, and fault sequences developed based on the success or failure of mitigating functions. Single and CCF PIEs include equipment failures and human errors. CCF are analyzed as software platform or mechanical failures.

The output of the fault evaluation is documented in a fault list. The fault list establishes traceability between the plant design and the safety analysis. The fault evaluations start in parallel with or prior to DSA and PSA activities. DSA and PSA mature with the design and the fault list is updated accordingly.

The fault evaluation process used for PIE identification and selection complies with CNSC REGDOC-2.4.1, Section 4.2, "Events to be analyzed" (Reference 15.3-1).

The fault evaluation includes the following activities as shown in Chapter 3, Figure 3.1-2 BWRX-300 Safety Strategy Implementation Process.

- Deterministic PIE Selection
- Complex Sequence Selection
- Severe Accident Scenario Selection

Deterministic PIE Selection

The deterministic PIE selection is the systematic process in organizing and selecting events for deterministic safety analyses. The activities performed during the deterministic PIE selection are described in Subsection 15.2.1. Selected PIEs and fault sequences are allocated to three DSA types in a fault list:

- PIE List for Baseline Deterministic Safety Analysis (BL-DSA)
- PIE/Fault Sequence List for Conservative Deterministic Safety Analysis (CN-DSA)
- PIE/Fault Sequence List for Extended Deterministic Safety Analysis (EX-DSA)

Complex Sequence Selection

Complex sequences are fault sequences involving failures of multiple mitigating features, which have not been included in the deterministic PIE selection but are identified in the Level 1 PSA as having the potential to lead to core damage with a frequency of occurrence or consequences judged to require analysis and DL mitigation function. These complex sequences are added to the fault list and analyzed in the EX-DSA (Subsection 15.1.5).

Severe Accident Scenario Selection

The scope of severe accident sequence selection corresponds to those sequences involving significant core damage, which could lead to a breach of containment and radioactive release in the Level 2 PSA that is described in Section 15.6. To the extent that core damage is not practically eliminated, representative severe accident sequences (DECs with core damage) are postulated and analyzed in the SAA. The primary objective of the severe accident sequence selection is to identify representative core damage scenarios and define corresponding plant damage states that are used as the basis for performing the SAA. The selected SA scenarios are documented in the fault list and are analyzed in the SAA (Subsection 15.1.6).

15.2.1 Basis for Categorization of Postulated Initiating Events, Accident Scenarios and Fault Evaluation

The hazard evaluations described in Subsection 15.1.3 results in the list of potential PIEs. These potential PIEs are evaluated, categorized, and grouped during the fault evaluation in the deterministic PIE selection. The hazard evaluations address a complete range of plant modes of operation, all sources of radioactivity and any consequential failure that occurs because of the PIE. During design development, the hazard evaluation is validated, and PIE selection is updated accordingly.

The activities included in the deterministic PIE selection and their bases are presented below:

1. Fault Sequences development – a Fault Sequence is developed starting with a PIE and considers the success or failure of the required mitigating functions. The DL of each credited mitigating function is established for each Fault Sequence.
2. Fault sequences are grouped into fault groups based on similar impact on a certain plant parameter: for example, events that lead to pressure increase in the reactor such as inadvertent closure of the Turbine Stop Valves and/or Turbine Control Valves or inadvertent closure of the Main Steam Reactor Isolation Valve(s) (MSRIVs) are grouped in the pressure increase fault group. Subsection 15.2.3 includes the output of the grouping activity.
3. Fault sequences are categorized within each fault group as AOO, DBA or DEC based on their frequency of occurrence. Subsection 15.2.2 describes this categorization.
4. Plant conditions are defined corresponding to each PIE supporting the scenario analysis.
5. Any exceptions are applied or justified to the standard PIE selection.

Fault sequences are allocated to three types of DSAs:

- BL-AOO and Baseline Design Basis Accident (BL-DBA) – Baseline DSA
- CN-AOO and CN-DBA –Conservative DSA
- EX-DEC –Extended DSA

A bounding set of PIEs and fault sequences that result in the most significant challenge to the FSFs are selected for evaluation in the DSA. DSA layers and events categories are combined so that limiting baseline events are AOO (BL-AOO), the limiting CN events are DBAs (CN-DBA) and limiting EX events are DECs (EX-DEC). This notation is used to identify the layer and event category.

Subsection 15.2.4 describes the bounding event selection for each fault group that is captured in a fault list. In addition, the fault list includes the complex sequences and the severe accident scenarios specified in Subsection 15.2.4.

A description of the three deterministic safety analyses aligned with the functional DLs (DL2, DL3 and DL4a) is included below.

15.2.1.1 Baseline Deterministic Safety Analysis

The primary objective of the BL-DSA is demonstrating the effectiveness of the DL2 functions. The scope of BL-DSA includes single failure PIEs categorized as bounding BL-AOOs and BL-DBAs. The BL-DSA models the expected response of the plant (no failure is postulated) to demonstrate that the event meets applicable acceptance criteria. The analysis end point is the controlled state condition. The mitigating DL functions credited in BL-DSA are DL2 functions. If a DL2 function fails or is not effective, then the corresponding DL3 function is credited.

15.2.1.2 Conservative Deterministic Safety Analysis

CN-DSA primary objective is demonstrating the effectiveness of DL3 functions. The CN-DSA scope includes events categorized as bounding CN-AOOs and CN-DBAs:

- PIEs due to single failure
- PIEs due to spurious CCF in DL2 or DL4a
- Baseline PIEs with postulated passive CCF of DL2 functions that were credited in BL-DSA

The CN-DSA is performed using conservative initial conditions with established acceptance criteria and applying a graded approach in quantifying the uncertainties (see Subsection 15.5.1.1). Single failure criterion is applied to DL3 SC1 SSCs. CN-DSA credits only DL3 mitigation functions. The end point of the analysis is a controlled state condition.

15.2.1.3 Extended Deterministic Safety Analyses

The EX-DSA primary objective is assessing the effectiveness of DL4 functions. The EX-DSA scope includes events categorized as DEC:

- PIEs due to spurious CCF of DL3 functions
- DBA fault sequences with postulated passive CCF of DL3 mitigating functions
- Complex sequences identified by the Level 1 PSA

An extended sequence for AOOs and DBAs is required in the following conditions:

1. If a DL3 function is credited to mitigate a single failure PIE in the BL-DSA, then the DEC fault sequence assumes a passive CCF in DL3 functions (no additional mitigation single failure is assumed).
2. If the hydraulic scram action is credited in an BL-AOO scenario, then the hydraulic scram action is assumed to have a mechanical CCF of the hydraulic scram where only the Control Rod Drive Motor (CRDM) run-in functions insert control rods. No additional failures are assumed.

15.2.2 Categorization of Events According to Their Frequencies

One fundamental element of the deterministic PIE selection (Section 15.1) and fault sequence selection is the assignment of fault sequences to categories based on their frequency of occurrence that complies with CNSC REGDOC-2.4.1, Sections 4.2.2.5 and 4.2.3, and CNSC REGDOC-2.5.2, Section 5.4.3:

- Anticipated Operational Occurrence (frequency greater than 1E-02 per reactor-year)
- Design Basis Accident (frequency between 1E-02 and 1E-05 per reactor-year)

- Design Extension Condition (frequency less than 1E-05 per reactor-year)

Qualitative frequencies are adopted as an interim measure and are used in the early design stages to progress the performance of deterministic analyses prior to availability of more mature PSA information. Quantitative frequencies based on Level 1 PSA results are adopted as the final, governing measure of the event sequence category.

A fault sequence consists of a combination of a PIE and can include an assumed failure of a mitigating function(s). The event sequence category is based on the sequence frequency not only the PIE frequency. The event category assigned to an event sequence may be different than the event category assigned to the PIE that initiated the sequence because the event sequence may include additional failures that make the sequence less likely to occur.

In addition to the event categorization frequency, the categorized events are allocated the following DSA types:

- Baseline Anticipated Operational Occurrence (BL-AOO)
- Conservative Anticipated Operational Occurrence (CN-AOO)
- Baseline Design Basis Accident (BL-DBA)
- Conservative Design Basis Accident (CN-DBA)
- Extended Design Extension Condition (EX-DEC)

15.2.3 Grouping of Events According to Type

One of the steps in fault evaluation is grouping the events according to their type. The fault evaluation includes external events, internal events, human operational errors, functional failures evaluated in hazard analysis, Level 1 PSA complex sequences and Level 2 PSA severe accident sequences.

PIEs (faults) are grouped according to the resultant change in plant parameter:

- Temperature decrease events – decrease in core coolant temperature
- Pressure increase events – increase in reactor pressure
- Reactivity increase events – reactivity and power distribution anomalies
- Inventory increase events – increase in reactor coolant inventory
- Inventory reduction events – decrease in reactor coolant inventory
- Non-reactor fault events – these events are non-core related such as fuel handling accident
- Radiological faults having dose consequences

Once the fault groups are identified, then the anticipated core physics response associated with each group is then selected. Once each group is identified, then the bounding fault sequence from that group is selected.

Table 15.2-1 provides the fault groups with an explanation of how anticipated core physics response (reactor response) was considered in development of the groups. Within these groups, fault sequences with similar responses are compared and used to select the bounding events.

15.2.3.1 Core Reactivity Effects of PIEs and Accident Scenarios

The anticipatory effects of core reactivity response provided in Table 15.2-1 are discussed below with focus on AOOs, DBAs, and DECAs without core damage because these are event categories where the core remains intact, and the reactivity feedback mechanisms are well understood.

Void Reactivity

Void reactivity is an important reactivity feedback mechanism in BWR transient and accident analyses. The void reactivity feedback is always negative and is typically stronger (more negative void coefficient) at the end of an operating cycle. It is also stronger for reload cores versus an initial reactor core that includes only fresh fuel. This reactivity feedback mechanism is the dominant feedback for some PIE groups. Table 15.2-1 focuses on events initiated from conditions of normal power operation. In the DSA, where it is important, void reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response analyzed using Transient Reactor Analysis Code General Electric (TRACG), the primary DSA computer code (see Subsection 15.5.1.2).

Moderator Temperature Reactivity

The moderator temperature coefficient of reactivity is defined as the change in reactivity produced by a unit change in moderator temperature. The value of this coefficient is important during the startup of a BWR. During power operation, the coefficient is not important, because the moderator is boiling and remains at the saturation temperature corresponding to the operating pressure. In addition, moderator density changes caused by boiling are much larger than changes from moderator temperature changes and therefore, mask any effects. The BWRX-300 core is designed and evaluated to conform to regulatory requirements as discussed in Chapter 4, Sections 4.2 - 4.4. Moderator temperature feedback is accounted for in the DSA analysis of startup conditions when coolant temperature is key to the event response and there is no significant voiding in the core. Once core boiling/voiding begins, void reactivity feedback becomes dominant.

Control Reactivity

Neutron absorbing control rods are the primary means to control reactivity in transient and accident analyses. During events that result in relatively fast positive reactivity feedback, control rods are inserted rapidly using stored hydraulic energy. This is referred to as a "scram". During events that result in relatively slow positive reactivity feedback and do not require a reactor "scram", the Fine Motion Control Rod Drive (FMCRDs) that are operated using electric motors can be used for slower control rod insertion, (see Chapter 4, Subsection 4.5.1). Reactivity feedback mechanism is key for some PIEs groups (see Table 15.2-1) in which rods are withdrawn in error. Control reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response in TRACG.

Doppler Reactivity

Doppler reactivity is a less important reactivity feedback mechanism in BWR transient and accident analyses than void or control reactivity. Doppler reactivity is negative with an increase in fuel temperature and becomes more important as the fuel temperature continues to increase. In DSA where reactivity feedback is important, doppler reactivity is modeled using 3-D kinetics coupled with the thermal hydraulic response in TRACG.

Boron Reactivity

For the BWRX-300, boron reactivity insertion (see Appendix 15B) is only needed for long-term shutdown in very low probability events where the hydraulic and electric motors fail to insert a sufficient number of control rods.

Xenon Reactivity

Xenon reactivity feedback is not typically accounted for during events in DSA because the rate of change of reactivity is slow. The effects of xenon are accounted for in analyses of shutdown margin.

15.2.4 Postulated Initiating Events and Accident Scenarios

PIEs and event frequency are first determined qualitatively based on system conceptual design, previous similar designs, and operating experience. PIEs are evaluated in the fault evaluation (see Section 15.2) where they are further screened for inclusion in the fault list.

The bounding event selection is performed for events that are initiated at full power conditions (Mode 1 operating condition) because they are expected to result in the most significant challenge to the fission product barriers.

Bounding events are selected in each fault group, for each event category (e.g., AOO, DBA, DEC without core damage) and for the applicable DSA layer (e.g., baseline, conservative and extended). The resulting events selected are listed in Table 15.2-2 and analyzed in Section 15.5. DEC events with core damage are part of the PSA and SAA.

The bounding event selection is performed for two event categories:

- Transient or non-LOCA described in Sections 15.2.4.1 through 15.2.4.6
- LOCA scenarios described in Section 15.2.4.7

Bounding Event Selection for Transient Events

Bounding events are selected for the transient (non-LOCA) DSA that pose the most challenges in meeting the derived acceptance criteria.

The selected bounding events are summarized in Table 15.2-2. Table 15.2-2 also points to a complete description of the bounding event in the DSA described in Section 15.5.

15.2.4.1 Decrease in Core Coolant Temperature Bounding Event

Events that result in core coolant temperature decreases are grouped as Temperature Decrease (TD) faults. A reduction in coolant temperature (at the core inlet) has the potential to challenge the fuel cladding barrier due to increasing reactivity as a result of reducing the core coolant void fraction. The Reactor Coolant Pressure Boundary (RCPB) is not challenged because there is not a significant increase in steam flow and normal pressure control is not affected. Because this is a reactivity driven event, it is only of concern when the reactor is not shutdown.

The largest source of coolant supply is from the FW pumps. FW flow enters the RPV downcomer through the FW piping. Extraction steam from the turbine is directed to heat FW in multiple stages of FW heaters. Failures, such as loss of extraction steam, can result in a reduction of the temperature entering the RPV.

The Isolation Condenser System (ICS), Control Rod Drive System (CRD), Reactor Water Cleanup System (CUW) and the Shutdown Cooling System (SDC) also have inflows to or outflows from the RPV that have the potential to reduce the coolant temperature. Inadvertent ICS initiation PIEs are included in the increase in reactor coolant inventory fault group. These systems can only reduce the coolant temperature a small fraction relative to Feedwater Heating (FWH)-related PIEs. Therefore, only FW heater related PIEs are considered potentially bounding.

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There is no fault group for increase in core coolant temperature for the following reasons:

1. The Feedwater (FW) temperature is near the highest temperature that is feasible during normal operation.
2. Any increase in FW temperature would increase the core void fraction and reduce core power due to the decrease in void reactivity.
3. An increase in FW temperature does not result in an increase in core temperature because the core is boiling and remains at saturated conditions.

Other than an increase in the FW temperature, the core coolant temperature may increase due to the following conditions:

- Core power increase resulting in a small increase in pressure covered by the reactivity and power distribution anomalies fault group
- Reactor pressure increase that effects saturation temperature covered by increase in reactor pressure fault groups

Therefore, any possible increase in FW temperature is small and results in a small decrease in power and poses no threat to fission product barriers resulting in no need for an increase in core coolant temperature fault group.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

Bounding AOO Event Selection

The event effect on the fuel Critical Power Ratio (CPR) and core power increase is used in selecting the bounding AOO events for this group.

The AOO that results in the largest reduction in FW temperature is the Loss of Feedwater Heating (LFWH) event. This BL-AOO event described in Subsection 15.5.3.1.1 is the bounding AOO event for this category.

Bounding DBA Event Selection

The bounding CN-DBA event described in Subsection 15.5.4.1.1 resulting in the largest postulated reduction in FW temperature is the CCF leading to loss of all FW heaters. The event effect on fuel Peak Cladding Temperature (PCT) is used in selecting the bounding DBA in this group.

Bounding DEC Without Core Damage Event Selection

There are no DEC events in this fault group because the DLs are established in the baseline and conservative DSA. No complex sequences in the fault group are identified.

15.2.4.2 Increase in Reactor Pressure Bounding Event Selection

Events that result in RCPB pressure increase are referred to as Pressure Increase (PI) faults. During full power operation, steam generated in the reactor exits the RPV through the Main Steam Lines (MSLs). There are normally open valves in the each MSL: Two Main Steam Reactor Isolation Valves (MSRIV) and one main steam Containment Isolation Valve (CIV). Outside containment, there is a main steam header upstream of the Turbine Stop Valves (TSVs). Downstream of the main steam header the TSVs are in series with the Turbine Control Valves (TCVs).

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The MSRIVs and CIVs close to isolate the RPV or containment. The TSVs and TCVs close to protect the turbine. Turbine Bypass Valves (TBVs) located in the main steam equalizing line in the Turbine Building (TB) allow steam to bypass the closed TSV or TCV.

Pressure increase faults are generally caused by closure of valves in the steam flow path. Closure of a single MSRIV terminates steam flow in one of the two MSLs. Closure of any two of these valves in separate MSLs causes a complete isolation of the MSLs upstream of the TBVs. Closure of both TSVs and/or TCVs, terminates steam flow to the turbine. A turbine trip or load rejection signal initiates closure of the TSVs and TCVs, and fast opening of the TBVs. When RPV pressure increases, TBVs are opened by the normally operating Reactor Pressure Control (RPC) which allows up to ~25% of the rated steam flow to the main condenser.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

There is no fault group for decrease in reactor pressure because a pressure decrease reduces reactivity. The decrease in reactor pressure is caused by either:

- Reactor power decrease that is covered by the reactivity at power distribution anomalies fault group
- Reactor coolant inventory is lost that is covered by the decrease in reactor coolant inventory fault groups.

Therefore, there is no need for a decrease in reactor pressure fault group.

Bounding AOO Event Selection

The event effect on the fuel CPR, core power increase, and the RCPB pressure are used in selecting the bounding AOO events for this group.

There are several AOO events in the PI group in the fault list. Most of the AOOs are PIEs that result in closure of the TSVs, TCVs or both. These events present similar challenge to the cladding and RCPB, and all of them are selected as potentially limiting (bounding) events:

- Turbine trip, load rejection
- Loss-of-preferred power
- Loss of condenser vacuum

Another AOO in this group is the BL-AOO closure of one Main Steam Reactor Isolation Valve (MSRIVC). The MSRIVs do not close as fast as the TCVs or TSVs; however, the MSRIVs are upstream of the TBVs, resulting in this event selected as a potentially limiting.

Bounding DBA Event Selection

The event effect on PCT and the RCPB pressure are parameters used in selecting the bounding DBA events for this group.

There are several DBA events that result in increased pressure in the PI group. There are two main types assumed in the PI group:

1. AOO events where the TSV or TCV closes with a CCF of the DL2 mitigation equipment
2. CCF results in closure of the MSRIV and FWRIV

Because the TCV fast closure is faster than the MSRIV closure, the closure of the TCVs with failure of the DL2 mitigation functions is a more significant pressure increase event. However,

because several of these events result in similar challenges to acceptance criteria, several CN-DBA events (described in Subsection 15.5.4) are chosen as potentially bounding.

Bounding DEC Without Core Damage Event Selection

The event effect on PCT and the RCPB pressure are parameters used in selecting the bounding DEC events for this group.

The main types of PI group DEC events are:

1. Bounding AOO events with failure of the hydraulic scram due to postulated CCF of the hydraulic components
2. Complex sequence AOO event with half of the control rods fail to insert either by HCUs or with the CRD motors run-in
3. CCF in DL3 functions initiating events

The events assume a CCF of the hydraulic scram that significantly challenges the fuel cladding and RCPB because the CRD motor run-in that backs up the hydraulic scram results in slower negative reactivity insertion. Several of these events are selected as potentially bounding. The event result from CCF in DL3 functions are not concerning because sufficient non-DL3 functions initiate the hydraulic scram.

In the TSV and TCV closure events, pressure control is achieved (at least momentarily) via the TBVs to the main condenser. In the loss of condenser vacuum and LOPP AOO events with failure of the hydraulic scram, the main condenser is available for a limited amount of time. These events result in more severe pressurization and are selected as potentially bounding relative to the turbine trip and load rejection events. The 1MSRIVC AOO with failure of the hydraulic scram is also selected.

A complex sequence is also considered in this fault group based on the PSA evaluation. This event is selected as a potentially limiting event.

15.2.4.3 Reactivity and Power Distribution Anomalies Bounding Event Selection

Events that result in reactivity and power distribution anomalies are grouped as Reactivity Increase (RI) faults. These events include failures in reactivity control that challenge the fuel cladding or RCPB integrity. The BWRX-300 controls reactivity with control rod movement. Normal control is accomplished using FMCRDs, which are individually connected to the control rods. HCUs are used to quickly shutdown the reactor by inserting control rods, ensuring reactivity control during off normal operation. Increases in reactivity can result from control rod withdrawal, an error in fuel loading, or control rod drop.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

Bounding AOO Event Selection

The fault list includes only one RI-AOO. This event is Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW). However, this event is not evaluated because a protection function of Automatic Thermal Limits Monitor (ATLM) initiates a control rod block. The event conditions remain within normal operation conditions; therefore, no analysis is needed. The design description of the ATLM is found in Chapter 7, Subsection 7.3.3.2.

Bounding DBA Event Selection

The event effect on PCT and cladding integrity are the parameters used in selecting the bounding events for this group.

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There are several DBA PIEs in the RI group to postulate in identifying the bounding event including the withdrawal of single or multiple control rods. The DBA PIEs are all mitigated by active DL2 functions ATLM and Multi-channel Rod Block Monitor (MRBM) that result in minimal change to core power in the baseline sequence. The ATLM blocks rod motion before departure from normal operation and MRBM blocks rod motion before significant fuel effects occur. These mitigation functions are reliable resulting in a failure sequence in the DEC frequency range. These events with the failure of ATLM and MRBM are selected as bounding DEC without core damage events because of the low frequency of the sequence. The following fault sequences are evaluated.

1. CCF - All Control Rod Insertion at Power - (CCF-ACRI): For some core conditions, this event may result in a momentary increase in reactivity in the top part of the core because higher density water moves up the core faster than the rod motion. As the rods insert, high density coolant increases the reactivity above the control rods. This is a momentary effect, does not challenge fuel cladding criteria or any other fission product barrier, and is not selected as a bounding event.
2. Fuel Loading Error (FLE): This event postulates that two fuel assemblies are swapped during refueling or a single assembly is inserted in a rotated position (180 degrees from normal). This can result in small local increase in reactivity. This event is selected as a potentially limiting DBA event.
3. Control Rod Drop Accident: (CRDA): This event postulates a fault that allows separation of the control rod from the drive mechanism (design precludes a separation such as this due to a separation detection device). The control rod becomes stuck and remains in its position when the drive mechanism is attempted to be withdrawn. Before the stuck rod is detected and is inserted, the control rod becomes unstuck and falls. This event is not a limiting RI condition and not explicitly analyzed because the design includes separation detection devices that limits the drop to a small distance, resulting in no significant reactivity changes. This event is listed in Mode 2 (startup conditions) because it is limiting at those conditions. However, the event is possible at normal full power operations.

Bounding DEC Without Core Damage Event Selection

The event effect on PCT and cladding integrity are the parameters used in selecting the bounding DEC events for this group. There are two control rod withdrawal error DEC events that are potentially limiting:

- Single rod withdrawal
- Inadvertent withdrawal of all rod groups

These DEC without core damage are compared to applicable acceptance criteria demonstrating that features to prevent core damage are adequate and no further complementary design features are required.

15.2.4.4 Increase in Reactor Coolant Inventory Bounding Event Selection

Events resulting in an increase in reactor coolant inventory are included as Inventory Increase (II) faults. These events may occur from a FW Inventory Increase. Inadvertent ICS initiation scenarios do not fit well into any fault group and are included in this increase in inventory event category. Inadvertent ICS initiation results in much less Inventory Increase than a FW increase. However, the dynamics of ICS flow into the chimney is different than an increase in FW flow. These scenarios are selected for analysis.

Other systems such as CRD, CUW, SDC or Boron Injection System (BIS) may result in inventory increases. However, these systems have low flow rates compared to FW and are not considered as bounding.

The events selected result in the most significant challenge to fission product barrier in this fault group and the selected events are summarized in Table 15.2-2.

Bounding AOO Event Selection

There is one BL-AOO event, Inadvertent Isolation Condenser Initiation – One Train, and it is selected for evaluation.

Bounding DBA Event Selection

The event effect on PCT and coolant inventory are the parameters used in selecting the bounding events in this group.

Two CN-DBA events are selected:

- Inadvertent injection of all ICS trains (bounds one train) (Inadvertent Isolation Condenser Initiation – All Trains (CCF-IIICI))
- Feedwater Flow Increase - All Pumps (CCF-FWFI). This event bounds the CN-DBA event for increase in flow of one FW pump.

Bounding DEC Without Core Damage Event Selection

There are no DEC events identified in the II fault group because for each single failure AOO or DBA PIE, there are two DLs established that mitigate the event. No AOO events credit the hydraulic scram. For all CCF PIEs, a DL is established.

15.2.4.5 Decrease in Reactor Coolant Inventory Bounding Event Selection

Events that result in a decrease in reactor coolant inventory are included as Inventory Reduction (IR) events. These events may occur from failures that result in:

- Reduction or loss in FW flow
- Opening of TBVs
- Pipe breaks (LOCA events) (discussed in Subsection 15.2.4.7)
- Misalignment of systems connected to the RPV
- Loss-of-preferred power (this event is included in the PI group)

The IR bounding events selection only includes non-LOCA events. In this fault group, maintaining inventory above the top active fuel ensures fuel cladding integrity. For non-LOCA events, there is no significant challenge to the RCPB or containment.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

Bounding AOO Event Selection

The event effect on coolant inventory is the parameter used in selecting the bounding AOO events in this group. There are two BL-AOO events in this group:

- FW pump trip - one pump (FWPT)
- Inadvertent opening of one TBV

The FWPT event is selected as potentially limiting because the TBV opening represents less inventory loss than the FW pump trip and FW makes up inventory loss during the TBV opening.

Bounding DBA Event Selection

The event effect on the coolant inventory are the parameters used in selecting the bounding DBA events for this group.

There are several PIEs events in the IR group:

- Loss of all FW flow
- Opening of all TCVs and TBVs
- RPV Pressure Control Open

The selected bounding event is Loss of FW flow (LOFW) CN-DBA event. FW may also be lost by a FW isolation valve closure DL4a CCF (BL-DBA) but is not selected because this is less severe than LOFW. This is also less frequent and is categorized as a DEC when combined with a CCF of DL2 mitigation. The RPV Pressure Control Open (CCF-RPCO) is selected as potentially limiting for events where inventory is lost via the main steam line.

LOCA events bound these events because they result in a more significant challenge to the RPV inventory (fuel cooling and long-term cooling) and challenge containment temperature and pressure. The non-LOCA DBA events are not bounding for this IR fault group.

Bounding DEC Without Core Damage Event Selection

The event effect on PCT and coolant inventory are the parameters used in selecting the bounding DEC events for this group.

The FW Isolation EX-DEC event is selected as a potentially limiting event; however, it is not expected to be a significantly different response than the CN-DBA event. LOCA events bound this event, and the non-LOCA events are not bounding for this IR fault group.

15.2.4.6 Bounding Event Selection for LOCA Scenarios

The initiating events involving pipe breaks, scram, and trip initiation are identified in the fault list for Baseline (BL), Conservative (CN) and Design Extension Conditions (DECs). The evaluation scenarios are then selected to bound groups of pipe breaks. The consequences of postulated LOCAs are analyzed for fuel cladding and containment integrity and shown to meet the design basis acceptance criteria in Table 15.3-2. CN-DSA LOCA analyses demonstrate that the reactor level does not decrease below the Top of Active Fuel (TAF) or the fuel cladding does not exceed the normal operating temperatures. Meeting these acceptance criteria assures the fuel cooling acceptance criteria in Table 15.3-2 are met. The CN-DSA LOCA analyses also demonstrate that the containment acceptance criteria in Table 15.3-2 are met.

The bounding scenarios for pipe breaks fall into two categories:

- Large breaks inside or outside containment
- Small breaks inside or outside containment

The breaks in each category may be in steam pipes or liquid pipes. CN-DSA sequences are mitigated by DL3 functions alone, assuming a CCF of DL2 functions. EX-DSA fault sequences (DECs) are mitigated by DL2 and DL4a functions (see Table 15.5-50 DL2 and DL4a functions credited).

Pipe breaks attached to the RPV may be as large as the complete rupture of the largest steam or feedwater pipes, or as small as leaks in smaller pipes attached to the RPV. Although the

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frequency of the largest breaks is lower than $1\text{E-}05/\text{year}$, all sizes of pipe breaks resulting from various PIEs are conservatively analyzed as DBAs. Further, the largest breaks are instantaneous double-ended guillotine ruptures of the large pipes. This assumption is made to bound the thermal hydraulic response of the fuel, RPV and containment for all isolatable pipe breaks.

The bounding LOCA events provided in Table 15.2-2 and evaluated in Subsections 15.5.4.6 (LOCA inside containment) and 15.5.9.2 (LOCA outside containment) provide the assumptions and DL components used in mitigating the event with the corresponding signals, times, and other design parameters.

A break between the RPV and the Reactor Isolation Valves (RIV) is not a credible postulated accident. Breaks inside containment are postulated to occur at any arbitrary location between the outer RIV, or the flow limiter for MS pipes, and the containment boundary.

Large Breaks

Pipes that are larger than 19 mm (0.75 in) inside diameter have two isolation valves attached directly to the RPV. The largest postulated pipe breaks are in the main steam, feedwater, and ICS lines. These pipes have RIVs, which close in less than 5 seconds once they start closing. Another 5-second delay is assumed before a RIV starts closing to account for delays in break detection and signal development. A break between the RPV and RIVs is not credible as a postulated accident in the current fault list.

MS pipes are also equipped with a flow limiter to prevent very large break flow prior to MSRIV closure. The flow limiters are placed close to the RPV. Breaks on the MS pipes are postulated to occur downstream of the flow limiter.

The analytical limit for the high containment pressure setpoint is reached within 1 second for large breaks (steam and liquid). Peak containment pressure occurs at approximately the time RIVs are fully closed and the break is isolated. The largest steam pipe breaks and the largest liquid pipe breaks are the most limiting for containment response because they have the highest mass and energy release until the break is isolated. For medium size breaks that are isolated on high containment pressure, there may be a delay in reaching the containment high pressure setpoint for isolation. However, containment pressure is increasing at a slower rate than it would for a larger break while the RIVs are closing. Because the RIV closure time is the same for all breaks, the containment peak pressure is smaller for a medium size break than it is for a large break.

Since the largest breaks are more limiting for the mass and energy release, fuel integrity and containment integrity, a break spectrum analysis is not required for isolatable breaks. This is also the case for breaks outside containment since they are isolated by the leak detection system in a similar manner.

The leakage detection system is designed to detect breaks in large pipes connected to the RPV. If the break is not detected, a conservative assumption is that the break remains unisolated since no operator action is credited.

For large break LOCAs, RIVs close rapidly and prevent significant loss of RPV inventory. The core remains covered throughout. The remainder of the event after the RIVs are closed is an isolation event during which the ICS has ample capacity to remove the decay heat and depressurize the RPV and maintain fuel cooling for at least 72 hours. Fuel integrity is not a concern for large breaks. In the long-term Passive Containment Cooling System (PCCS) reduces containment pressure.

Small Breaks

The smaller pipes connected to the RPV have an inside diameter ≤ 19 mm, do not have automatic RIVs, and are considered unisolated breaks. Small unisolated breaks may also occur in larger pipes. The leakage detection system detects breaks in large pipes connected to the RPV but may not be capable of detecting breaks that are smaller than the area of a circle with a 19 mm diameter. If the break is not detected, it is assumed to remain unisolated for 72 hours for CN sequences because no operator action is credited for this duration.

Small breaks conservatively credit only two of the three isolation condensers even though a break of less than 19 mm equivalent diameter on an isolation condenser does not cause degradation in the isolation condenser heat removal rate. There is sufficient steam in the RPV to feed the condensation in the isolation condenser. Not having sufficient steam in the RPV to feed the isolation condenser can only occur if the RPV is depressurized so far that almost all of the steam is escaping the break. This would be the case if the RPV pressure is even lower than that calculated for an instrument pipe break. However, in this case the break flow is less than the break flow calculated for an instrument pipe break, making the isolation condenser small break less limiting than the instrument pipe break. It can be concluded that whether the break flow rates are the same for a small steam pipe break regardless of the break location, or the RPV pressure is too low to feed the isolation condenser, the breaks on isolation condenser steam pipes are no more limiting than a break on an instrument steam pipe break.

For unisolated breaks, the RPV inventory is depleted faster as the break area becomes larger. The largest of the unisolated breaks is most limiting with respect to RPV inventory. Peak containment pressure is also higher as the break size is increased for an unisolated break.

Large and Small Pipe Break Summary

The largest break sizes are the most limiting for isolatable (i.e., in the large break category) and un-isolatable (i.e., in the small break category) breaks.

In selecting the scenarios to evaluate for the pipe breaks, each of the sequences in the fault list is assessed with respect to the largest mass and energy release to containment and RPV inventory. Additional conservative assumptions that are described in Subsections 15.5.4.6 and 15.5.9 were made in constructing the bounding scenarios to reduce the number of analysis cases. Therefore, the bounding scenarios analyzed for pipe breaks in large and small break categories are scenarios bounding all scenarios in that category. However, the liquid and steam pipe breaks are analyzed separately.

The following common features are used in selecting the bounding sequences for pipe breaks:

1. A pipe break does not cause loss of feedwater or loss of normal containment cooling unless a direct or indirect effect of the pipe break causes the pump(s) to trip. Feedwater and normal containment cooling are lost concurrent with the break for the sequences involving a pipe break concurrent with Loss-of-Preferred Power (LOPP). This observation is used to determine whether the breaks are more limiting with or without LOPP.
2. All scrams credited in the LOCA analyses are direct scrams. A scram signal is initiated when the setpoint of the first scram function is reached for the scram functions that are available in the credited DL. Scram functions and setpoints in each DL and the trip parameters are provided in Subsections 15.5.4.6 and 15.5.9 and meet the guidance of CNSC REGDOC-2.4.1, Section 4.4.4.4.
3. In the CN-DBA sequences, a CCF of DL2 functions concurrent with the pipe break are assumed and credit only DL3 functions.

4. In the EX-DEC sequences, CCF of DL3 functions are assumed in the analysis and credit only DL2 and DL4a mitigation functions.

The guidelines for considering CCFs in the design and safety analyses are discussed in Chapter 3, Subsection 3.1.7.8.

The system responses, including the direct and indirect effects of large and small pipe breaks, are described in Subsections 15.5.4.6 and 15.5.9.

15.2.4.6.1 Large Steam Pipe Breaks

Large steam pipe breaks are postulated to occur in the following systems:

- Main Steam (MS) pipes
- ICS steam supply pipes

For breaks inside containment, scram is initiated on high containment pressure (DL3-07). For breaks outside containment, scram is initiated on detection of a steam pipe break (DL3-09).

MSRIV closes on high containment pressure for breaks inside containment (DL3-22), and on MS pipe break detection for breaks outside containment (DL3-20). ICS RIVs close for an ICS train when an ICS break inside or outside containment is detected in the respective ICS train (DL3-27, 28 or 29).

Because the isolation valves close rapidly, the effect on isolation condenser availability is not significant for large breaks mass and energy releases. After break isolation, one ICS is sufficient to remove decay heat and depressurize the RPV. As a bounding assumption, only one ICS train is credited for MS and ICS pipe breaks. This assumption is made so that an ICS steam supply break is bounded by an analyzed MS pipe break. One ICS train initiates on high containment pressure (DL3-15) for breaks inside containment. One ICS train initiates on pipe break indication detection (DL3-16) in MS or ICS pipes outside containment.

For a MS pipe break, the total break flow is the sum of break flows from both ends of the break. To bound all break locations, the break location is assumed as close to the RPV as possible, right outside the second or outboard RIV. Because two MS lines are connected through a header, the intact steam pipe also supplies the break location from the turbine side of the break. Break flow from the turbine side of the break is contributed by the flow from the RPV into the intact loop and the initial inventory in the piping. To maximize flow from the turbine side of the break, the isolation valves outside the containment are assumed to remain open, and TSV/TCV close rapidly. Because closure of CIVs outside containment are not credited, the calculated mass and energy release is applicable to breaks inside and outside containment.

If the break occurs when the plant is at very low power or hot shutdown, break flow from an MS pipe break may be higher due to carryover. At low power, there is more saturated water in the RPV to flash. This may increase the two-phase downcomer level much higher than that in the rated initial conditions case, contributing to the break flow due to increased liquid content although the break flow enthalpy is lower. In calculating the radiological consequences for breaks outside containment, the break mass flow rate for hot shutdown initial conditions may be more limiting. Both the rated initial conditions and the hot shutdown initial conditions are included in the bounding scenarios for MS pipe breaks.

The above scenario assumes LOPP concurrent with the break is bounding for the scenario when preferred power is available. If the preferred power is available, FW will continue to be injected to the RPV, which may increase mass release to the containment due to carryover. However, if preferred power is available, TSV/TCV also remains open, discharging much of the steam in the

intact loop to the turbine rather than to the break from the turbine side of the break location. As a result, the break scenario with LOPP is the more bounding scenario.

Bounding CN Scenario for Large Steam Pipe Breaks

The bounding scenario analyzed MS pipe breaks inside and outside containment concurrent with LOPP:

The case is analyzed for rated initial power and hot shutdown conditions. The limiting break scenario is analyzed in Subsection 15.5.4.6.1.

15.2.4.6.2 Large Liquid Pipe Breaks

Large liquid breaks may occur in:

- FW pipe
- ICS condensate return pipe
- CUW pipe

The largest pipe from this list is the FW pipe and is analyzed in Subsection 15.5.4.6.2.

A large break in the ICS condensate return pipe is listed as a liquid break because the condensate return pipe is filled with water initially. However, this water is highly subcooled. The ICS condensate return valves are normally closed and the steam supply pipe is in communication with the RPV during normal operation. If a pipe break occurs at the condensate return pipe, the highly subcooled water in the condensate pipe is purged. The energy release to containment for a break inside containment or to the ICS pool for a break outside containment from the purged highly subcooled water is insignificant. After the liquid inventory in the pipe is depleted, the break flow becomes steam flow supplied from the RPV through the steam supply pipe until the RIVs close for the broken ICS unit.

The ICS steam supply pipe is fitted with a 70 mm inside diameter orifice at the steam distribution pipes. Therefore, flow from a break in the condensate pipe is only steam flow through a 70 mm orifice after the subcooled water is purged. This is less limiting than a break in the ICS steam supply pipe included in the large steam pipe break cases. This pipe break requires no further analysis because it is included in the large steam pipe break category for CN sequences.

CUW pipe breaks may occur inside or outside the containment. CUW pipe is a smaller-bore pipe and CUW breaks are isolated the same as the FW pipe breaks. Therefore, the CUW pipe breaks inside containment are bounded by the FW pipe breaks. CUW pipe breaks are routed through the same compartments that house the FW pipes in the reactor building. Therefore, CUW pipe breaks outside containment are also bounded by the FW pipe breaks for reactor building pressure and temperatures.

The limiting liquid pipe breaks inside and outside the containment are the FW pipe breaks. The break isolation, scram, and ICS initiating are discussed below.

Bounding CN Scenario for Large Liquid Pipe Breaks

For a pipe break inside containment, scram is initiated on high containment pressure (DL3-07). For breaks outside containment, scram is initiated on pipe break indication in FW or ICS pipes (DL3-09). Scram does not occur for the CUW pipe breaks outside containment.

In accordance with the fault list, preferred power is available for CUW breaks outside containment and FW injection continues.

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LOPP concurrent with a break is assumed for the FW pipe breaks inside the containment. The TCV is conservatively assumed to close rapidly with LOPP. In the LOPP cases, there is flow from the pump side of the break before the CIVs close outside containment at 10 seconds due to pump coast down, but more importantly, due to flashing of the water in the FW piping. If the FW pumps continue running in spite of the break when preferred power is available, there is some increase in flow from the pump side of the break. However, normal containment cooling also continues to run when preferred power is available, compensating for the effect on containment pressure from an increase in the flow from the pump side of the break.

For breaks outside containment in the RB pressure and temperature calculations, the cases with and without FW running are included.

The radiological consequences for breaks outside containment uses the LOPP case that is more limiting. FW pump flow has no consequence because flow coming from the FW pump is decontaminated water and can only retard the break flow coming from the RPV through the intact loop. Only the water leaving the RPV is important in the radiological analyses that is higher if the pump is not running.

For breaks inside containment, RIVs close on high containment pressure (DL3-22). CIVs outside containment also close on high containment pressure (DL3-22).

For FW pipe breaks outside containment, FW RIVs close on break detection. MS RIVs also close on break detection (DL3-21).

For CUW pipe breaks outside containment, only CUW RIVs close on break detection (DL3-26).

For breaks inside containment, ICS is initiated on high containment pressure (DL3-15).

For FW and ICS pipe breaks outside containment, ICS is initiated on-line break detection (DL3-16). For CUW breaks outside containment, ICS initiation is not needed since preferred power is available.

Bounding Scenario Summary for Large Liquid Pipe Breaks

The bounding liquid pipe break for containment response is analyzed for a FW pipe break inside containment:

- Double-ended guillotine FW pipe break inside containment concurrent with LOPP
- TSV/TCV are conservatively assumed to close rapidly retaining more energy
- FW pump trips and coast down
- Scram initiation within 1 second after the break
- ICs initiate on high containment pressure
- MS and FW RIVs start closing in 5 seconds and are fully closed in 10 seconds
- FW CIVs outside containment start closing in 5 seconds and are fully closed in 10 seconds
- FW conservatively assumed to trip at time zero and coasts down with a 3 second time constant

The bounding liquid pipe break scenario outside containment is similar to breaks inside containment with the following differences:

- CIVs are conservatively assumed not to close
- FW pump trip may or may not occur. Area pressure and temperature calculations consider both cases. The radiological analyses conservatively assume a FW pump trip

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The above scenario does not rely on any scram or isolation function that is a result of the LOPP. Therefore, the same scenario bounds both preferred power available and LOPP cases.

15.2.4.6.3 Small Breaks

Small, unisolated steam or liquid pipe breaks may occur in instrument lines. Small breaks in the large pipes may also remain unisolated if they are below the leak detection system threshold, i.e., less than 19 mm inside diameter. The lowest location for a liquid pipe break is four (4) meters above the TAF.

Small breaks are analyzed using conservative assumptions demonstrating that fuel and containment integrity are maintained for at least 72 hours using only passive systems after which injection is recovered and the event is terminated. The LOCA acceptance criterion for demonstrating fuel integrity is to show that fuel cladding does not heat-up beyond normal operating temperature. This satisfies the fuel integrity acceptance criteria in Table 15.3-2 with large margin.

The fault list includes unisolated small breaks inside and outside the containment, with and without concurrent LOPP.

The bounding CN sequences in the fault list for a small break concurrent with LOPP are evaluated with respect to the fuel cladding and containment. For a break inside containment concurrent with LOPP, normal containment cooling system is also assumed to be lost. The energy discharged from the break to containment is removed by the PCCS (discussed in Chapter 6, Section 6.3.3) and through the containment dome. PCCS does not require actuation; it is always in service.

When the preferred power is available, FW continues to run. Because the FW pump can make up for the break flow, fuel integrity is not a concern. If preferred power is available, normal containment cooling also continues to run. If the normal containment cooling and PCCS cannot keep up with the break flow and containment pressure increases, the reactor scrams, isolation condensers initiate and RPV depressurizes reducing the break flow. Containment cooling continues to operate maintaining containment pressure at a lower value than PCCS alone maintains in the LOPP case.

The bounding small liquid and steam pipe break scenarios are the same, the only difference is the discharge from the small liquid pipe break is initially from the liquid water space. It becomes steam flow after level falls below the RPV nozzle elevation of the broken pipe.

Following an unisolated break in an instrument pipe concurrent with LOPP, turbine pressure decreases rapidly resulting in a decrease in the steam pipe pressure and an increase in steam flow. Although a consequential closure of TCVs may occur on LOPP, this is not credited in the analysis. TCV is assumed to remain in its initial position. Scram is initiated when the steam pipe pressure decreases to low steam pipe pressure setpoint (DL3-02) with a 1.7 second delay. Power is assumed to remain at the initial value for an additional 2 seconds to account for the time elapsed until prompt fission power is diminished after the control rods start inserting. MSRVs also start closing on low steam pipe pressure (DL3-17) over 5 seconds and are fully closed in 10 seconds.

FW pumps trip concurrent with LOPP and coast down with a 3 second time constant.

All available ICS train condensate return valves start opening with a delay of 1 second when the level falls to L2 level setpoint (DL3-14). Only two isolation condenser trains are assumed available. ICS condensate return valves are fully open 10 seconds after they start opening. There are no further actuations assumed for the remainder of the event.

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Mass and energy releases from small pipe breaks do not credit containment back pressure for breaks inside containment. In addition, reactor scram and isolations initiated by high containment pressure are also not credited. Therefore, the above scenario applies to breaks outside containment as well as breaks inside containment. Because a break outside containment occurs from a longer pipe, break mass and energy releases calculated for a break inside containment bounds a small break outside containment.

Bounding Small Pipe Break

The bounding scenario is summarized as follows:

- Small steam or liquid pipe break concurrent with LOPP
- FW pump trips and feedwater flow coasts down with a time constant of 3 seconds
- Pressure controller failure, TSV/TCV position remain open at their initial position
- Reactor scrams when steam pipe pressure decreases to low steam pipe pressure setpoint with a 1.7 second delay
- MSRV closure when steam pipe pressure decreases to low steam pipe pressure setpoint with a 5-second delay
- ICS initiation when level is less than L2

Bounding Scenarios for DEC Pipe Breaks

DECs assume DL3 CCF in addition to the pipe break. Only DL2 and DL4a functions are credited in DECs. There is either a DL2 or DL4a function for all credited DL3 functions in Table 15.5-49, except the isolation condenser pipe breaks. Because the heat removal by the ICs is a higher-class safety function than isolation by a DL4a function, no DL4a associated function exists. An unisolated isolation condenser pipe break is similar to an unisolated MS pipe break and is evaluated separately subject to different acceptance criteria.

DL2 and DL4a functions performing the same function for the credited DL3 functions are listed in Table 15.5-50. The setpoints and timing of these functions are the same as or close to the DL3 functions. Therefore, except for the isolation condenser pipe breaks, the analyzed design basis LOCA analyses bound the DEC pipe breaks for the isolatable large pipe breaks and un-isolatable small breaks.

15.2.5 References

- 15.2-1 IAEA Safety Standards Series No. SSR-1, "Site Evaluation for Nuclear Installations Safety Requirements," International Atomic Energy Agency.

Table 15.2-1: Fault or Event Groups and Explanation of Core Reactivity Response Basis

Name (ID)	Description	Discussion of Reactivity Effects
Temperature Decrease (TD)	Decrease in Core Coolant Temperature	Void reactivity is key. The decrease in temperature results in an increase in the core inlet subcooling. More core thermal power goes into heating up the water and less into void production. The core void fraction decreases and causes the power to increase. This is a relatively slow increase in core power due to the thermal inertia of the coolant.
Pressure Increase (PI)	Increase in Reactor Pressure	Void reactivity is key. Pressurization results in decrease of core voids and increase in core power. For rapid pressurization, control rod scram is needed to mitigate. Core exposure effects on void reactivity and control rod position are important aspects that are modeled and included in the analysis.
Reactivity Increase (RI)	Reactivity and Power Distribution Anomalies	Control rod reactivity is key. Errors or failure in control rod movement are expected event initiators as these events add reactivity and change the local and core wide power. Fuel loading errors are also included.
Inventory Increase (II)	Increase in Reactor Coolant Inventory	Void reactivity is key. An increase in coolant inventory results in a reactor water level increase. The increase in reactor water level normally has the following effects: <ul style="list-style-type: none"> - Core flow increase (small reactivity effect at rated power conditions) - Core inlet subcooling increase (because the additional inventory is expected to be from lower temperature coolant, larger reactivity effect than the core flow increase) These effects tend to increase void reactivity.
Inventory Reduction (IR)	Decrease in Reactor Coolant Inventory	Void reactivity is key. The decrease in reactor water level has the following potential results: <ul style="list-style-type: none"> - Core flow decrease (small reactivity effect at rated power conditions) - Core pressure decrease (if break in coolant pressure boundary cannot be compensated for by pressure control) - Core inlet subcooling decrease (comes with pressure decrease) These effects tend to insert negative reactivity, resulting in core power decrease, due to void reactivity. It is typical in loss-of-coolant accident analysis to ignore these effects as the protection systems typically act quickly to insert control rods and shutdown the core before the negative void reactivity feedback has significant effect.
Non-Reactor Faults	Event specific. These events are not core related.	Fuel Handling Accident

Table 15.2-2: Bounding Events Transient (Non-LOCA) and LOCA

DSA Layer / Event Category	Event and Fault Sequence ID	Corresponding DSA Sections 15.5.3 Through 15.5.5 Event Summary Results
Decrease in Core Coolant Temperature Bounding Event Summary		
BL-AOO	Loss of Feedwater Heating (LFWH) TD-LFWH_BL-AOO	15.5.3.1.1
CN-DBA	Common Cause Failure – Loss of Feedwater Heater CCF- LFWH, Passive CCF DL2 Technology Platform; TD-CCF- LFWH_CCF-DL2_CN-DBA	15.5.4.1.1
EX-DEC	None	N/A
Increase in Reactor Pressure Bounding Event Summary		
BL-AOO	Generator Load Rejection or Turbine Trip (LR-TT); PI-LR-TT_BL-AOO	15.5.3.2.1
	Closure of One MSRV 1MSRIVC; PI-1MSRIVC_BL-AOO	15.5.3.2.2
	Loss of Condenser Vacuum (LOCV) PI-LOCV_BL-AOO	15.5.3.2.3
	Loss-of-Preferred Power (LOPP) PI-LOPP_BL-AOO	15.5.3.2.4
CN-DBA	Load Rejection or Turbine Trip LR-TT, Passive CCF DL2 Technology Platform (CCF-DL2); PI-LR-TT_CCF- DL2_CN-DBA	15.5.4.2.1
	Loss-of-Preferred Power LOPP, Passive CCF DL2 Technology Platform (CCF-DL2); PI-LOPP_CCF- DL2_CN-DBA	15.5.4.2.2
	RPV Pressure Control Downscale CCF - RPV Pressure Control Downscale (CCF-RPCD), Passive CCF DL2 Technology Platform (CCF-DL2); PI-CCF- RPCD_CCF-DL2_CN-DBA	15.5.4.2.3
	Closure of All MSRVs and FW Isolation CCF - Closure of All MSRVs and FW isolation valves (CCF-DL4a-MSRIVC- FWIV); PI-CCF-DL4a-MSRIVC- FWIV_CN-DBA	15.5.4.2.4

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DSA Layer / Event Category	Event and Fault Sequence ID	Corresponding DSA Sections 15.5.3 Through 15.5.5 Event Summary Results
EX-DEC	Closure of One MSRIV PI-1MSRIVC_CCF-Hydraulic-Scram_EX-DEC Complex Sequence of Generator Load Rejection or Turbine Trip Complex Sequence of LR-TT + CCF-Mechanical-Scram; CSS-LR-TT_CCF-Mechanical-Scram_EX-DEC Loss of Condenser Vacuum LOCV, CCF-Hydraulic-Scram; PI-LOCV_CCF-Hydraulic-Scram_EX-DEC Loss-of-Preferred Power LOPP, CCF-Hydraulic-Scram; PI-LOPP_CCF-Hydraulic-Scram_EX-DEC	15.5.5.2.1 15.5.5.2.2 15.5.5.2.3 15.5.5.2.4
Reactivity and Power Distribution Anomalies Bounding Event Summary		
BL-AOO	None	N/A
CN-DBA	Fuel Loading Error (FLE) RI-FLE_CN-DBA	15.5.4.3.1
EX-DEC	CCF- All Control Rod Withdrawal at Power - All Rods (CCF-ACRW) Passive CCF DL2 Technology Platform (CCF-DL2); RI-CCF-ACRW_CCF-DL2_EX-DEC Inadvertent Control Rod Withdrawal at Power - Single rod (ICRW) RI-ICRW_DL2-CCF_EX-DEC	15.5.5.6.1 15.5.5.6.2
Increase in Reactor Coolant Inventory Bounding Event Summary		
BL-AOO	Inadvertent Isolation Condenser Initiation – One Train (IICI-1) II-IICI-1_BL-AOO	15.5.3.4.1

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DSA Layer / Event Category	Event and Fault Sequence ID	Corresponding DSA Sections 15.5.3 Through 15.5.5 Event Summary Results
CN-DBA	Feedwater Flow Increase – All Pumps CCF-FWFI with Passive CCF DL2 Technology Platform (CCF-DL2); II-CCF_FWFI_CCF-DL2_CN- DBA	15.5.4.4.1
	Inadvertent Isolation Condenser Initiation - All Trains (CCF-DL4a-IIICI), Passive CCF DL2 Technology Platform (CCF-DL2); II-CCF-IIICI_CCF-DL2_CN- DBA	15.5.4.4.2
EX-DEC	None	N/A
Decrease in Reactor Coolant Inventory Bounding Event Summary (non-LOCA)		
BL-AOO	Feedwater Pump Trip – One Pump FWPT; IR-FWPT_BL-AOO	15.5.3.3.1
CN-DBA	CCF Loss of FW Flow Passive CCF DL2 Technology Platform (CCF-DL2); IR-CCF-LOFW_CCF- DL2_CN-DBA	15.5.4.5.1
	Reactor Pressure Vessel Pressure Controller Open CCF-RPCO, Passive CCF DL2 Technology Platform (CCF-DL2); IR-CCF- RPCO_CCF-DL2_CN-DBA	15.5.4.5.2
EX-DEC	Feedwater Isolation FW Isolation (CCF-FWI-DL3); IR-CCF- FWDL3_EX-DEC	15.5.5.8
Decrease in Reactor Coolant Inventory Bounding Event Summary (LOCA)		
CN-DBA	Main Steam Pipe Breaks Inside the Containment, Conservative Case	15.5.4.6.1
CN-DBA	Feedwater Pipe Break Inside the Containment, Conservative Case	15.5.4.6.2
CN-DBA	Large Isolation Condenser Pipe Breaks Inside the Containment	15.5.4.6.3
CN-DBA	Small Steam and Liquid Pipe Breaks Inside the Containment	15.5.4.6.4
CN-DBA	Large Main Steam Pipe Break Outside the Containment	15.5.9.2.1
CN-DBA	Large Feedwater Pipe Break Outside the Containment	15.5.9.2.2

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DSA Layer / Event Category	Event and Fault Sequence ID	Corresponding DSA Sections 15.5.3 Through 15.5.5 Event Summary Results
CN-DBA	Large Isolation Condenser Pipe Breaks Outside the Containment	15.5.9.2.3
CN-DBA	Small Breaks Outside the Containment	15.5.9.2.4

15.3 Safety Objectives and Acceptance Criteria

Implementation of the safety objectives established by the IAEA Safety Standards Series No SF-1 Fundamental Safety Principles ensure that the BWRX-300 facility when operated achieves the highest standard of reactor safety that can be reasonably achieved. The IAEA SF-1 Safety Objectives are discussed in Chapter 3, Subsections 3.1.1. and 15.1.2.

The deterministic and probabilistic safety analysis acceptance criteria are based on or derived from ensuring that the dose acceptance criteria of CNSC REGDOC-2.4.1, Section 4.3.2 (Reference 15.3-1), and CNSC REGDOC-2.5.2, Section 4.2.1 (Reference 15.3-2) are met.

15.3.1 Deterministic Safety Analysis Acceptance Criteria

The BWRX-300 design complies with acceptance criteria of CNSC REGDOC-2.4.1, Section 4.3.2 (Reference 15.3-1), and the established dose acceptance criteria in CNSC REGDOC-2.5.2 Section 4.2.1.

CNSC REGDOC-2.5.2 (Reference 15.3-2) states “that acceptance criteria shall be assigned to each plant state in the design, considering the principle that frequent PIEs have only minor or no radiological consequences, and that any events that may result in severe consequences are of extremely low probability.”

Qualitative acceptance criteria are defined and met for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. Safety goals are defined in REGDOC-2.5.2, Section 4.2.2 and are discussed in Chapter 3, Section 3.1.2.

Derived qualitative and quantitative acceptance criteria are used to analyze AOOs or DBAs. Qualitative acceptance criteria are supported by experimental data, prescribed by regulatory requirements, or prescribed by applicable codes and standards. The results of the quantitative safety analysis confirm the derived acceptance criteria (i.e., the limiting event in an event group).

Certain accidents with a predicted frequency of occurrence less than $1\text{E-}5/\text{rx-yr}$ DECAs may be used as design basis events for a safety system. In this case, the results are compared to the DBA dose limits, and qualitative acceptance are established. The DBA derived acceptance criteria are used as screening criteria to evaluate core damage and are used as input in determining the PSA safety goals.

The committed whole-body dose for average members of the critical groups who are most at risk, is calculated in the DSA for a period of 30 days after the analyzed event. As stated in Section 3.1.2, the calculated dose is less than or equal to the dose acceptance criteria of CNSC REGDOC-2.5.2, Section 4.2.1:

- 0.5 millisievert (mSv) for any AOO
- 20 mSv for any DBA

The dose results provided in Section 15.7 from the limiting events identified in Section 15.5 demonstrate that the radiological consequences of the analyzed events do not exceed the AOO, and DBA acceptance criteria listed in Tables 15.3-1 and 15.3-2, respectively.

15.3.1.1 Acceptance Criteria for Analysis of Anticipated Operational Occurrences

CNSC REGDOC-2.4.1, Section 4.3.2 requires derived acceptance criteria be established for AOOs and DBAs per CNSC REGDOC-2.4.1, Section 4.3.4. The derived acceptance criteria for the DSA of AOOs are shown in Table 15.3-1. These derived acceptance criteria are based upon the fission product barrier or FSF.

15.3.1.2 Acceptance Criteria for Analysis of Design Basis Accidents

CNSC REGDOC-2.4.1, Section 4.3.2 requires derived acceptance criteria be established for AOOs and DBAs per CNSC REGDOC-2.4.1, Section 4.3.4. The derived acceptance criteria for the deterministic safety assessment of DBAs are shown in Table 15.3-2.

15.3.2 Acceptance Criteria for Probabilistic Safety Assessment

The safety goals for the PSA of core damage frequency, small release and large release frequencies are shown in Table 15.3-3 and are consistent with the quantitative safety goals in CNSC REGDOC-2.5.2, Section 4.2.2.

15.3.3 References

- 15.3-1 CNSC Regulatory Document REGDOC-2.4.1, "Deterministic Safety Analysis."
- 15.3-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 15.3-3 IAEA Safety Standards Series No. SSR-2/1, "Safety of Nuclear Power Plants: Design" International Atomic Energy Agency.
- 15.3-4 NEDC-33840P, "The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance," GE-Hitachi Nuclear Energy Americas, LLC.

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**Table 15.3-1: Anticipated Operational Occurrence Deterministic Safety Assessment
Acceptance Criteria**

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
General	An AOO will not escalate to a more serious plant condition unless other faults occur independently.	Not applicable
	There is no loss of function of any fission product barrier.	Not applicable
Fuel Rod	Loss of fuel rod mechanical integrity will not occur due to fuel melting.	The calculated maximum fuel center temperature T_{center} remains below the fuel melting point T_{melt} . Subsection 4.2.3.4 describes the method used in calculating the maximum fuel pellet temperature.
	Loss of fuel rod mechanical integrity will not occur due to pellet-cladding mechanical interaction.	The cladding strain acceptance criteria defined in Section 5.0 of Reference 15.3-4. Chapter 4, Subsection 4.2.3.4 describes the code methodology used in calculating the cladding strain acceptance criteria.
	Fuel rod failure will not occur due to overheating of cladding	The calculated core Minimum Critical Power Ratio (MCPR) ensures that 99.9% of the fuel rods in the core are not susceptible to boiling transition during AOO events. Chapter 4, Section 4.4.1 describes the establishment of the fuel cladding safety limit. With the reactor steam dome pressure less than 4.72 MPaG (685 psig), the calculated reactor thermal power is less than 25% of rated thermal power.
Reactor Coolant Pressure Boundary	Design conditions of the reactor coolant pressure boundary are not exceeded during the most severe pressurization transient.	The calculated peak pressure associated with the reactor coolant pressure boundary shall not exceed 110% of the design pressure or 11.38 MPaG (1650 psig).
	The reactor coolant pressure boundary maintains sufficient reactor coolant inventory for core cooling.	The calculated reactor water level is maintained at or above TAF.

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Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
Primary Containment	Containment integrity is maintained. If an AOO results in an energy release to the containment, or loss of containment heat removal, then containment stresses (i.e., pressure and temperature) are limited such that there is no loss of a containment barrier safety function, and thus, the containment remains within its design limit values.	No AOOs result in a significant energy release to containment, or prolonged loss of normal containment cooling. The normal operation limits and conditions are applied to containment, and no AOO containment quantitative criteria is needed. Chapter 9, Subsection 9A.5.6 describes the containment cooling system functional design.
Long-Term Heat Removal	SSC important for preserving the integrity of the reactor core and the containment are capable of removing residual heat for an extended period both during and after all applicable PIEs considered in all Operational States, including AOOs.	Following AOO events that do not result in shutdown, a controlled condition is achieved. Following AOO events that require shutdown, the core remains shutdown independent of operator action or offsite support for at least 72 hours. AOO events that rely on DL3 mitigation for long-term cooling are capable of providing cooling for at least 72 hours without operator action or offsite support.

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Table 15.3-2: Design Basis Accident Acceptance Criteria

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
General	Except for fuel cladding, there is no loss of function of any fission product barrier.	Not Applicable
Fuel Rod Failure	The number of fuel rod failures is conservatively estimated for DBAs.	The calculated number of failed rods does not result in exceeding the applicable radiological dose acceptance criteria.
	Mechanical fracturing of a fuel assembly under DBA loading conditions does not result in losing the ability to cool the fuel assembly.	The mechanical integrity of the fuel is established from the mechanical and thermal fuel analysis described in Chapter 4, Subsection 4.2.2.
Fuel Cooling	The calculated fuel cladding temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.	The calculated PCT remains less than 1204°C (2200°F). The calculated total oxidation of the cladding nowhere exceeds 0.17 times the total cladding thickness before oxidation for DBAs where exceeding the oxidation thickness challenges the capability to cool the core.
Reactor Coolant Pressure Boundary	Design conditions of the reactor coolant pressure boundary are not exceeded during the most severe pressurization transient as a result of a DBA.	The calculated peak pressure associated with the RCPB shall not exceed 120% of the design pressure or 12.41 MPaG (1800 psig).
	The reactor coolant pressure boundary maintains sufficient reactor coolant inventory for core cooling.	Conformance is demonstrated by meeting the fuel cooling and long-term heat removal criteria.

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Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
Primary Containment	Containment pressures and temperatures are maintained below the design values.	The calculated containment pressure does not exceed the design pressure 0.414 MPaG (60 psig). The calculated containment shell temperature does not exceed the design temperature 165.6°C (330°F).
	The local combustible gas concentrations in the containment are within the range where deflagration or detonation cannot occur.	Containment atmosphere remains sufficiently mixed such that deflagration or detonation thresholds are not exceeded.
	Containment energy management systems are capable of reducing the containment pressure and temperature following a DBA to minimize the release of fission products to the environment and to preserve containment integrity and leak tightness.	The calculated containment pressure reduces to less than 50% of the calculated peak pressure for the most limiting LOCA within 24 hours.
Reactivity Control	Reactivity control required to bring the reactor to cold shutdown is maintained.	Shutdown margin is established to assure that the reactor can be brought subcritical with the highest-worth control rod pair withdrawn when the core is in its most reactive condition. The subcriticality value is 0.38% $\Delta k/k$ with the highest-worth control rod pair analytically determined.
Long-Term Heat Removal	SSCs important for preserving the integrity of the reactor core and the containment are capable of removing residual heat for an extended period both during and after all applicable PIEs considered in all operational states, and DBAs.	Long-term cooling is maintained for a minimum of 72 hours independent of operator action and offsite support, and for 30 days with credit for operator actions and on-site resources. For DBA events that result in shutdown, the plant can achieve and maintain safe-shutdown conditions with the average reactor coolant temperature below 215.6°C (420°F).

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Table 15.3-3: Probabilistic Safety Goals

Qualitative Acceptance Criteria	Derived Quantitative Acceptance Criteria
Core damage frequency	The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than $1\text{E-}5/\text{rx-yr}$.
Small release frequency	The calculated sum of frequencies of all event sequences that can lead to any release to the environment that requires temporary evacuation of the local population or a release to the environment of more than $1\text{E}15$ becquerels of iodine-131, shall be less than $1\text{E-}5/\text{rx-yr}$.
Large release frequency	The calculated sum of frequencies of all event sequences that can lead to any release to the environment that requires long-term relocation of the local population or a release to the environment of more than $1\text{E}14$ becquerels of Cesium-137 shall be less than $1\text{E-}6/\text{rx-yr}$.

15.4 Human Actions

The BWRX-300 design approach minimizes the complexity of the SSC design, while enhancing reliability and reducing the potential for human error. The systems and components are designed to reduce the necessity for human actions (refer to Chapter 18 Human Factors Engineering). Where complexity is necessary in the design (e.g., self-diagnostic tools, redundancy in equipment in a single division), the design complexity is documented and justified for enhancing reliability, surveillance, calibration, and other equipment attributes.

Chapter 18 discusses the human factors engineering and human-machine interface consideration during development of the facility design that facilitate interaction between operating personnel and the plant. Subsection 15.1.3 discusses the human operation hazard evaluation under the safety assessment framework.

Chapter 13 provides a program description to manage operational aspects that are affected by human factor considerations, including the continued review and development of measures in place. The chapter also describes the organizational provisions that ensure operators are able to effectively perform in the main and secondary control rooms and other parts of the plant under all operational circumstances, including proposed shift schemes and rotations, assessment of operator's fitness for duty, and other human factors related issues.

15.4.1 Human Actions in Deterministic Safety Analysis

There are no operator actions credited in responding to the events analyzed in Section 15.5 for the DSA. This assurance is attributed to the following design features:

1. Fail-safe (not reliant on external power) safety system actuations ensure that the FSFs are fulfilled for a DBA
2. Automatic, reliable actuation of the control rods with either stored energy or motors to shut down the reactor and maintain it in a guaranteed shutdown state via latching mechanisms
3. ICS provides passive decay heat removal
4. Fail-safe containment isolation and passive containment heat removal

Critical safety parameter monitoring is provided in the Secondary Control Room (SCR) if the Main Control Room (MCR) becomes uninhabitable.

The HOHE identifies failures that involve an erroneous decision or action taken by a human that can lead to an unplanned plant transient that is evaluated as a PIE in the hazard analysis in Section 15.1.3.

15.4.2 Human Actions in Probabilistic Safety Assessment

Human actions resulting from PSA event evaluations are discussed in Subsection 15.6.1.3.5.

15.5 Deterministic Safety Analysis

15.5.1 General Description of the Approach

The DSA is divided into two parts:

- Part One – the plant response to fault sequences is evaluated and analyzed to confirm the performance of the fission product barriers against the derived acceptance criteria
- Part Two - the event dose consequences resulting from a fission product release or other source of radiation, such as the reactor coolant, is radiologically analyzed

The DSA is performed based on the outputs of the hazard analysis and fault evaluations. There are three layers of DSA performed (see 15.1 and 15.2.1 for additional details) that credit different sets of DLs:

Deterministic Safety Analysis Approach for Non-LOCA Events

Transient DSA analyzes fault sequences where the reactor coolant pressure boundary remains intact. These events are broken down into groups that result in similar core responses. Section 15.2.4 describes the core response during off-normal conditions, the groups determined for BWRX-300 (Table 15.2-1), and the selected bounding event scenarios for AOO, DBA and DEC without core damage analyses (summarized in Table 15.2-2).

The BWRX-300 scenarios are identified through the fault evaluations. The methods and assumptions described in the TRACG Application for BWRX-300 (Reference 15.5-3) are used to confirm the performance of the fission product barriers for the DSA non-LOCA events. The TRACG Application for both non-LOCA and LOCA event analysis is discussed in Subsection 15.5.1.2.1.

The TRACG Application for BWRX-300 also includes the stability analysis that evaluates potential coupled thermal-hydraulic – neutronic instabilities in the reactor core. TRACG is used to perform transient safety analysis and stability analysis for both forced flow and natural circulation BWR designs. Previous TRACG applications as well as BWRX-300 use the systematic approach developed for the US NRC, called Code Scaling, Applicability and Uncertainty (CSAU) to confirm the applicability of a computer code for DSA. This approach involves systematic evaluation of the phenomena that are important for the plant design and accident scenarios identified. A qualitative process is used to identify and rank the importance of phenomena. Through this process a Phenomenon Identification and Ranking Table (PIRT) is established. The PIRT is used together with the TRACG documentation to systematically demonstrate the applicability of TRACG models and the qualification of the TRACG model to predict the phenomena. Defining the nodalization and evaluation of the effects of scale are included. In addition to code applicability and qualification, the PIRT is also used as the basis to perform quantitative uncertainty analysis of transient scenarios, if needed. Additional information regarding the approach for addressing uncertainty in the DSA is provided in Subsection 15.5.1.1

The TRACG applicability to model phenomena also requires that the code capability be demonstrated to apply the code in the intended manner with a qualifying result achieved. TRACG capability to model phenomena is important to BWRX-300 simulation and is consistent with modern best practices. TRACG qualification is based upon proven practices for verification and validation using acceptable codes and standards (see Chapter 3, Appendix 3G). Experiments and plant events used to validate TRACG provide evidence that TRACG can be applied for the BWRX-300 design.

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Integral to the capability of TRACG for transient DSA is the use of three-dimensional nuclear kinetics input. This input comes directly from and essentially uses the same methods as the steady-state core simulator, PANAC11. PANAC11 is used in the BWRX-300 as described in Subsections 4.3.2 and 15.5.1.2.3. Other code interfaces are described in TRACG Application for the BWRX-300 (Reference 15.5-3).

Design control procedures require independent verification of safety analysis calculations to ensure that results are properly summarized from calculations, physically sound/correct, and consistent with expected results when compared to previous calculations. The results are then confirmed to meet the appropriate acceptance criteria. Table 15.5-48 "Conservatisms Used in the Non-LOCA DSA" provide additional insight in conservatisms used in the transient analysis.

Deterministic Safety Analysis Approach for LOCA Events

The methods and assumptions of the DSA confirming the performance of the fission product barriers for LOCAs are described in Licensing Topical Report (LTR) NEDC-33922P-A, Revision 3, BWRX-300 Containment Method (Reference 15.5-2).

TRACG calculates the mass and energy release from modeled breaks of various sizes and locations. Atmospheric pressure is used for the TRACG pressure boundary condition for any breaks. This approach provides no credit for the back pressure from containment. Consequently, the retained RPV inventory calculated by TRACG represents the minimum coolant volume. This modeling provides results as if the break occurred outside containment.

Breaks inside containment realistically experience back pressure from containment that reduces the mass and energy calculated by TRACG once the break flow becomes unchoked. However, this effect is not treated explicitly because it requires two-way coupling between the TRACG calculation and the GOTHIC containment calculation. Instead, the methodology has a one-way coupling with the mass and energy release rates conservatively calculated by TRACG that supplies inputs to the GOTHIC calculation up until the point in time when the containment and RPV pressures first equalize. Choked flow naturally satisfies the assumed one-way coupling because choked flow does not depend on the downstream pressure. Select TRACG inputs are specified so that mass and energy release rates are conservatively calculated.

Rapid mass and energy releases into containment occur before a large break is isolated. This leads to the highest containment peak pressure at approximately the same time that the break is isolated. For large breaks, the containment shell is the dominant short-term energy sink, and it causes containment pressure to decrease from its peak value after isolation of the break occurs.

Compared to large breaks, small unisolated breaks have a much slower mass and energy release rate from the RPV into containment. The lowest break on the RPV that remains unisolated and occurs outside containment produces the most limiting scenario for minimum RPV inventory. Regardless of break location and whether it is inside or outside containment, break flow slowly decreases with time because the RPV is being depressurized largely due to the ICS and to a much lesser extent by the break flow.

Containment pressure slowly increases and eventually equals the RPV pressure for breaks inside containment. It is not realistic to use the TRACG break flow that was calculated using an atmospheric pressure boundary condition as input to the GOTHIC containment calculation after the point in time when containment and RPV pressures first equal each other. A better approximation is to assume zero break flow after this point in time, but this could potentially be nonconservative with respect to the longer-term calculated containment pressures. The proposed methodology does not require the GOTHIC calculation to continue beyond the point where the containment and RPV pressures equalize, because the longer-term containment pressure is bounded by the RPV pressure calculated.

The methods and assumption for normal coolant radiological analyses are described for these events in Section 15.7.

15.5.1.1 Safety Margins in Safety Analysis

The safety margin is the result of the conservative assumptions used in the analysis and design rules applied to the SSC design capabilities. In addition, the DSA demonstrates that the challenges to the physical barriers do not exceed their physical capacity.

Uncertainties in initial conditions and methods are accounted for in the CN-DSA. The BL-DSA and EX-DSA allow best-estimate methods to be consistently applied using REGDOC-2.4.1, Section 4.4.2 guidance. For CN-DSA thermal-hydraulic analysis, a graded approach is used in combining uncertainties. The graded approach involves a qualitative assessment of the safety margin on a case-by-case basis and includes a review of the magnitude of results compared to acceptance criteria along with the judgment of conservatism in the derived acceptance criteria.

The DSA confirms the FSFs successfully keep plant radioactive material releases within the acceptance criteria with adequate safety margins.

15.5.1.1.1 Large Margin

For events with large margin or substantially non-limiting, there is no need to apply uncertainty to the analysis methodology. Judgment is used to establish what is “large margin” or “substantially non-limiting”. Instead of a quantitative evaluation of uncertainty, the event is dispositioned qualitatively based on the uncertainty evaluation performed for a limiting event of a similar type, historical analysis of similar type, or other qualitative based disposition.

Large Margin Examples

The inadvertent isolation condenser initiation in Subsection 15.5.4.4.2 and the closure of all MSRIVs and FWIVs in Subsection 15.5.4.2.4 are examples of large margin events. There are many DBA events that have minimal impacts compared to the acceptance criteria.

Inadvertent Isolation Condenser Initiation

This CN-DBA event results in a peak pressure of 7.32 MPaG, much lower than the acceptance criteria of 12.41 MPaG and is bounded by pressure increase CN-DBA events in Subsection 15.5.4.2.1. Also, the PCT is 307.9°C, much lower than the acceptance criteria in Table 15.3-2 and is bounded by the generator load rejection CN-DBA event in Subsection 15.5.4.2.1. In this event, there is no concern for cladding oxidation, and there is no threat to the containment pressure boundary.

Closure of All Main Steam Reactor Pressure Vessel Isolation Valves and FW Isolation Valves

This CN-DBA event results in a peak pressure of 8.73 MPa, much lower than the acceptance criteria of 12.41 MPaG. The peak pressure is bounded by other pressure increase CN-DBA events in Subsection 15.5.4.2. Also, the PCT is 312.47°C, much lower than the acceptance criteria in Table 15.3-2. This event has a large safety margin to the DBA acceptance criteria. In this event, there is no concern for cladding oxidation, and there is no threat to the containment pressure boundary. As a result, for the above events and any other events with similar margin, there is no need for any quantification of uncertainty and a qualitative disposition is adequate.

15.5.1.1.2 Medium Margin

In medium margin scenarios, method uncertainty is addressed by biasing key important phenomena in a conservative direction (typically one or two sigma). Input parameters such as power, pressure, level, temperature are based on using the most limiting normal operating values. In these cases, the selection of key important phenomena is dependent on the specific event evaluated. Important phenomena can be different for the output parameters when multiple output parameters are considered for selecting the bias direction. The selection of the key important phenomena and determining the bounding bias direction is considered for each output parameter that has medium margin and compared to the derived acceptance criteria.

Medium Margin Example

A medium margin event is the large break inside containment described in Subsection 15.5.4.6.

Large Pipe Breaks Inside Containment

There are multiple large pipe breaks inside containment event scenarios evaluated that have commonalities. These events result in no significant fuel cladding heat-up and are not bounding with respect to maintaining inventory above the fuel (to ensure continued cooling). They represent the largest challenge to the containment fission product barrier. These events are treated as medium margin events and the initial conditions and modeling parameters are biased to ensure conservative containment conditions are calculated. The initial conditions used are provided in Table 15.5-1. The modeling parameters biased in this analysis are discussed in NEDC-33922P (Reference 15.5-2). The combination of the conservative biased inputs combined with the observation of the margin available results in conservative analyses.

15.5.1.1.3 Low Margin or Quantitative Evaluation of Uncertainty is Desired

A proven Monte Carlo technique is used to combine the individual biases and uncertainties into an overall bias and uncertainty for low margin events. This process is described in the TRACG Application for BWRX-300 (Reference 15.5-3). There are no events identified as low margin.

15.5.1.2 Description of the Computer Codes Used in the Safety Analyses

There is a large amount of data available from operating BWR plants and from the testing and licensing efforts to licence the predecessor BWR/Advanced Boiling Water Reactor (ABWR)/Economic Boiling Water Reactor (ESBWR) designs and individual plants. The vast database of feature performance in licensed reactors, combined with the recent thorough licensing review of the ABWR and ESBWR performed by the USNRC provides an extremely well qualified foundation from which to make the modest extrapolations to the BWRX-300. The following codes, methods, and accompanying assumptions are used in evaluating the performance of the BWRX-300 during postulated initiating events. The radiological responses to DBAs and DECAs are presented in Section 15.7 but use USNRC-accepted RADTRAD and PAVAN codes for doses at the exclusion boundary and atmospheric dispersions, respectively. A description of codes used in the BWRX-300 safety analysis is provided in Chapter 3, Appendix 3G.

15.5.1.2.1 TRACG

TRACG is a GEH proprietary version of the Transient Reactor Analysis Code (TRAC). TRACG is the primary licensing analysis tool for LOCA and transient analyses for PIEs with a large range of frequencies up to events that do not involve significant core damage (severe accidents). TRACG has been used in a variety of applications for operating BWRs as well as design/analysis for the ESBWR.

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TRACG uses advanced realistic one-dimensional and three-dimensional methods to model the phenomena that are important in evaluating the operation of BWRs. It is a best-estimate code for analysis of BWR transients ranging from simple operational transients to design basis LOCAs and failure to scram transients. TRACG has an extensive qualification base for separate effects, BWR fuel and components, and integral tests. It has been reviewed and approved by the US NRC for a number of analysis applications such as AOOs, ECCS/LOCA and failure to scram overpressure (a BDBA event for BWRs) analyses.

TRACG is used to analyze the challenges to the fuel, RPV, and the mass and energy releases to the containment, for LOCA and non-LOCA DSA. TRACG draws from the licensed BWRs database, which includes design features of the BWRX-300 (albeit in various configurations) and appropriate testing and allows direct application to BWRX-300 design and analysis. TRACG is maintained and updated by GEH.

Scope and Capabilities

TRACG is based on a multi-dimensional two-fluid model for the reactor thermal-hydraulics and a three-dimensional neutron kinetics model. The two-fluid model used for the thermal-hydraulics solves the conservation equations for mass, momentum, and energy for the gas and liquid phases. TRACG does not include any assumptions of thermal or mechanical equilibrium between phases. The gas phase may consist of a mixture of steam and a non-condensable gas, and the liquid phase may contain dissolved boron. The thermal-hydraulic model is a multi-dimensional formulation for the vessel component and a one-dimensional formulation for all other components.

The conservation equations for mass, momentum, and energy are closed through an extensive set of basic models consisting of constitutive correlations for shear and heat transfer at the gas/liquid interface as well as at the wall. The constitutive correlations are flow regime-dependent and are determined based on a single flow regime map, which is used consistently throughout the code. In addition to the basic thermal-hydraulic models, TRACG contains a set of component models for BWR components, such as fuel channels, steam separators, and can simulate BWR steam dryers as part of its vessel model. TRACG also contains a control system model capable of simulating the major BWR control systems such as those for pressure and water level.

The neutron kinetics model is consistent with the GEH BWR Core Simulator PANACEA. It solves a modified one-group diffusion model with six delayed neutron precursor groups. Feedback is provided from the thermal-hydraulic model to the kinetics model for moderator density, fuel temperature, boron concentration and control rod position.

The TRACG structure is based on a modular approach. The TRACG thermal-hydraulic model contains a set of basic components, such as pipe, valve, tee, channel, steam separator, heat exchanger and vessel. System simulations are constructed using these components as building blocks. Any number of these components may be combined. The number of components, their interaction, and the detail in each component are specified through code input. Consequently, TRACG has the capability to simulate a wide range of facilities, ranging from simple separate effects tests to complete BWR plants.

TRACG has been extensively qualified against separate effects tests, component performance data, integral system effects tests and full-scale BWR plant data. A detailed documentation of the TRACG qualification is contained in the TRACG Qualification Report (Reference 15.5-1).

The total effort and extent of qualification performed on TRACG, since its inception in 1979, now exceeds, both in extent and breadth, that of any other engineering computer program GE/GEH has submitted to the USNRC for design application approval.

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The analysis also determines the most limiting overpressure protection events in terms of peak vessel pressure. The results are used to demonstrate adequate pressure margin to the reactor vessel design limit.

Scope of Application of TRACG to BWRX-300

The PIRT discussed in References 15.5-2 and 15.5-3 identify specific governing phenomena, where a significant fraction were concluded to be "important" in predicting BWRX-300 transient and LOCA performance. Most of these phenomena are common to operating BWRs. This section examines specific SBWR/ESBWR-related tests and test facilities beyond the previous qualification database. Early in the SBWR program, it was identified that there was no information in the data base for a heat transfer correlation for steam condensation in tubes in the presence of non-condensable gases. A Single Tube Condensation Test Program was conducted to secure this information and reported to the USNRC in TRACG Qualification Report for SBWR (NEDC-32725P, Revision 1) and ESBWR NEDC-33080P-A, Revision 2) that are used in the TRACG model for the BWRX-300 as described in LTR NEDC-33922P (Reference 15.5-2).

The test program was conducted to investigate steam condensation inside tubes in the presence of non-condensable gases. The work was independently conducted at the University of California at Berkeley and at the Massachusetts Institute of Technology (MIT). The work was initiated to obtain a database and a correlation for heat transfer and condensation inside tubes. Three researchers utilized three separate experimental configurations at the University of California at Berkeley, while two researchers utilized one configuration at MIT. The researchers ran tests with pure steam, steam/air, and steam/helium mixtures with representative and bounding flow rates and non-condensable mass fractions. The researchers found the system well-behaved for all tests with either of the non-condensable gases. The results of the tests at the University of California at Berkeley are the basis for the condensation heat transfer correlation used in the TRACG computer code.

TRACG ICS modeling is qualified by the PANTHERS IC test using a representative configuration. The steady-state heat exchanger performance was predicted by the PANTHERS IC prototypical geometry full-scale test.

Because the BWRX-300 RPV and ICS are similar to those of the ESBWR, the TRACG method developed for the ESBWR RPV thermal-hydraulics and mass energy release is also used for the BWRX-300 RPV thermal-hydraulics and mass and energy release. The TRACG code and the application method developed for ESBWR was reviewed and approved by the USNRC. That application method was developed using the Code Scaling, Applicability and Uncertainty (CSAU) guidance. GEH Licensing Topical Report NEDC-33922P BWRX-300 Containment Evaluation Method (Reference 15.5-2) provides an overview of the TRACG thermal-hydraulics method for the mass and energy release and its applicability to the BWRX-300 RPV.

15.5.1.2.2 GOTHIC

Containment analysis is performed by using the GOTHIC code (References 15.5-4 and 15.5-5).

The GOTHIC computer code is a state-of-the-art program for modeling multiphase, multicomponent fluid flow for performing both containment DBA analyses and analyses to support equipment qualification. The GOTHIC code is developed by Numerical Applications Incorporated (NAI), and the development program is sponsored by the Electric Power Research Institute (EPRI).

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The GOTHIC code has a nodding structure that allows both lumped parameter and 3-D modeling capabilities. The multi-dimensional analysis capability facilitates the study of non-condensable gas and stratification and the calculation of flow field details within any given volume. The code has undergone extensive review and validation against a large test array. The validation program scope examines the code capability for predicting pressure and temperature as well as hydrogen distribution and mixing under various conditions.

GOTHIC is a continuously maintained and improved computer code. The GOTHIC code has been developed compliant with US Title 10 Code of Federal Regulations Part 50, Appendix B, Quality, meeting the GEH software quality requirements and complies with REGDOC-2.4.1, Section 4.4.5 / N286.7 (Reference 15.5-6) requirements. The PSAR results generated used GOTHIC Version 8.3 that is the latest released version. Future BWRX-300 containment analyses may be performed using newer versions of the GOTHIC code provided the newer versions meet the same USNRC 10 CFR 50, Appendix B quality and REGDOC-2.4.1, Section 4.4.5 / N286.7 requirements, as well as changes in calculated results for the BWRX-300 containment application caused by any code changes that can be successfully dispositioned.

15.5.1.2.3 PANAC11

The BWR Core Simulator (PANAC11 or P11) is a steady-state, 3-D coupled nuclear-thermal-hydraulic computer program representing the BWR core exclusive of the external flow loop. An Automated Plant Heat Balance (APHB) is available to model an external flow loop. Provisions are made for fuel cycle and thermal limits calculations. Neutronic parameters used by core simulator are obtained from two-dimensional lattice physics and parametrically fitted as a function of moderator density, exposure, control, and moderator density history for a given fuel type. The simulator is used for detailed 3-D design and operational calculations of BWR neutron flux and power distributions and thermal performance as a function of control rod position, refueling pattern, coolant flow, reactor pressure, and other operational and design variables. A special power exposure iteration option is available for target exposure distribution and cycle length predictions. PANAC11 includes the effect of Doppler broadening as a function of moderator density, exposure, control, and moderator density history for a given fuel type. The lattice physics and core simulator form the nuclear simulator and is used for both core design and core exposure tracking.

The nuclear model is based on a coarse-mesh nodal, 1-1/2 group (quasi-two group), static diffusion theory. The diffusion equations are solved using the fast energy group. Resonance energy and thermal energy neutronic effects are included in the model by relating the resonance and thermal energy fluxes to the fast energy flux. Eigenvalue iteration yields the fundamental mode solution. Control blade history local peaking effects are considered. A pin power reconstruction model is incorporated to account for the effect of flux gradients across the nodes on the local peaking distribution. Instrumentation predictions are made for neutron and gamma sensitive detectors.

Neutronic parameters used by PANAC11 are obtained from the 2-D lattice physics code TGBLA06, as described in the preceding section, and parametrically fitted as a function of moderator density, exposure, control, and moderator density history for a given fuel type.

The nuclear model is coupled to a static, parallel channel, thermal-hydraulics model and is consistent with the approach described in the Global Nuclear Fuel (GNF) BWR steady-state thermal-hydraulic method NEDC-32082P (Reference 15.5-7).

The GE BWR Core Simulator has undergone extensive validation by comparing calculated results with alternate 3-D methods, end of cycle gamma scan data, operating reactor data, and transient experimental data.

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PANAC11 is an USNRC-approved method used for production core design and licensing analysis for the fleet of BWRs with GNF fuel. TGBLA06 and PANAC11 form the nuclear simulator and are used for both core design and core exposure tracking. This tracking process is a continuous comparison and validation of the nuclear methods descriptions, a demonstration of methods performance through operational qualification through similar reactors for the BWRX-300, and the quantification of uncertainties that are applicable in thermal margin evaluations that are a function of this performance are provided in NEDC-33939P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology," (Reference 15.5-16).

15.5.1.2.4 ANSI/ANS-18.1-2020

Radiation concentrations in the reactor coolant and steam during normal operations are determined based on ANSI/ANS-18.1-2020 (Reference 15.5-10). This standard provides the bases for estimating typical concentrations of the principal radionuclides that may be anticipated over the lifetime of a BWR plant. The source term data is based on the cumulative industry experience at operating BWR plants, including measurements at several stations. The operating data reflects the influence of a number of observations made during the transition period from operation with fuel of older designs to operation with fuel of current improved designs such as the GNF2 fuel used in the BWRX-300.

15.5.1.2.5 ADDAM

The ADDAM (Atmospheric Dispersion and Dose Analysis Method) computer code (Reference 15.5-8) computes the statistical distribution of radiation doses to an individual or population after the airborne release of radioactive material into the environment following a design-based nuclear accident at a nuclear facility. The dispersion of the release is highly influenced predominately by different characteristics of release, existing meteorological scenarios, and the overall nearby terrain and building dimensions. Doses can be computed for various age groups, organs, and receptor locations which are classified based on release and exposure pathway. Input data into ADDAM is divided into four sections:

- Meteorological conditions
- Site characteristics
- Release characteristics
- Receptor characteristics

The execution of the ADDAM code is accomplished by the following input data files:

- Meteorological
- Radionuclide
- Release activity
- Stack exit temperature and velocity file
- Release thermodynamic data files

ADDAM complies with the requirements of CSA N288.2-19 (Reference 15.5-9) standard and is the current CANDU Owners Group's Industry Standard Toolset Version for analyses. CSA N288.2-19, Clause 1.5 states that the "standard covers local atmospheric dispersion, which for Gaussian plume models is defined as dispersion that occurs in the range of 300 m to 100 km." While it does not explicitly preclude use below exclusion zone radii below 300 m, it does mention the need for specialized models. The application within is the current recommendations of the standard as implemented in ADDAM. The individual public doses, dilution factors (χ/Q), and deposition factors (D/Q) at a 350m exclusion zone boundary radius for a 95th percentile cumulative frequency of occurrence cut-off have been calculated using a modified reference calculation and output options is used for all the production runs for dilution factors and dose assessment as these are standard output options of ADDAM.

15.5.2 Analysis of Normal Operation

The normal operation DSA demonstrates that plant parameters are maintained within specified Operating Limits and Conditions (OLCs) ensuring the plant conforms with the safety analysis assumptions. The main objective of the first defence level of a D-in-D strategy is to prevent challenges to plant equipment and to protect the primary physical barriers – fuel cladding and RCPB. DL1 measures are described in Chapter 3, Section 3.1.6.2.

The measures used to limit radiological releases to the public during normal operations are described in Chapter 11 Management of Radioactive Waste Systems, Chapter 12 Radiation Protection, and Chapter 20 Environmental Aspects.

15.5.2.1 Description of Normal Operational Modes

Normal operation is defined as operation within specified OLCs. An analysis of normal operation includes all operational modes (Modes 1-6 defined in Chapter 16, Appendix 16A).

15.5.2.2 Method and Scope of Analysis

The normal operation deterministic safety analysis demonstrates that the plant parameters are maintained within the specified OLCs ensuring that the plant conforms with the safety analysis assumptions. The normal operation of the plant is monitored and controlled so that PIEs that may lead to AOOs are mitigated before evolving into DBAs. OLCs are important DL1 measures that are readily checked by the operators ensuring the facility is operated in accordance with the applicable safety analyses. OLCs define minimum levels and allowable configurations of the plant, equipment, and associated resources needed to enact safety measures. Setpoint limits and conditions define where safety measures are intended to be activated or initiated to protect against or mitigate fault sequence consequences.

Source term analysis is provided in Chapter 11, Section 11.1 for normal operation sources. Radiation protection measures analyzed and implemented for the plant, shielding and radiation zone designation is provided in Chapter 12, Sections 12.4 and 12.5, respectively.

15.5.2.3 Results of Analysis

The limiting safety system setpoints are determined to demonstrate to the extent practicable that the highest safety class (DL3) systems are only initiated when needed. Process controls and alarms are established that demonstrate their effectiveness in reducing or avoiding the need for safety system actuations. The core parameters evaluation ensures that reactivity control required to bring the reactor to a cold shutdown condition is achieved and maintained. Normal operational doses are provided in Chapter 20, Section 20.8.

15.5.2.4 Stability Analysis

Part of normal operational analysis is to confirm that the core will remain stable during normal operation. The stability considerations during normal operation are described in Chapter 4, Section 4.8.

The BWRX-300 240-bundle core is evaluated for beginning of cycle (BOC), middle of cycle (MOC) and end of rated power (EOR) exposures in determining both the core-wide decay ratios and regional mode oscillations. The core-wide decay ratios (DR) are evaluated using a step perturbation in pressure while the regional mode decay ratios are evaluated using channel velocity perturbations.

Core-Wide Decay Ratio

The primary stability evaluation is performed at nominal conditions including nominal feedwater temperature 241.9°C (467.4°F).

Another stability evaluation is performed for the state that is reached after a LFWH AOO. A Select Control Rod Run-In (SCRRI) (described in Chapter 7, Subsection 7.3.3.1) is initiated as a mitigating response for a LFWH AOO with a feedwater temperature reduction of 16.7°C (30.0°F) or higher and a new lower power steady condition is achieved. Once the reactor achieves a new steady-state condition, a step pressure perturbation is applied to evaluate core stability response. The analysis of LFWH AOO described in Section 15.5.3.1.1 assumes that the feedwater temperature is reduced to 191.9°C (377.4°F) at BOC, MOC and EOR.

Core-Wide Dominance

For the regional mode evaluation, based on the harmonic modes distribution of the core, the inlet velocities for all channels were perturbed by $\pm 20\%$ at time = 0. This harmonic power distribution is predicted by the steady-state core simulator PANAC11 and results in a line of symmetry between the two halves of the core with higher and lower predicted harmonic power. The velocity perturbations are made positive on one side of the line of symmetry and negative on the other side. This stimulates the potential harmonic oscillations (regional oscillations). The resulting channel power response of limiting channels is evaluated for susceptibility to regional mode oscillations. If the core is not susceptible to regional mode oscillations after a velocity perturbation, the initially symmetric, out of phase channel power responses come into phase after a short duration, confirming the dominance of core-wide oscillations.

Results

DR / SCRRI

The maximum nominal core-wide decay ratio design limit is 0.80. The calculated core-wide decay ratios at nominal conditions are below the maximum decay ratio allowed. The calculated DR values at the end of the LFWH event are below the maximum allowed DR and are lower than the DR values at nominal conditions. The calculated DR values at nominal temperature and LFWH conditions are presented in Table 15.5-2. The nominal stability response is presented on Figure 15.5-1 for the MOC exposure. The LFWH stability response is presented on Figure 15.5-2 for the MOC exposure.

Core-Wide Dominance

A limiting core-wide dominance evaluation is performed at 115% rated power. At MOC, the limiting channel power values follow the expected behavior where initially symmetric, out of phase channel power responses come into phase after a short duration. The core is not susceptible to regional mode oscillations at nominal conditions, and this conclusion also applies to normal operation. The regional stability response using the limiting channels is presented on Figure 15.5-3 at the MOC exposure. This is a hypothetical evaluation, and any growing core-wide oscillations are mitigated by DL3-05 (see Table 15.5-5).

Based on these stability evaluations, the following stability claims are supported:

1. Power oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible
2. Regional instability is not possible
3. Design features prevent the loss of stability margin for upset events

15.5.3 Analysis of Anticipated Operational Occurrences

15.5.3.1 Decrease in Core Coolant Temperature AOO

This section describes the bounding BL-AOO event for Temperature Decrease (TD) Group.

15.5.3.1.1 Loss of Feedwater Heating AOO

This event is designated as a BL-AOO event. The fault sequence name is Loss of Feedwater Heating (LFWH) and the fault sequence ID is TD-LFWH_BL-AOO.

Additional LFWH AOO cases support the detailed evaluation and demonstrate the thermal-hydraulic stability of the BWRX after a LFWH AOO (see 15.5.2.4).

Postulated Initiating Event

The event assumes a loss of FW heating from a single failure of either the closure of one extraction steam valve or the inadvertent bypass of a FW heater. This failure is conservatively modeled as an instantaneous decrease in FW temperature that bounds the maximum FW temperature decrease resulting from a single failure. A PIE with AOO frequency results in the maximum FW temperature reduction identified in Table 15.5-3 from a loss of one feedwater heater.

Sequence of Events

The event fault sequence summary:

- Loss of FW temperature occurs instantly resulting in an increase in power
- Reactor Level Control (RLC) compensates initially by lowering flow rate minimizing the effect on power
- SCRRRI inserts control rods on indication of FW temperature reduction
- Reactor Pressure Control (RPC) maintains pressure and RLC maintains level
- A new controlled steady-state condition achieved with a new power distribution

Table 15.5-6 lists the sequence of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-27 – SCRRRI on FW Temperature Decrease
- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by decreasing the FW temperature. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The analysis is performed using an equilibrium core design. The event is run at BOC, MOC and EOR cycle exposure conditions.

Results

The results of the simulated loss of FW heating event are presented on Figures 15.5-5 through 15.5-10. The results are presented in Table 15.7.2-1 for the exposure with the limiting CPR response.

The reduced temperature FW enters the core and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. RLC compensates initially by lowering FW flow rate and a SCRRRI is initiated to minimize the core power increase and decrease the final steady-state power. Steam flow and FW flow then stabilize at a lower level. The RPV water level decreases and then returns to normal level. The pressure and level remain well within the RPV water level and RCPB pressure acceptance criteria in Section 15.3.2.

The core power increase is limited. Thermal-mechanical evaluations confirm there is significant margin to centerline fuel temperature or cladding strain acceptance criteria in Section 15.3.2. Limits on the Linear Heat Generation Rate (LHGR) are included each operating cycle ensuring the centerline fuel temperature and cladding strain acceptance criteria are met. The limits on LHGR are included in the Core Operating Limits Report (COLR).

The calculated $\Delta\text{CPR}/\text{ICPR}$ is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle to determine the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Sensitivity studies were performed on maximum FW pump flow (120% of rated), initial FW temperature reduction (-6°C), and FW controller settings. The sensitivity studies demonstrated no significant change in the event sequence or results.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there are no radiological consequences associated with this event.

15.5.3.2 Increase in Reactor Pressure AOO

15.5.3.2.1 Generator Load Rejection or Turbine Trip AOO

This event is in the Pressure Increase (PI) Group and is designated a BL-AOO event. The fault sequence name is Generator Load Rejection or Turbine Trip (LR-TT), and the Fault Sequence ID is PI-LR-TT_BL-AOO.

Postulated Initiating Event

The initiating event is either a generator load rejection or a turbine trip. TCVs have a fast closure function to protect the turbine during a generator load rejection. The TSVs close at a fast rate following a turbine trip. RPC remains unaffected and demands the TBVs open to control reactor pressure.

Sequence of Events

The fault sequence summary for the event:

- TCVs and/or TSVs close quickly causing pressure increase
- Anticipatory scram occurs on indication of a load rejection or turbine trip
- RPC opens TBVs to control pressure
- RLC maintains level
- Controlled state achieved

Table 15.5-7 lists the sequencing of events for this analysis.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-08 – Anticipatory Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a generator load rejection or turbine trip resulting in a fast closure of the TCVs or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated generator load rejection / turbine trip is presented on Figures 15.5-11 through 15.5-16, and the results are presented in Table 15.7.2-1. The results are shown for the case with the limiting CPR result. Automatic reactor scram occurs following indication of a generator load rejection or turbine trip. Pressure increases but is limited by the TBVs opening.

The core thermal power does not increase above the initial power and there is no concern in approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3. This event is not limiting and is not considered during LHGR limits development

The calculated $\Delta\text{CPR}/\text{ICPR}$ is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR Acceptance Criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there are no radiological consequences associated with this event.

15.5.3.2.2 Closure of One Main Steam Reactor Isolation Valve AOO

This event is in the Pressure Increase (PI) Group and is designated as a BL-AOO event. The fault sequence name is Closure of one Main Steam Reactor Isolation Valve (MSRIV) and the Fault Sequence ID is PI-1MSRIVC_BL-AOO.

Postulated Initiating Event

There are two main steam lines. The event is an inadvertent closure of one MSRIV that terminates flow in one of the main steam lines. A minimum MSRIV closure time results in the most severe event.

Postulated Event

The fault sequence summary for the event:

- One MSRIV closes causing RPV pressure and power to increase
- ATS scram occurs on MSRIV position
- RLC controls levels
- Second MSRIV in the second steam line closes on leak detection indication (this is assumed because it makes the event more severe)
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-8 lists the sequencing of events for the closure of one MSRIV.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-21 – Anticipatory Hydraulic Scram on MSRIV/MSIV Position
- DL2-31 – ICS Pressure Control on High Reactor Pressure

Core and System Performance

The event is simulated by initiating a closure of one MSRIV. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated closure of one MSRIV is presented on Figures 15.5-17 through 15.5-22 and the results are presented in Table 15.7.2-1. The results are shown for the case with the limiting CPR and reactor pressure response. The neutron flux and pressure increase resulting from the closure of one MSRIV are limited by an anticipatory scram on MSRIV position. The pressure increase is also initially limited because the MSRIV remains open in the second steam line. The second steam line is then assumed to close on main steam line break indication. This conservative assumption makes the event more severe. Pressure then increases and ICS is initiated on high pressure.

The core thermal power increase is not significant and there is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated $\Delta\text{CPR}/\text{ICPR}$ is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

This event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.3.2.3 Loss of Condenser Vacuum AOO

This event is in the Pressure Increase (PI) Group and is designated a BL-AOO event. The fault sequence name is Loss of Condenser Vacuum (LOCV) and the Fault Sequence ID is PI-LOCV_BL-AOO.

Postulated Initiating Event

There are a few potential causes of a loss of condenser vacuum including loss of one or more circulating water pumps. The loss of condenser vacuum results in a turbine trip. The turbine stop valves close at a fast rate following a turbine trip.

Sequence of Events

The fault sequence summary for the event:

- The TSVs close and main turbine trips on low main condenser vacuum causing pressure increase
- Anticipatory scram occurs on a turbine trip
- RPC opens TBVs to control pressure
- RLC maintains level
- TBVs close on high main condenser pressure and pressure increases slowly due to decay heat (the simulation is ended before TBV closure because the key mitigation DL functions are demonstrated, and a single ICS train is capable of controlling pressure and removing decay heat)
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-9 lists the sequencing of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-13 – Turbine Trip on High Main Condenser Pressure Setpoint 2
- DL2-08 – Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-14 – TBV Closure on High Main Condenser Pressure Setpoint 3
- DL2-31 – ICS Pressure Control on High Reactor Pressure

Core and System Performance

Input Parameters and Initial Conditions

The loss of condenser vacuum results in a turbine trip. The event is simulated by initiating a turbine trip resulting in a fast TSV closure. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

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The analysis is performed using an equilibrium core design. The event is run at BOC, MOC and EOR cycle exposure conditions.

Results

The simulated loss of condenser vacuum is presented on Figure 15.5-23 through 15.5-27 and the results are presented in Table 15.7.2-1. The results are shown for the case with the limiting CPR result. The pressure increases due to the fast TSV closure and is limited by the anticipatory scram on indication of a turbine trip. A scram on high main condenser pressure may occur sooner but conservatively not modeled. The pressure increase is also initially limited by the TBVs opening. Condenser vacuum loss is assumed to continue, resulting in TBVs closing. Once TBVs are closed, reactor pressure increases, and one ICS train initiates on high RPV pressure.

The core thermal power does not increase above the initial power and there is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3. This event is not limiting and is not considered during LHGR limits development.

The calculated $\Delta\text{CPR}/\text{ICPR}$ is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.3.2.4 Loss-of-Preferred Power AOO

This event is in the Pressure Increase (PI) Group and is designated a BL-AOO event. The fault sequence name is Loss-of-Preferred Power (LOPP) and the Fault Sequence ID is PI-LOPP_BL-AOO.

Postulated Initiating Event

A LOPP is initiated by offsite power supply failure. The loss of power results in the generator output breakers opening and the TCVs fast closure.

Sequence of Events

The fault sequence summary for the event is:

- LOPP occurs
- TCVs close quickly causing pressure increase
- Feedwater pumps lose power, FW pump discharge check valves maintain coolant inventory
- Circulating water pumps lose power
- Anticipatory scram occurs on generator load rejection
- RPC opens TBVs to control pressure

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- TBVs close on high main condenser pressure
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-10 lists the sequencing of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-01 – Maintain Target Pressure
- DL2-08 – Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-31 – ICS Pressure Control on High Reactor Pressure
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The LOPP results in the generator output breakers opening and a loss of power to the feedwater pumps. The event is simulated by initiating a FW trip and a load rejection resulting in fast TCVs closure. Anticipatory scram occurs on a generator load rejection. A scram on low bus voltage may occur sooner but is conservatively not modeled. This scram timing is the same as the anticipatory scram on a load rejection. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions. The results are shown for the case with the limiting result for CPR.

Results

The simulated LOPP is presented on Figure 15.5-28 through 15.5-33 and the results are presented in Table 15.7.2-1. The results are shown for the case with the limiting result for CPR. The pressure increase due to TCV closure is limited by the anticipatory scram on the generator load rejection. Scram on a low electric bus voltage may occur sooner but is conservatively not credited. The pressure increase is also initially limited by the TBVs opening. The TBVs later close on loss of power. Once the TBVs are closed, reactor pressure increases and ICS initiates on high RPV pressure. The ICS continues to limit the pressure increase.

The core thermal power increase is not significant and there is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated $\Delta\text{CPR}/\text{ICPR}$ is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.3.3 Decrease in Reactor Coolant Inventory AOOs

15.5.3.3.1 Feedwater Pump Trip – One Pump

The section analyzes the bounding BL-AOO event for the Inventory Reduction (IR) group. The fault sequence name is FW Pump Trip – One Pump (FWPT) and the Fault Sequence ID is IR-FWPT_BL-AOO.

Postulated Initiating Event

There is one FW pump normally operating and a second FW pump in standby. This event assumes a failure resulting in a trip of the operating FW pump. The RLC remains unaffected by the failure and increases the flow demand on the standby FW pump to maintain RPV water level.

Sequence of Events

The fault sequence summary for the event:

- One FW trips causing RPV water level decrease
- Standby FW pump starts and increases to rated FW flow
- Power decreases temporarily from a reduction in core flow and core inlet subcooling
- RPC controls pressure
- RLC maintains level
- RPV water low level scram and high-level FW isolation are avoided
- Controlled state achieved

Table 15.5-11 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-25 – Start Standby FW pump on Loss of Operating FW Pump
- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a FW pump trip. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The FW pump trip event is presented on Figures 15.5-34 through 15.5-39 and the results are presented in Table 15.7.2-1 for the exposure with the limiting CPR response. Reduction in FW flow results in a reduction of vessel inventory, causing the vessel water level to drop. The standby FW pump starts on confirmed low FW flow conditions, RPC throttles TCVs to control pressure, and the RLC increases the FW pump flow to rated conditions to maintain level. Low reactor water level (L3) scram is avoided.

The core thermal power does not increase and there is no concern for in approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3.2. This event is not limiting and is not considered during development of LHGR limits.

The calculated $\Delta\text{CPR}/\text{ICPR}$ is provided. The event is not limiting and is not considered in the OLMCPR development.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.3.4 Increase in Reactor Coolant Inventory AOOs

15.5.3.4.1 Inadvertent Isolation Condenser Initiation – One Train

This event is designated as a BL-AOO event. The fault sequence ID is II-IIIC-1_BL-AOO. This event assumes a failure causes a single Isolation Condenser (IC) condensate return valve to open. The event assumes that RLC remains unaffected by the failure and is able to maintain level. The event also assumes that RPC remains unaffected by the failure and is able to maintain pressure.

Postulated Initiating Event

The ICs are normally in standby mode. This event assumes spurious opening of a single ICS condensate return valve, resulting in the introduction of cold water into the reactor. The event assumes that RLC and RPC remain unaffected by the failure and are available to control reactor level and reactor pressure.

Sequence of Events

The fault sequence summary for the II-IIIC-1_BL-AOO event:

- ICS condensate return valve on one train opens
- Cold ICS condensate water drains into the chimney

- RLC maintains water level
- RPC maintains pressure
- Controlled state achieved

Table 15.5-12 lists the sequencing of events for the feedwater flow increase event.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

The credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

Input Parameters and Initial Conditions

The event is simulated by opening the ICS condensate return valve on one IC train. The initial conditions are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The inadvertent initiation of one IC train event is presented on Figures 15.5-40 through 15.5-45. Table 15.7.2-1 shows the limiting results for CPR response. When the IC condensate return valve is opened, cold water is introduced into the chimney region. After an initial small perturbation, the increased density in the chimney reduces core flow, water level, and power temporarily. After the initial reduction, core flow, level, and power increase. When the increase in level is sensed, the FW controller starts to demand the operating FW pump to reduce flow. After the initial surge, as condensate water drains into the chimney and IC flow reduces, the FW controller demands the operating FW pump to increase flow. The RPV water level, core flow, and core power settle back to their initial values. RPV pressure increases insignificantly. The level and pressure remain well within the RPV water level and RCPB pressure acceptance criteria in Subsection 15.3.2.

The core power increase is limited. There is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Subsection 15.3.2. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated $\Delta\text{CPR}/\text{ICPR}$ is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle to determine the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4 Analysis of Design Basis Accidents

This section evaluates the bounding BWRX-300 non-LOCA and LOCA PIEs. Subsections 15.5.4.1 through 15.5.4.5 describe the DSA non-LOCA DBAs, while Section 15.5.4.6 describes the DSA LOCAs inside containment. Subsection 15.5.9.2 describes the DSA for LOCAs outside containment.

15.5.4.1 Decrease in Reactor Coolant Temperature Event

15.5.4.1.1 Loss of All Feedwater Heating

This event is in the Temperature Decrease (TD) group and is designated CN-DBA event. The fault sequence name is CCF-LFWH, Passive Digital CCF DL2 Technology Platform (CCF-DL2), and the Fault Sequence ID is TD-CCF-LFWH_CCF-DL2_CN-DBA.

Postulated Initiating Event

A CCF results in the loss of all FW heating. Any CCF that results in the loss of all FW heating occurs gradually because of the thermal inertia inherent in the FW heaters. The FW temperature lowers to the main condenser temperature with the assumed time constant shown in Table 15.5-3.

Sequence of Events

- CCF results in the loss of all FW heating
- FW temperature decreases causing positive reactivity insertion
- RLC and RPC fail as-is
- Scram on high Simulated Thermal Power (STP) causing negative reactivity insertion
- RPV pressure decreases. The downcomer level decreases temporarily to lower than L3 because of void collapse
- Main steam isolation occurs on low RPV pressure
- RLC continues at initial flow causing RPV level to increase
- FW isolation occurs on high RPV level
- An ICS initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the two remaining ICS trains sufficiently controls pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-13 lists the sequence of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-23 – FW Isolation on High RPV Water Level

- DL3-17 – MSRV/MSCV Isolation on Low RPV Pressure
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by decreasing the FW temperature. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated loss of all FW heating event is presented on Figures 15.5-46 through 15.5-51, and the results are presented in Table 15.7.3-1 for the exposure with the limiting PCT response. The reduced FW temperature enters the core and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient, and a scram occurs on high STP. MSRV isolate on low RPV pressure.

FW flow remains at 100% due to the RLC CCF and RPV water level rises until FW isolates on high RPV water level. Decay heat causes RPV pressure to rise and an ICS train initiates. Only one ICS train is needed to prevent RPV pressure increase and maintain long-term cooling. A single failure of an ICS train starting on high RPV pressure does not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and the RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

This event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.2 Increase in Reactor Pressure Events

15.5.4.2.1 Generator Load Rejection or Turbine Trip

This event is in the Pressure Increase (PI) group and is designated a CN-DBA event. The fault sequence name is Generator Load Rejection or Turbine Trip (LR-TT), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is PI-LR-TT_CCF-DL2_CN-DBA.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in RPC and RLC failing as-is, which are continually operating, and failure of the anticipatory scram.

Sequence of Events

The fault sequence summary for the event is as follows:

- TCVs and/or TSVs close quickly causing pressure and power increase
- RLC fails as-is at initial condition, the TBVs remain closed, and anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, immediate challenge to cladding and RCPB integrity is over
- RPV pressure continues to increase because RPC fails as-is
- RPV level reduces due to the pressure increase
- One ICS train initiates on high pressure. First IC train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure
- With RLC failing as-is, initial feedwater flow continues causing RPV level to increase
- FW isolates on high RPV water level (L9)
- Controlled state achieved

Table 15.5-14 lists the sequencing of events, for the fast closure of TCV.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High neutron flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a fast closure of the TCVs or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. This event is limiting for PCT. Therefore, additional initial conditions and phenomena that impact PCT described in the TRACG Application (Reference 15.5-3) are applied in the limiting direction by at least one standard deviation consistent with the approach for an event with medium margin:

- Core void coefficient
- Channel interfacial shear
- Chimney interfacial shear
- Separator steam carry under

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- Critical quality used in boiling transition correlation
- Channel radial peaking factor
- Hot rod power
- Total initial power

Analysis is performed as needed to confirm the conservative PCT direction. Then a bounding case is created with all inputs in the conservative direction.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated generator load rejection / turbine trip is presented on Figures 15.5-52 through 15.5-57 and the results are presented in Table 15.7.3-1. The results are presented for the bounding case described above. Reactor scram occurs following high neutron flux. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. TBVs fail to open; however, ICS limits the pressure increase. Only one ICS train is needed to prevent pressure increase and maintain long-term cooling. A single failure of an ICS train to start on high pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria, provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Applying these conservatisms have little effect on peak pressure because peak pressure is primarily driven by the IC initiation setpoint. Once the pressure setpoint is reached, ICS initiates, and pressure rapidly reduces. For PCT, these uncertainties result in a PCT 91°C (163°F) higher than the base case with conservatisms similar to other DBA analyses. The results are within the acceptance criteria.

Barrier Performance

The effect of this event does not result in any challenge to the temperature or pressure transient derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.2.2 Loss-of-Preferred Power

This event is in the Pressure Increase (PI) group and is designated a CN-DBA event. The fault sequence name is LOPP, and the Fault Sequence ID is PI-LOPP_CCF-DL2_CN-DBA.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in RPC and RLC failing as-is at the initial condition, which are continually operating, and failure of the anticipatory scram.

Sequence of Events

The fault sequence summary for the LOPP event:

- TCV closes slowly due to loss of turbine control hydraulic pumps
- FW pumps lose power and coast down
- RLC fails as-is at the initial condition, TBVs remain closed, and the anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, immediate challenge to cladding and RCPB integrity is over
- RPV pressure continues to increase because TBVs remain closed
- An ICS train initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the remaining two trains is sufficient to control pressure
- Controlled state achieved

Table 15.5-15 lists the sequencing of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-12 – ICS Train 2 initiation on High RPV Pressure
- DL3-39 – FW Isolation on Loss of Normal FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a slow closure of the TCVs and feedwater pump trip. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated LOPP is presented on Figures 15.5-58 through 15.5-63 and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. Reactor scram occurs following high neutron flux. Bypass valves fail to open; however, the pressure increase is limited by the ICS initiation. Only one ICS train is needed to prevent pressure increase and maintain long-term cooling; therefore, a single failure of an ICS train to start on high pressure does not affect the event mitigation. The closure of the TCVs in the LOPP AOO is due to DL2 active mitigation. In the LOPP CN-DBA sequence, the TCVs close as a result of the PIE. If there is no power to maintain hydraulic pressure, the valves slowly close.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient challenge to the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.2.3 RPV Pressure Control Downscale

This event is in the Pressure Increase (PI) group and is designated a CN-DBA event. The fault sequence name is CCF – RPV Pressure Control Downscale (CCF-RPCD), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is PI-CCF-RPCD_CCF-DL2_CN-DBA.

Postulated Initiating Event

The PIE is a spurious CCF of the RPV pressure control. This failure results in a demand to close the TCVs (normal servo closure). This PIE also prevents the TBVs from opening. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in the RLC failing as-is at initial conditions, and failure of the anticipatory scram.

Sequence of Events

The fault sequence summary for the RPCD event:

- RPC demands TCVs to slow close and the TBVs remain closed
- RLC fails as-is at the initial condition and the anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, immediate challenge to cladding and RCPB integrity is over
- With RLC failing as-is at the initial condition, the initial feedwater flow continues causing RPV level to increase
- FW isolates on high RPV water level
- One ICS train initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the two remaining ICS trains is sufficient to control pressure
- Controlled state achieved

Table 15.5-16 lists the sequencing of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a slow closure of the TCVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated RPV Pressure Control Downscale is presented on Figures 15.5-64 through 15.5-69 and the results are presented in Table 15.7.3-1. The results are shown with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. Reactor scram occurs following high neutron flux. TBVs fail to open, and feedwater pump trips on high RPV level (L9). Natural circulation continues at a rate consistent with decay heat power. Only one ICS train is needed to prevent pressure increase and maintain long-term cooling; therefore, a single failure of an ICS train to start on high pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.2.4 Closure of All Main Steam Reactor Isolation Valves and FW Isolation Valves

This event is in the Pressure Increase (PI) group and is designated a CN-DBA event. The fault sequence name is CCF-Closure of All MSRIVs and FW isolation valves (CCF-DL4a-MSRIVC-FWIV) and the Fault Sequence ID is PI-CCF-DL4a-MSRIVC-FWIV_CN-DBA.

Postulated Initiating Event

The PIE is a spurious CCF DL4a function that affects all MSRIV and FW isolation valves.

Sequence of Events

The fault sequence summary for this MSRIV and FWIV closure event:

- Closure of all MSRIV and FW isolation valves
- Scram occurs on high neutron flux
- After scram, immediate challenge to cladding and RCPB integrity is over
- An ICS train initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure
- Controlled state achieved

Table 15.5-17 lists the sequence of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 - Hydraulic Scram on High neutron flux
- DL3-12 – ICS Train 2 initiation on High RPV pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a closure of the MSRIV and FW isolation valves. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated closure of MSRIVs and FW isolation valves is presented on Figures 15.5-70 through 15.5-75 and the results are presented in Table 15.7.3-1. The results are shown for the limiting case for cladding temperature and reactor pressure response. Reactor scram occurs following high neutron flux. The pressure increase is limited by ICS initiation. Only one ICS train is needed to prevent pressure increase and maintain long-term cooling. A single failure of an ICS train to start on high pressure does not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient that challenges the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there are no radiological consequences associated with this event.

15.5.4.3 Reactivity and Power Distribution Anomalies (DBA)

This event group is in the Reactivity Increase (RI) group.

15.5.4.3.1 Fuel Loading Error Event

The event is designated as a CN-DBA event. The fault sequence name is Fuel Loading Error (FLE) and the Fault Sequence ID is RI-FLE_CN-DBA.

Both the mislocated and the misoriented fuel bundle events are referred to as the fuel loading error event. For a fuel loading error event, it is assumed that the improper loading of a fuel assembly is not discovered and corrected through the core verification program, and the plant is operated throughout the operating cycle assuming that the design core configuration has been correctly implemented.

Mislocated Fuel Bundle

The mislocated fuel bundle error involves the loading error of two fuel bundles-the misloaded bundle and the bundle that belongs in that location. The scenario includes: 1) one location loaded with a bundle that operates at a lower power than planned and 2) another location with a bundle operating at a higher power than planned.

Three errors must occur for the mislocated event to take place:

1. Bundle is misloaded into a wrong core position.
2. Fuel bundle is loaded in the location where the first mislocated bundle occurred is overlooked, resulting in another mislocated bundle.
3. Both misplaced bundles are overlooked during the core verification performed following core loading.

The misoriented fuel bundle error involves the loading error of one fuel bundle to be rotated 180 degrees from the intended orientation.

For the misorientation event, two errors must take place:

1. The assembly is rotated while being lowered into position.
2. The misoriented bundle is overlooked during the core verification performed following core loading.

It is assumed that the FLE is not detected and that fuel rods operate above the thermal and mechanical limits. The potential exists that one or more fuel rods experience cladding failure. If this were to occur, the adverse consequences are detectable and can be suppressed during operation similar to leaking fuel rods resulting from other failure mechanisms. For the FLE, the initial adverse consequences consist of perforation of a small number of fuel rods in the assembly. Any perforations in the fuel cladding that occurs is localized and does not propagate to other

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assemblies. A control rod inserted in the vicinity of the leaking fuel rod(s) suppresses the power in the leaking fuel rod(s) and reduces the fission product release and offgas activity.

The FLE may exceed the operating mechanical LHGR limit because it may have worse peaking than the normally-loaded bundle. If the FLE operates above the operating mechanical LHGR limit, one or more rods may approach the design limit and experience cladding failure. If this occurs, the adverse consequences are the perforation of a small number of fuel rods in the misplaced bundle. The subsequent release of fission products to the reactor coolant is detected by the offgas system and processed accordingly.

Sequence of Events

- Multiple layers of independent administrative controls barrier due to human actions fail
- Startup occurs with a bundle rotated or with two bundles swapped locations
- Bundle swap or rotation results in potential for limited fuel cladding failures
- Rod failures (if they occur) occur over an extended period of time
- Elevated radiation in reactor coolant

Identification of Operator Actions

No operator action is credited for mitigating this event.

Systems Operation

No mitigation functions are credited in this event.

Core and System Performance

Input Parameters and Initial Conditions

The core operates at normal conditions during the cycle.

Results

It is possible that operators recognize the mislocated or misoriented bundle during startup or initial core operation using operating procedures because local monitoring indicates the reactivity anomaly. This allows the reactor operators to mitigate the event, which is not credited in the analysis. It is also possible that the mislocated or misoriented bundles do not result in significant local reactivity changes or fuel cladding failures.

If there is a significant increase in local reactivity, then the fuel could operate above LHGR limits or enter boiling transition. Analysis shows that the possibility of entering boiling transition could occur in relatively hot rods and in one of the mislocated or misoriented bundles and potentially the four face adjacent bundles. The mislocated bundle bounds the misoriented bundle for the potential to increase the local reactivity and result in failed fuel rods.

Even if some rods did enter boiling transition or operate above the LHGR limits, the rod failures occur over an extended time (weeks or months) or fail intermittently. Rod failures may be recognized as essentially fuel leakers. Operators would observe the increase in offgas system activity and take appropriate actions. Once recognized, procedures require the operators to locate the failed rods and, once located, insert a control rod in the location to suppress the power. If power suppression is not successful, the reactor is shutdown. Neither the slow developing fuel failures nor the operator actions are credited in the radiological analysis.

Barrier Performance

The reactor remains at normal operating pressure throughout the event. There is no challenge to the RPV and primary systems such as process barrier stress limitations.

The containment also remains at normal operating pressure and temperature. Containment integrity is not challenged.

As discussed above, there is potential for fuel failure. Analysis of potential mislocated bundles indicates that the number of fuel rod failures is bounded by assuming all the fuel rods in five bundles fail.

Radiological Consequences

A conservative approach is used to evaluate the radiological consequences. Instead of one or two rods failing, the event is analyzed for all the fuel rods in a mislocated or misoriented fuel assembly, and all the rods in the mislocated bundle and the four adjacent fuel assemblies experience instantaneous failure during normal operation.

To further assure that the fuel bundles contain the maximum fission product release, all five bundles (array independent) are multiplied by:

1. A factor of 1.4 to account for variations in fission product inventory over the operational cycle
2. A second factor of 2.5 to account for variations in cycle-dependent bundle power as a ratio to the end of cycle average bundle power

This results in a total factor of $1.4 \times 2.5 = 3.5$ to bound the bundle end of cycle inventory.

The radiological consequences of failing all of the fuel rods in five fuel bundles have been analyzed to provide results for no isolation trip with the release treated by an augmented offgas system.

The result for this analysis is presented in Table 15.7.3-2.

The radiological consequence of failing all fuel rods in five fuel bundles was analyzed for the case where the plant does not have a main steam line high radiation isolation trip. With no automatic MSIV closure, the activity is transported to an offgas system. The activity release to the environment would occur from the normal offgas release point after holdup in the offgas treatment system.

The results listed in Table 15.7.3-2 demonstrate that the radiological consequence do not exceed the offsite dose requirements for DBAs. The fuel loading error event is not a limiting event for DBA criteria for RCPB or containment integrity because the reactor remains at normal operating pressure throughout the event. There is no challenge to the RPV and primary systems, such as process barrier stress limitations. Containment also remains at normal operating pressure and temperature.

15.5.4.4 Increase in Reactor Coolant Inventory DBAs

This event group is in the Inventory Increase (II) group.

15.5.4.4.1 Feedwater Flow Increase All Pumps

This event is designated as a CN-DBA event. The fault sequence name is Feedwater Flow Increase – All Pumps (CCF-FWFI), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is II-CCF_FWFI_CCF-DL2_CN-DBA.

Postulated Initiating Event

One FW pump is normally operating and a second FW pump in standby. The RLC adjusts the pump speed to adjust FW flow to maintain RPV water level. This event assumes a spurious CCF that causes both FW pumps to increase flow to maximum speed that results in the maximum FW flow. Although not possible, the increase in flow is assumed to occur instantaneously.

Sequence of Events

The fault sequence event summary:

- Both FW pumps increase to maximum flow causing RPV level increase
- RPC remains as-is at initial condition
- Level, pressure, and power increase
- Automatic FW isolation on high RPV level
- Scram on high simulated thermal power
- RPV pressure and level decrease
- Main steam isolation on low RPV pressure
- An ICS train initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure and remove decay heat as demonstrated in the pressure increase DBA analysis (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated)
- Controlled state achieved

Table 15.5-18 lists the sequencing of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

The credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-17 – MSRIV/MS CIV Isolation on Low RPV Pressure
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by increasing both feedwater pumps flow to the maximum speed, resulting in the maximum FW flow. The initial conditions and total maximum flow for both FW pumps is provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated maximum FW pumps flow event is presented on Figures 15.5-76 through 15.5-81, and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting PCT response results. The increase in FW flow causes reactor level, pressure, and power to increase as reactor pressure control and reactor level control are unavailable. FW isolation occurs on high reactor level. Scram occurs on high simulated thermal power. Pressure decreases until MSRV isolation initiation occurs on low pressure. Pressure then increases and one ICS train is initiated on high pressure. Only one ICS train is needed to prevent RPV pressure from increasing and maintaining long-term cooling. A single failure of an ICS train to start on high RPV pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the design criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.4.2 Inadvertent Isolation Condenser Initiation – All Trains

This event is designated as a CN-DBA event. The fault sequence name is Inadvertent Isolation Condenser Initiation – All Trains (CCF-DL4a-IIIC), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is II-CCF-IIIC_CCF-DL2_CN-DBA.

Postulated Initiating Event

The ICS is normally in standby mode. This event assumes a spurious CCF that causes all ICS condensate return valves to open, resulting in introducing cold water into the reactor.

Sequence of Events

The fault sequence event summary:

- All ICS condensate return valves open
- RLC and RPC fail as-is at the initial condition
- Cold ICS condensate water drains into the chimney
- Core flow, reactor pressure, and power decrease
- RPV water level increases due to RLC failing as-is
- FW isolation occurs on high RPV water level
- Scram occurs on low RPV pressure

- Main steam isolation occurs on low RPV pressure
- Controlled state achieved

Table 15.5-19 lists the sequence of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

The credited DL3 functions:

- DL3-23 – FW Isolation on High RPV Water Level
- DL3-02 – Hydraulic Scram on Low RPV Pressure
- DL3-17 – MSRV/MSIV Isolation on Low RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by opening all ICS condensate return valves in one second. The initial conditions are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The inadvertent initiation of all IC trains event is presented on Figures 15.5-82 through 15.5-87, and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting PCT response results. When the IC condensate return valves are opened, cold water is introduced into the chimney region, reducing reactor power, reactor pressure, and core flow. Reactor scram and MSRV isolation initiation occurs on low pressure. After an initial reduction in reactor level, level begins to rise as reactor level control is unavailable. FW isolation occurs on high reactor level.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs due to metal water reaction.

Sensitivity studies were performed at the ICS minimum initial temperature in Table 15.5-3 and for an assumed increase in ICS return line volume of 50%. There was no significant change in the event sequence or results. Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.5 Decrease in Reactor Coolant Inventory – DBAs

This event group is in the Inventory Reduction (IR) group.

15.5.4.5.1 Loss of Feedwater Flow

This event is designated as a CN-DBA event. The fault sequence name is CCF Loss of FW Flow (CCF-LOFW), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is IR-CCF-LOFW _CCF-DL2_CN-DBA.

Postulated Initiating Event

The event sequence assumes a spurious CCF causes the loss of all FW flow and a passive CCF of the DL2 function results in a fails as-is RPC.

Sequence of Events

The fault sequence event summary:

- Loss of FW flow causes RPV level and power to decrease
- RPV pressure decreases due to frozen RPC
- Scram occurs on low RPV level
- FW isolation on a loss of normal FW flow indication
- Main steam isolation on low RPV pressure
- All ICS trains initiate on low level
- Controlled state achieved

Table 15.5-20 lists the sequencing of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-03 – Hydraulic Scram on Low RPV Level
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-14 – ICS Initiation on Low RPV Water Level
- DL3-39 – FW Isolation on Loss of Normal FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The event is conservatively simulated by initiating a trip of all FW pumps. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

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The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The loss of FW flow is presented on Figures 15.5-88 through 15.5-93 and the results are presented in Table 15.7.3-1 for the exposure with the limiting PCT response. The trip of all FW pumps results in a reduction of vessel inventory, causing the pressure and vessel water level to drop. Reactor scram occurs on low RPV level. FW isolates on loss of normal FW flow. The MSRIV close on low pressure. RPV water level continues to decrease until ICS initiates. Three ICS trains are modeled to open. Only one ICS train is needed to prevent RPV pressure increase and maintain long-term cooling. A single failure of an ICS train to start on low RPV level will not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs from metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

This event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.5.2 Reactor Pressure Vessel Pressure Control Open

This event is designated as a CN-DBA event. The fault sequence name is RPV Pressure Control Open (CCF-RPCO), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is IR-CCF-RPCO_CCF-DL2_CN-DBA.

Postulated Initiating Event

The event assumes all TCVs and TBVs are fully opened by a spurious RPC CCF. The event sequence assumes a passive CCF of the DL2 functions resulting in the RLC failing as-is.

Sequence of Events

The fault sequence event summary:

- All TCVs and TBVs open causing RPV pressure and power to decrease
- FW flow remains at 100% due to DL2 CCF
- Reactor scram and main steam isolation on low RPV pressure
- FW isolates on high RPV level

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- An ICS train initiates on high pressure. The first IC train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure and removes decay heat as demonstrated in the pressure increase DBA analysis (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated)
- Controlled state achieved

Table 15.5-21 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-02 – Hydraulic Scram on Low RPV Pressure
- DL3-17 – MSRIV/MS CIV Isolation on Low RPV Pressure
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by fully opening all TCVs and TBVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated RPV Pressure Control Open event is presented on Figures 15.5-94 through 15.5-99 and the results are presented in Table 15.7.3-1 for the exposure with the limiting PCT response. The opening of the TCVs and TBVs results in a decrease in reactor pressure causing voids to increase and power to decrease. FW remains at 100% rated flow due to the RLC CCF. The reactor scrams and main steam isolates on low RPV pressure. RPV water level rises until FW isolates on high RPV water level. RPV pressure then rises due to decay heat. An ICS train initiates on high RPV pressure after the first IC train fails to actuate (assumed single failure). One ICS train sufficiently controls pressure.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there are no radiological consequence associated with this event.

15.5.4.6 Loss-of-Coolant Accidents – DBAs

The scenarios for LOCA developed in Section 15.2.4 bound the CN-DBA sequences, demonstrating the fuel and containment integrity acceptance criteria are met for at least 72 hours using only passive heat removal systems.

The LOCA method used in containment analyses is described in NEDC-33922P BWRX-300 Containment Methods (Reference 15.5-2). The initial conditions and the modeling parameters are biased to account for uncertainties. The Defense Lines (DL) credited in the conservative LOCA breaks inside containment are identified in Table 15.5-49.

Meeting the acceptance criteria for fuel integrity is demonstrated by showing that level does not fall below the TAF, or fuel cladding temperature does not exceed the fuel cladding temperature during normal operating conditions.

As discussed in Section 15.2.4, all large breaks are isolated rapidly (10 seconds). Therefore, RPV inventory loss does not threaten fuel integrity in a large break LOCA. After RPV isolation, decay heat is removed by the isolation condensers directly from the RPV. The limiting parameters for large break LOCA events is containment pressure and temperature. Containment peak pressure reaches its peak value at approximately the time of RPV isolation.

The LOCA analyses demonstrate the core remains covered or fuel cladding temperature remains below the normal operating temperature for at least 72 hours using conservative assumptions for unisolated small break LOCAs. Therefore, fuel cladding temperature remains well below the fuel acceptance criteria, oxidation does not occur, and there is no hydrogen generation from cladding oxidation.

15.5.4.6.1 Main Steam Pipe Break Inside Containment, Conservative Case

Postulated Initiating Event

A break in the main steam pipe occurs at an arbitrary location between the flow limiter, which is placed as close as possible to the RPV and the containment penetration. The most limiting break is the double-ended instantaneous guillotine break of the main steam pipe.

The break flow occurs from both ends of the break. Break flow from the RPV side of the break is choked at the flow limiter. The TSV/TCV is conservatively assumed to close rapidly because this results in retaining more energy in containment. Steam flows from the RPV to the closed TSV/TCV through the intact steam pipe and to the break location in the reverse direction to the normal flow through the broken steam pipe, discharging through the turbine side of the break. Steam flow from the RPV to the intact loop is also choked at the flow limiters. CIV closure is conservatively not credited, and the entire steam line volume inventory contributes to containment pressurization.

Sequence of Events

The bounding scenario analyzed:

- Double-ended guillotine rupture of main steam line break inside containment concurrent with LOPP
- FW pump trip and coast down
- TSV or TCV start closing rapidly
- Scram initiated from high containment pressure
- Control rods start to insert
- ICS condensate return valve starts opening
- CUW stops
- Controls rods inserted sufficiently to diminish fission from prompt neutrons
- RIVs fully close
- Condensate return valve for one ICS train is fully open
- Peak containment pressure is reached
- Containment pressure starts decreasing

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

As shown on Figure 15.5-102, break flow from the turbine side is generally higher than the break flow from the RPV side of the break because of the pipe inventory, and the break flow continues even after the RIVs are fully closed at 10 seconds.

Feedwater pumps are assumed to trip concurrent with the break initiation because the bounding scenario assumes LOPP concurrent with the pipe break. FW pumps are assumed to trip concurrent with the break initiation since the bounding scenario assumes LOPP concurrent with the break. One isolation condenser is started when the high containment pressure setpoint is reached, which occurs within 1 second. Although two isolation condensers are available, only one isolation condenser is credited to bound the large break case in the isolation condenser steam supply pipe. Figure 15.5-100 shows the heat removal rate of one IC exceeds the power generation decay heat after 20 seconds. As a result, reactor pressure decreases rapidly even after the break isolation shown on Figure 15.5-101. RPV level, labeled as "Collapsed Downcomer Level" on Figure 15.5-103 stabilizes well above the TAF in 3 hours. The decrease in the downcomer level during the first 3 hours is due to the gradual decrease in the void fraction in the core and chimney, not due to RPV water inventory losses. There is no RPV water inventory loss after the RIVs close. Fuel never heats up because the core remains covered throughout the event. After the fission power is diminished, fuel cladding temperature remains near saturation temperature.

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Figure 15.5-104 shows the containment pressure in response to a large main steam pipe break inside containment. The break location is assumed away from the containment shell at the lowest main steam pipe elevation and directed upwards as discussed in LTR NEDC-33922P-A. This configuration maximizes containment pressure. The break location is assumed next to the containment shell and directed towards the containment shell in calculating the shell temperature on Figure 15.5-105 as discussed in LTR NEDC-33922P-A. This maximizes shell temperature.

The initial containment pressure includes a bias to account for uncertainties and is assumed at the containment high pressure setpoint for scram, reactor isolation and isolation condenser initiation. Although it appears that the setpoint is reached as soon as the break occurs, this is an artifact of the conservative initial condition assumption. A finite amount of time would have to elapse for containment pressure to reach the setpoint if the containment initial pressure is at the nominal pressure in normal operation. The pressure trend on Figure 15.5-104 shows that the containment pressure increases by 18.5 kPa in less than 1 second, indicating that the containment high pressure setpoint is reached in less than 1 second when the initial containment pressure is at the nominal pressure in normal operation. This confirms the break flow calculation assumption that containment high pressure setpoint will be reached in less than 1 second.

The initial reactor power in the conservative case calculations is 102% of the rated power to account for the power uncertainty. However, hot shutdown conditions may be more limiting for the mass release from the break because the initial RPV void fraction is lower resulting in higher liquid carryover to the break location. This also causes the break flow enthalpy to be lower. Both the rated initial conditions and hot shutdown initial conditions are analyzed. Figure 15.5-104 shows that the rated initial condition is more limiting for containment peak pressure.

PCCS does not require actuation, it is always in service and rejects heat to the equipment pool that is connected to the reactor cavity pool during normal operation. The calculations assume there is no heat loss from the containment shell to the concrete as discussed in the containment method LTR NEDC-33922P-A. Heat removal from containment atmosphere is by the containment shell heating up, by PCCS and through the containment dome to the pool. After RIV closure for large breaks, the only energy addition to the containment is due to the heat transfer from the RPV and hot piping walls. Because only one isolation condenser is sufficient to depressurize the RPV rapidly, heat load to containment also becomes small in the long term.

Core, System and Barrier Performance

Results

Event timing is summarized in Table 15.5-23. Key results are summarized in Table 15.7.3-3. The plots for RPV parameters and containment parameters are shown on Figures 15.5-100 through 15.5-105. The peak pressure is less than the design pressure with more than 20% margin. Containment shell temperature is also well below the containment shell design temperature of 166 °C. Peak accident pressure is approximately 322 kPaG (423 kPa) and half of the peak pressure is approximately 161 kPaG (262 kPa). As shown on Figure 15.5-104 **Error! Reference source not found.**, containment is depressurized rapidly, and containment pressure is reduced to 185 kPa at 6 hours. This meets the acceptance criterion for the containment response to pipe breaks that the containment pressure should be reduced to less than half of the peak pressure in 24 hours.

Barrier Performance

There is no fuel damage as a result of an MSLB inside containment. The only activity available for release is normal reactor coolant concentration in the vessel and piping prior to the break.

Radiological Consequences

The radiological consequences for a MSLB inside containment are bounded by the consequences for MSLB outside containment presented in Subsection 15.5.9.2.1.

15.5.4.6.2 Feedwater Pipe Break Inside Containment, Conservative Case

Postulated Initiating Event

A double-ended guillotine break occurs in the larger diameter segment of one feedwater pipe. This is more limiting than a break occurring in the smaller diameter FW pipe segments closer to the RPV. The bounding scenario is the same as that described in Subsection 15.5.4.6.1 for Main Steam Pipe Break Inside Containment

Sequence of Events

The bounding scenario analyzed:

- Double-ended guillotine rupture of the FW pipe break inside containment concurrent with LOPP
- FW trip and coast down
- TSV and TCV start closing rapidly
- Scram initiated from high containment pressure
- Control rods start inserting on scram initiation
- Condensate return valves on two ICS trains start opening
- Control rods are inserted sufficiently to diminish fission from prompt neutrons
- FWRIVs and CIVs are fully closed
- Peak containment pressure is reached
- IC valves are fully open

Identification of Operator Actions

No operator action is required to mitigate the event.

System Operation

RPV and containment response to a large FW break is similar to the RPV and containment response to a large main steam pipe break discussed in Subsection 15.5.4.6.1. Containment pressure reaches the containment high pressure setpoint for scram, reactor and containment isolation, and isolation condenser initiation in less than 1 second. Power generated by prompt fission is diminished in 3 seconds after the break. Condensate return valves in two of the three IC trains start opening in 1 second and fully open in 11 seconds. As shown on Figure 15.5-170, heat removal rate is much larger than the decay heat. As a result (shown on Figure 15.5-171), RPV pressure decreases much faster than the main steam pipe break case. Reactor water level is shown on Figure 15.5-172. The indicated water level stabilizes above the actual collapsed downcomer level. This is because the wide range level is off scale when the actual level falls below the lower tap and no longer indicates level. The actual collapsed downcomer level stabilizes well above TAF.

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Break flow from the pump side decreases initially because the break location is far away from the pump and the enthalpy becomes saturated locally right after the break although the pump is still coasting down as shown on Figure 15.5-173. The break flow from the pump side recovers and exceeds that of the RPV break side. This is because the pipe water inventory and the pump coasting down. The break flow becomes zero when the CIV is closed at 10 seconds. Enthalpy value after this point is not meaningful because there is no break flow.

As in the main steam pipe break cases, containment pressure on Figure 15.5-174 and temperature on Figure 15.5-175 are calculated for a break location maximizing pressure and temperature.

An additional feedwater pipe break case was included accounting for the lower initial feedwater temperature because the containment peak pressure may be higher if the FW pipe break occurs when the plant is operating at reduced FW temperature. Break flow rate is higher at higher subcooling. However, break flow enthalpy is also lower. Figure 15.5-174 shows the containment pressure for normal FW temperature and reduced FW temperature. Normal FW temperature results in a higher containment pressure. The peak pressure for both cases is bounded by the peak pressure resulting from a main steam pipe break.

Core, System and Barrier Performance

Results

Timing of events is summarized in Table 15.5-24. Plots for RPV and containment parameters are shown on Figures 15.5-170 through 15.5-175. Key results are summarized in Table 15.7.3-3 and show that the peak containment pressure and temperature resulting from feedwater pipe breaks are bounded by the main steam pipe breaks. Containment pressure and temperatures are less limiting than the main steam pipe cases and meet the acceptance criteria. Containment pressure calculated for FW pipe break at 6 hours is also less than half the peak containment pressure calculated for the main steam pipe case and decreasing.

Barrier Performance

There is no fuel damage as a result of an FWLB inside containment. The only activity available for release is normal reactor coolant concentration in the pipe prior to the break.

Radiological Consequences

The radiological consequences for a FWLB inside containment are bounded by the consequences for FWLB outside containment presented in Subsection 15.5.9.2.2.

15.5.4.6.3 Large Isolation Condenser Pipe Breaks Inside Containment

An isolation condenser system break larger than the area of a 19 mm equivalent diameter pipe is detected by the leakage detection for each ICS train separately. When a break is detected in one ICS train, both the steam supply pipe and the condensate return pipe of the affected ICS train are closed. The stroke time and delay time assumed for the isolation condenser isolation valves in the analysis are the same as those for all other RIVs and bound all other equipment initiation delays starting from the time of the pipe break. The other two unaffected isolation condensers are available to remove decay heat. For conservatism, the analysis assumes only one of the two remaining isolation condensers is put in-service on high containment pressure. Therefore, the number of isolation condensers available in this case is only one, which is the same as the number of isolation condensers available in the main steam pipe break cases as analyzed for all breaks larger than a 19 mm diameter.

Although the isolation condenser steam supply pipe diameter may be as large as the MS flow limiter diameter, the break flow rate from an isolation condenser steam supply pipe break is less than the break flow rate from the main steam pipe break. This is due to the much larger inventory in the main steam piping connected to both ends of the break.

The liquid in the isolation condenser is subcooled and does not contribute to high energy discharge from the break. Therefore, the main steam pipe break for containment response is more limiting than the isolation condenser steam supply pipe break.

ICS condensate return pipe diameter is much smaller than the FW pipe diameter used in the FWLB analysis. Therefore, large breaks in the isolation condenser condensate return pipe are bounded by large breaks in the FW pipe or MS pipe.

Because the isolation condenser pipe breaks are bounded by either the MS pipe or the FW pipe breaks, no further analysis of isolation condenser pipe breaks is needed.

15.5.4.6.4 Small Steam and Liquid Pipe Breaks Inside Containment

Postulated Initiating Event

A break area of ≤ 19 mm equivalent diameter remains unisolated. These breaks are analyzed for fuel integrity and containment integrity for at least 72 hours using conservative assumptions.

All liquid pipe break nozzles are at least 4 meters above TAF. A small pipe break on instrument lines may remain unisolated indefinitely.

A small liquid pipe break and a small steam pipe break have similar break flow rates after the level falls to 4 m above TAF. Because the isolation condensers depressurize the RPV, the break flow becomes very small in a few hours. Fuel heat-up does not occur even without injection to the RPV. Containment heat removal occurs through the PCCS to the equipment pool and through the containment head to the reactor cavity pool.

Sequence of Events – Small Steam and Liquid Pipe Breaks

The bounding scenario analyzed:

- Small steam pipe break concurrent with LOPP
- Pressure controller remains as-is. TCV remains at initial position, turbine chest pressure decreases rapidly
- FW pump trips and coasts down
- Main steam pipe low pressure setpoint is reached
- Reactor scrams
- MSRIVs are fully closed
- Level decreases to Level 2
- Condensate return valves on two ICS trains are fully open
- Peak containment pressure is reached

Identification of Operator Actions

No operator action is required to mitigate the event.

System Operation

The conservative cases assume LOPP concurrent with the pipe break, which is more limiting than the case where preferred power is available (discussed in Section 15.2). TCV/TSV closure is expected to occur as a consequence of the LOPP. However, TCV/TSV closure on LOPP is not credited. Rather, MSRV closure on low steam pipe pressure is conservatively credited in the analysis. The TCV/TSV back pressure is assumed to decrease rapidly maximizing the RPV water inventory loss to the turbine. Reactor scram also occurs on low steam line pressure accounting for the delays after the low steam pipe pressure is reached.

Isolation condenser condensate return valves start opening when the level falls to Level 2. As shown on Figure 15.5-115 and Figure 15.5-116, decay heat is removed by two isolation condensers. RPV depressurizes initially when the sum of the decay heat removal rate by the isolation condensers and the energy discharge from the break exceeds the decay heat power. Isolation condensers remove less power at lower pressure because of the lower temperature difference between the RPV steam and pool water. Reactor pressure stabilizes at a low value and the depressurization rate becomes very small. As shown on Figure 15.5-117, there is a rapid decrease in the collapsed downcomer level. This decrease is primarily due to the void collapse in the RPV. There are two small increases in level at approximately 69000 and 91000 seconds on Figure 15.5-117. These increases are due to void redistribution in the vessel. There is no increase in the RPV water inventory. As shown on Figure 15.5-117, downcomer collapsed level falls below TAF at 206000 seconds. However, the two-phase level in the core remains above TAF. Fuel remains wetted and thus never heats up.

The break mass and energy release are calculated assuming there is no back pressure. This assumption was made to bound breaks outside containment and accounts for the expected lower containment pressure than calculated because of the conservative assumptions used in the containment analyses. Even without break back pressure, Figure 15.5-119 shows that the break flow becomes very small in the long term.

Containment pressure calculated by using conservative assumptions and the small liquid pipe break flow without back pressure is shown on Figure 15.5-120. RPV pressure is also shown on the same figure. The calculated containment pressure increases to the RPV pressure at approximately 232800 seconds. Containment pressure is not higher than the RPV pressure because the break flow stops if the containment pressure becomes equal to the RPV pressure.

However, there is a potential that if ICS depressurizes the RPV faster than PCCS depressurizes containment in the absence of a break, reverse flow from containment to the RPV may occur. Non-condensables ingested into the RPV may collect in the isolation condensers and reduce their efficiency. Both the RPV and the containment could start repressurizing if back flow were to occur. In order to investigate this possibility, the containment pressure is calculated starting from 232800 seconds until the end of the 72 hour period assuming no break flow. The dashed line on Figure 15.5-118 shows that containment is rapidly depressurized in the absence of break flow. Because containment pressure decreases faster than the RPV pressure when there is no break flow, the RPV cannot depressurize below the containment pressure and reverse flow cannot occur. Energy released from the RPV through the break is a small fraction of the decay heat in the long term. A much larger fraction of the decay heat is removed by the isolation condensers. Therefore, RPV pressure calculated with and without a break are approximately the same at the time the RPV depressurizes to near containment pressure.

Figure 15.5-120 shows that containment pressure remains well below 262 kPa in the long term, which is 50% of the peak accident pressure calculated in Subsection 15.5.4.6.1.

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Figure 15.5-108 shows that level remains above TAF for small steam pipe breaks. Containment response shown on Figure 15.5-110 and Figure 15.5-111 for a small steam pipe break is similar to the containment response in Figure 15.5-120 and Figure 15.5-121 for a small liquid pipe break.

A small pipe break on an isolation condenser steam pipe does not cause a more limiting core or containment response than an instrument line break. The small breaks conservatively credit only two of the three isolation condensers even though a break of ≤ 19 mm equivalent diameter on an isolation condenser does not cause degradation in the isolation condenser. There is sufficient steam in the RPV to feed the condensation in the isolation condenser. Insufficient steam in the RPV to feed the isolation condenser only occurs if the RPV is depressurized to the point where almost all of the steam escapes the break. This is the case if the RPV pressure is lower than that calculated for an instrument pipe break. However, in this case, the break flow is also less than the break flow calculated for an instrument pipe break, resulting in the isolation condenser small break less limiting than the instrument pipe break. Breaks on isolation condenser steam pipes are no more limiting than a break on an instrument steam pipe break. This is because the break flow rates are the same for a small steam pipe break regardless of the break location, or the RPV pressure is too low to feed the isolation condenser.

Core, System and Barrier Performance

Results

For small steam pipe breaks, timing of events is summarized in Table 15.5-25. Key results are summarized in Table 15.7.3-3. The plots for RPV parameters and containment are shown on Figures 15.5-106 through 15.5-111.

All liquid pipe break nozzles are at least 4 meters above the TAF, and the timing of the events is summarized in Table 15.5-26. Key results are summarized in Table 15.7.3-3. The plots for RPV and containment parameters are shown on Figures 15.5-115 through 15.5-121. A small pipe break on the instrument lines may remain unisolated indefinitely. Since the isolation condensers depressurize the RPV, the break flow becomes very small in a few hours. Fuel heat-up does not occur even without injection to the RPV.

Barrier Performance

There is no fuel damage as a result of a small liquid and feedwater breaks inside containment. The only activity available for release is normal reactor coolant concentration in the pipe prior to the break.

Radiological Consequences

The radiological consequences for a small liquid or feedwater breaks inside containment are bounded by the consequences for small liquid and steam pipe breaks outside containment presented in Subsection 15.5.9.2.4.

15.5.5 Analysis of Design Extension Conditions Without Core Damage

The bounding transient event selection in Subsection 15.2.3 determines the list of DEC events evaluated in the following subsections. Tables 15.5-3 through 15.5-4 provide input parameters and initial conditions used in the DEC analyses. Table 15.5-5 provides the DLs used in the DEC analyses. The analysis is performed consistent with the DSA analysis approach described in Subsections 15.1.3 and 15.2.4.

15.5.5.1 Control Rod Drop Accident – Practically Eliminated

The BWRX-300 GNF-2 fuel has a core design that is similar to the BWR operating fleet.

The FMCRD uses a bayonet style coupling that requires a 45-degree rotation to uncouple. Since the FMCRD is firmly bolted into its position under the reactor vessel and the control rod is constrained from rotation by the fuel assemblies, it is not possible for the control rod to uncouple from the FMCRD during reactor operation. The hollow piston is the component within the FMCRD coupled to the control rod. The hollow piston normally rests on the ball nut internal to the FMCRD. There are dual FMCRD separation switches that sense that the hollow piston along with the associated control rod are resting on the ball nut. If the sensor detects that the hollow piston is no longer on the ball nut, then control rod withdrawal is blocked. Additionally, the hollow piston has latches that prevent inadvertent withdrawal of the assembly when not attached to the ball nut. This essentially limits possible separation such that it is not physically possible for a control rod drop accident involving a single control rod falling completely out of the core to occur.

Control rod ejection is prevented by physical constraints including the attachment of the control rod guide tube to the core plate and the CRD connection to the control rod guide tube. The FMCRD includes a brake that further prevents inadvertent rod withdrawal. The FMCRD also includes an internal ball check valve, which reduces the likelihood of rapid rod withdrawal. The ball check valve is a SC1 DL3 function because it prevents:

- Reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line
- The loss of pressure from the underside of the hollow piston
- The generation of loads on the drive that could cause a rapid rod withdrawal and associated reactivity insertion

Normal rod movement and the rod withdrawal rate are limited by the FMCRD. The rod control system controls rod patterns and provides control rod blocks limiting the rate and amount of reactivity addition for control rod movement.

The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows limited reactivity additions from rod withdrawal. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive design means. Control rod drop is prevented using a bayonet style coupling, CRD mechanism latches, and CRD separation switches. As a result, the CRDA and control rod ejection event have been practically eliminated.

15.5.5.2 Pressure Increase – DECs

15.5.5.2.1 Closure of One Main Steam Reactor Isolation Valve

This event is designated as an EX-DEC event. The fault sequence name is Closure of One MSRIV (1MSRIVC), and the Fault Sequence ID is PI-1MSRIVC_CCF-Hydraulic-Scram_EX-DEC.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. The analysis assumes a CCF hydraulic scram failure. The control rods enter the core using the CRDM run-in function. This event demonstrates that the CRDM run-in function performs the FSF control of reactivity without hydraulic scram.

Sequence of Events

The fault sequence event summary:

- One MSRIV closes causing pressure and power to increase
- Hydraulic scram signal on MSRIV position. Scram fails
- Hydraulic scram on any signal fails
- MSRIV in the second steam line closes on leak detection indication (assumed because it makes the event more severe)
- CRDM run-in initiation on high flux after scram signal
- All ICS trains initiate on high flux after scram signal
- FW pumps trip on high flux after scram signal
- Controlled state achieved

Table 15.5-27 lists the sequencing of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-21 – Anticipatory Hydraulic Scram Signal on MSRIV/MSIV Position (scram fails)
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

Credited DL4a functions:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a-41 – FW Pump/Condensate Pump Trip on High Flux after Scram Signal
- DL4a – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating closure of one MSRIV. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated closure of one MSRIV is presented on Figures 15.5-124 through 15.5-129 and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. The pressure increase is limited due to the initiation of ICS, FW pump trip and CRDM run-in that occur on high flux after scram signal (i.e., indications of high power level post scram initiation). The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The

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acceptance criteria are provided in Section 15.3. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

This event resulted in the highest peak cladding temperature and peak vessel pressure. Sensitivities are performed to examine cliff edge effects. Sensitivities on key initial conditions and phenomena that impact cladding temperature and peak vessel pressure described in the TRACG Application (Reference 15.5-3) are applied separately by at least one standard deviation:

- Core void coefficient
- Channel interfacial shear
- Chimney interfacial shear
- Separator steam carry under
- Critical quality used in boiling transition correlation
- Hot rod power

Results indicate no significant cliff edge effects. No excessive vessel pressure and no core damage occurs.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.2.2 Complex Sequence of Generator Load Rejection or Turbine Trip

This event is designated as an EX-DEC event. The fault sequence name is Complex Sequence Generator Load Rejection or Turbine Trip (LR-TT) plus CCF-Mechanical CRD, and the Fault Sequence ID is CSS-LR-TT_CCF-Mechanical-CRD_EX-DEC. This event demonstrates that with multiple failures to insert independent control rods with diverse motive forces combined with a very frequent PIE, that the remaining control rods perform the FSF reactivity control.

Postulated Initiating Event

The postulated initiating event is the same as for the LR-TT AOO event. Additionally, the event assumes that half of the control rods with the highest rod worth fail to scram and the CRDM run-in fails to insert the rods that failed to scram. No other failures are assumed.

Sequence of Events

The fault sequence event summary:

- TCVs and/or TSVs close quickly causing pressure and power increase
- Anticipatory scram occurs on indication of a turbine trip or load rejection signal, but half of the control rods fail to insert
- RPC opens TBVs to control pressure

- RLC maintains level
- Controlled state achieved

Table 15.5-28 lists the sequence of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-08 – Anticipatory Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a generator load rejection or turbine trip that results in a fast closure of the TCVs and/or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated generator load rejection / turbine trip is presented on Figures 15.5-130 through 15.5-135 and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting result for reactor pressure response. Automatic anticipatory reactor scram occurs following indication of a generator load rejection or turbine trip with half of the rods failing to insert. The neutron flux increases rapidly because of the void reduction caused by the pressure increase.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

A controlled state is achieved with steam bypassed to the main condenser and fed back into the reactor by feedwater. This is maintained for a significant amount of time as long as power is available. Given the conditions, operators initiate additional CRDM run-in signals or manually insert CRDM to insert the remaining control rods into the core. If operator actions are unsuccessful, operators inject boron to shut the reactor down. Another option available to the operators is to decrease power by reducing FW flow. With reduced FW flow, reactor water level decreases, reducing core flow and reducing reactivity through void reactivity feedback until the steam flow matches the FW flow.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.2.3 Loss of Condenser Vacuum with CCF Hydraulic Scram

This event is designated as an EX-DEC event. The fault sequence name Loss of Condenser Vacuum (LOCV), CCF-Hydraulic-Scram and the Fault Sequence ID is PI-LOCV_CCF-Hydraulic-Scram_EX-DEC. This event demonstrates that the CRDM run-in function performs the FSF controlling reactivity without the hydraulic scram.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. A CCF results in failure of hydraulic scram. The control rods enter the core using the CRDM run-in function. No other failures are assumed.

Sequence of Events

The fault sequence event summary:

- Loss of vacuum results in a turbine trip
- TSVs close quickly causing pressure and power increase
- Hydraulic scram signal on either turbine trip or high main condenser pressure scram fails
- Hydraulic scram fails on any signal
- CRDM run-in initiation occurs on high flux after scram signal
- RPC opens TBVs to control pressure
- RLC maintains level
- ICS initiates on high flux after scram signal
- FW pumps trip on high flux after scram signal
- TBVs close on high main condenser pressure
- Controlled state achieved

Table 15.5-29 lists the sequencing of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited Defense Line functions:

DL2:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

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- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-13 – Turbine Trip on High Main Condenser Pressure Setpoint 2
- DL2-14 – TBV Closure on High Main Condenser Pressure Setpoint 3
- DL2-08 – Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand (scram fails)

DL4a:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a-41 – FW Pump/Condensate Pump Trip on High Flux after Scram Signal
- DL4a – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

Core and System Performance

Input Parameters and Initial Conditions

The loss of condenser vacuum results in a turbine trip. The event is simulated by initiating a turbine trip that results in a fast closure of the TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated loss of condenser vacuum is presented on Figures 15.5-136 through 15.5-141 and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the pressure increase is initially limited by the TBVs opening. The pressure increase is limited due to the initiation of ICS, FW trip, and CRDM run-in that occur on high flux after scram signal.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.2.4 Loss-of-Preferred Power with CCF Hydraulic Scram

This event is designated as an EX-DEC event. The fault sequence name is Loss-of-Preferred Power (LOPP), CCF-Hydraulic-Scram and the Fault Sequence ID is PI-LOPP_CCF-Hydraulic-Scram_EX-DEC. This event demonstrates that the CRDM run-in function that performs the FSF controlling reactivity without the hydraulic scram.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. A CCF results in failure of hydraulic scram. The control rods enter the core using the CRDM run-in function. No other failures are postulated in the event.

Sequence of Events

The fault sequence event summary:

- LOPP results in generator output breakers opening
- TCVs close quickly
- Feedwater pumps lose power
- Hydraulic scram signal fails on either generator load rejection or low electric bus voltage
- Hydraulic scram fails on any signal
- CRDM run-in initiation on high flux after scram signal
- TBVs close on loss of power
- ICS initiates on high flux after scram signal
- Controlled state achieved

Table 15.5-30 lists the sequencing of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited Defense Line functions:

DL2:

- DL2-01 – Maintain Target Pressure
- DL2-08 – Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand (scram fails)
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

DL4a:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

Core and System Performance

Input Parameters and Initial Conditions

The Loss-of-Preferred Power results in the generator output breakers opening and a loss of power to the feedwater pumps. The event is simulated by initiating a FW pump trip and a load rejection that results in a fast closure of the TCVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated Loss-of-Preferred Power with CCF hydraulic scram failure is shown on Figures 15.5-142 through 15.5-147 and the results are presented in Table 15.7.3-1. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The rapid closure of the TCVs results in a pressure increase. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the pressure increase is initially limited by the opening of the TBVs. The TBVs later close, ICS initiates, and CRDMs run-in on high flux after scram signal. The ICS continues to limit the pressure increase.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.3 Reactivity and Power Distribution Anomalies – DEC

15.5.5.3.1 All Control Rod Withdrawal at Power (ACRW)

This event is designated as an EX-DEC event. The fault sequence name is CCF - All Control Rod Withdrawal at Power- All Rods (CCF-ACRW), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Fault Sequence ID is RI-CCF-ACRW_CCF-DL2_EX-DEC.

Postulated Initiating Event

All control rods inserted in the core start to withdraw due to rod control spurious CCF. A passive CCF of DL2 technology platform results in DL2 function failure.

Sequence of Events

The fault sequence event summary:

- All control rods start to withdraw resulting in a power increase
- ATLM (Automatic Thermal Limiting Monitor) and MRBM (Multi-Channel Rod Block Monitor) fail to block rod withdrawal
- RPC and RLC fail as-is at the initial condition
- Scram on STP power or neutron flux
- After scram, the immediate challenge to cladding and RCPB integrity is over
- RPV pressure decreases
- RPV level decreases temporarily due to the void collapse in the core and chimney
- Sensed level increases due to continuing FW flow and flashing in the downcomer
- FW isolation occurs on high RPV level
- Main steam isolation occurs on low RPV pressure and pressure slowly increases
- One ICS train initiates on high pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-31 lists the sequencing of events.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-17 – MSRV/MSRV Isolation on Low RPV pressure
- DL3-11 – ICS Train 1 initiation on High RPV pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by withdrawing all the control rods in the core using the initial conditions, plant parameters, and control rod speed specified in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC and MOC cycle exposure conditions. EOC exposure is not run because all control rods are fully withdrawn.

Results

The simulated ACRW is presented on Figure 15.5-148 through 15.5-153. The analysis results are presented in Table 15.7.3-1. The results are shown for the case with the limiting PCT response result.

When the control rods are withdrawn, the power increases and scram occurs on high simulated thermal power or neutron flux. The RPV water level increases and FW is isolated. RPV pressure decreases and the MSRIVs close. Eventually the RPV pressure increases, and one ICS train initiates. The pressure remains well within the DBA event RCPB pressure acceptance criteria in Subsection 15.3.1. There is no challenge to containment.

The long-term core cooling capability is assured by meeting the DBA acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed and there is no core damage.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.3.2 Inadvertent Single Control Rod Withdrawal at Power (ICRW) - DEC

This event is designated as an EX-DEC event. The fault sequence name is Inadvertent Control Rod Withdrawal at Power - single rod (ICRW), and the Fault Sequence ID is RI-ICRW_CCF_DL2_EX-DEC.

Postulated Initiating Event

A control rod inserted in the core is withdrawn due to a failure. A passive CCF of DL2 results in failure of the DL2 functions.

Sequence of Events

The fault sequence event summary:

- Single rod (with highest reactivity worth) starts to withdraw
- ATLM and MRBM fail to block the rod withdrawal
- RPC and RLC are assumed to function normally because this prolongs the event and makes it more severe for cladding temperature effects
- Reactor power increases but the scram level is not reached
- Local power and cladding temperature increase
- Power and the cladding temperature reach a stable level
- Operator action to initiate scram is expected due to the high power level (Not credited in analysis simulation as these DL functions demonstrate achieving and maintaining a controlled state)

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- After the scram, no credit is taken for RPC or RLC (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- RPV pressure decreases and the RPV level increases (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- FW isolation occurs on high RPV level (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- Main steam isolation occurs on low RPV pressure and pressure slowly increases (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- One ICS train initiates on high pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-32 lists the sequencing of events.

Identification of Operator Actions

No operator action is credited.

Systems Operation

Credited DL3 functions:

- DL3-17 – MSRV/MSIV isolation on Low RPV Pressure
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-11 – ICS Train 1 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by withdrawing a single control rod in the core using the initial conditions, plant parameters, and control rod speed specified in Tables 15.5-3 through 15.5-5. The initial CPR and the hot rod power are consistent with the reference core design.

The analysis is performed using an equilibrium core design. The event is run at BOC and MOC cycle exposure conditions. EOC exposure is not run because all control rods are fully withdrawn.

Results

The simulated ICRW event is presented on Figures 15.5-154 through 15.5-159. The analysis results are presented in Table 15.7.3-1. The results are shown for the case with the limiting PCT response result.

When the control rod is withdrawn, the power increases. The RPV level and pressure vary insignificantly because RLC and RPC function to maintain level and pressure preventing scram, thus maximizing the impact of fuel cladding temperature increase.

In this event, the PCT values are well below the DBA acceptance criteria. The long-term core cooling capability is assured because the effects are local. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria.

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Sensitivities are performed examining cliff edge effects. Key initial conditions are conservatively biased to cause transition boiling even though this does not occur at the nominal / realistic conditions associated with DEC conditions. With initial CPR conservatively biased low (by approximately 0.05), and the hot rod power (LHGR) conservatively biased high (approximately 30%), local high cladding temperatures occur in hot rods in a few high power bundles located near the control rod withdrawn in error. This result is expected in fuel cladding failure in a very limited number of rods. However, the fuel failures are localized, the core remains cooled, and no core damage occurs.

Barrier Performance

There is no challenge to the RCPB and containment. The fuel cladding may experience local failures if initial LHGR and CPR are more severe. The predicted number of rod failures is limited to high-powered fuel rods in a few high-powered bundles near the control rod withdrawn in error. However, the fuel failures are localized, the core remains cooled, and no core damage occurs.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.4 Decrease in Reactor Coolant Inventory – DEC

This event is in the IR group and is designated as an EX-DEC event. The fault sequence name is FW Isolation (CCF-FWI-DL3) and the Fault Sequence ID is IR-CCF-FWI-DL3_EX-DEC.

Postulated Initiating Event

The event sequence assumes a spurious CCF isolates all FW flow and a passive CCF of the DL3 functions.

Sequence of Events

The fault sequence event summary:

- FW flow isolation causes RPV water level and power to decrease
- RPC maintains RPV pressure
- Scram and MSRIV isolation on sustained low FW flow
- ICS pressure control initiates on high RPV pressure
- Controlled state achieved

Table 15.5-33 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited Defense Line functions:

DL2:

- DL2-01 – Maintains Target Pressure
- DL2-42 – Anticipatory Hydraulic Scram on Sustained Low FW Flow
- DL2-31 – ICS Pressure Control on High Reactor Pressure

DL4:

- DL4a -12 – MSRIV/MS CIV Isolation on Sustained Low FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a conservatively fast isolation of all FW flow. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The CCF FW isolation event is presented on Figures 15.5-160 through 15.5-165 and the results are presented in Table 15.7.3-1 for the exposure with the limiting PCT response. The loss of FW flow results in a reduction of vessel inventory, causing the power and RPV water level to decrease. RPC maintains reactor pressure. Reactor scram and main steam isolation occurs based on sustained low FW flow. ICS pressure control initiates based on high RPV pressure.

The long-term core cooling capability is conservatively assured by meeting the DBA acceptance criteria for fuel cladding and RCPB. The reactor integrity is conservatively assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the DBA acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs from metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.6 Analysis of Design Extension Conditions with Core Damage

The analysis of DEC with core damage are addressed in the Level 2 PSA described in Section 15.6.4.

15.5.7 Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Fuel Pool

The analysis fault sequences associated with the fuel pool are DEC analyzed in the Level 1 PSA described in Subsection 15.6.3.

15.5.8 Analysis of Fuel Handling Events

The Fuel Handling Accident (FHA) is categorized as a non-reactor group DEC event. The fault sequence ID is FHA_EX-DEC. The event occurs as a result of a failure of the fuel assembly lifting mechanism, resulting in the drop of a raised irradiated fuel assembly onto the reactor core or into the fuel pool. The dropped irradiated fuel assembly results in cladding failure in the dropped and impacted bundles. The sequence of events for the postulated FHA is provided in Table 15.5-34.

The dose results are conservatively compared to the criterion in Table 15.7.6-1 to demonstrate compliance to the DBA acceptance criterion.

Fuel Damage

Because of the complex nature of the impact and the resulting damage to fuel assembly components, predicting the number of failed rods is not possible. For this reason, a simplified energy approach is taken. Numerous conservative assumptions are made to assure that the number of failed rods is conservatively analyzed.

The key assumption for the FHA is that during a refueling operation, a fuel assembly is moved over the top of the reactor core or fuel pool. While the fuel grapple is in the over-hoist condition with the bottom of the assembly at the maximum height allowed when using the fuel handling equipment, the main hoist cable and a redundant cable fails. This results in the fuel assembly, the fuel grapple mast and head falling on top of the core impacting a group of four fuel assemblies. The grapple head and mast are fixed vertically to the dropped assembly so that all the kinetic energy is transferred through the dropped assembly to the group of impacted assemblies. The dropped assembly impacts the core at a slight angle and the rods in this assembly might bend. After the assembly impacts the core, the assembly, grapple head and mast fall onto the core horizontally without contacting the side of the pressure vessel or fuel pool.

During refueling operations, the bounding radiological event is a drop over the core due to the maximum drop height.

Fuel rod failure is assumed at 1% circumferential strain. The associated axial strain is $(0.01)/\nu$, where ν is Poisson's ratio, plastic deformation is assigned a value of 0.5, and the energy per rod failure is expressed by:

$$E_f = \sigma_y \times \epsilon \times V \quad (\text{Equation } E_f)$$

Where:

E_f = energy per rod failure

σ_y = yield stress

ϵ = axial cladding strain

V = volume of fuel cladding

Kinetic energy is calculated for the dropped fuel bundle that accounts for the influences of buoyancy and resistance from the reactor cavity pool water. Finite element analysis simulations are used to determine the kinetic energy based on the drop distance in air or water. The simulation results revealed that the drop distance of a fuel bundle in air is greater than 2.3 m (7.5 ft), while the kinetic energy of the fuel bundle drop in water is less than 50% of that in air. When the bundle drop height is 10.4 m (34.0 ft) the energy is approximately 22% of that in air. This analysis credits a 50% reduction in the kinetic energy of the dropped bundle although the limiting case drop height correlates to a larger reduction.

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The fuel assembly wet weight and the mast wet weight are used in applying the 50% kinetic energy reduction to the fuel assembly due to dropping through water that is expressed as:

$$E_1 = \frac{h_{\text{drop}} \times (W_{\text{fuel}} + W_{\text{mast}})}{2}$$

Where:

- E_1 = energy from initial drop
- W_{fuel} = weight of fuel bundle
- W_{mast} = weight of refueling mast
- h_{drop} = drop height

Substituting numerical values yields Equation E_1 :

It is assumed that half of the energy is absorbed by the cladding. The ratio of the cladding to the non-fuel mass for GNF2 fuel is 0.4997. The calculated yield strength using the methodology described above (see Equation E_1) is 37.503 kgf-m/rod (271.26 ft-lbf/rod). Therefore, the number of failed rods in the impacted assemblies from the initial drop is 31 rods.

Additional energy is generated in a secondary impact as the bundle falls over from a vertical orientation to a horizontal orientation, and damages additional rods in the impacted bundles. Incorporating the 50% reduction due to the kinetic energy in water is expressed as:

$$E_2 = 50\% \times (h_{\text{fuel}} \times W_{\text{mast}} + \frac{1}{2} \times h_{\text{fuel}} \times W_{\text{fuel}})$$

Where:

- E_2 = energy of dropped bundle and mast from secondary impact
- h_{fuel} = height of refueling mast

Fifty percent of the kinetic energy is absorbed by the impacted assemblies resulting in the number of failed rods in the impacted assemblies of 5 rods.

All the rods in the dropped assembly are assumed to fail and are full-length. GNF2 fuel assemblies contain both full-length and part-length rods accounting for the difference in lengths resulting in 85.6 effective full-length rods per bundle. The number of failed rods used in determining the radiological consequences is 128 failed fuel rods or 1.495 failed fuel bundles.

There are 240 fuel bundles in the BWRX-300 core. The fraction of the core damaged in an FHA is determined by:

$$\text{Core Damage Fraction} = \frac{1.495 \text{ bundles damaged}}{240 \text{ bundles in the core}} = 6.23\text{E-}03$$

Core Inventory of Isotopes

A BWRX-300 core inventory of isotopes is calculated using the Oak Ridge National Laboratory code ORIGEN2, version 2.1 and the BWRUE.LIB cross section library in units of Ci/MWth for the bulk operating parameters in Table 15.5-35.

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A subset of the more than 700 isotopes from this inventory are used to model DBA dose consequences. The 60 isotopes used for DBAs are the dominant contributors to immersion and inhalation doses from airborne activity released during a DBA. This set of nuclides consist of 54 isotopes identified in WASH-1400 (NUREG-75/014) and 6 isotopes identified in SAND-85-2575 (NUREG/CR-4467). This is the group of isotopes typically used for alternate source term (AST) dose evaluations.

The BWRX-300 reactor is subcritical for at least 24 hours prior to initiating refueling operations. The BWRX-300 core inventory of the 60 dose-significant isotopes after 24 hours of decay are shown in Table 15.5-36.

Gap Fractions

For events that are non-LOCAs where some fuel damage is postulated like the FHA, the fractions of the core inventory assumed in the fuel rod gap region for the various radionuclides are taken from USNRC Regulatory Guide (RG) 1.183 and reported in Table 15.5-37.

Radial Peaking Factor

The radioactive material available for release in an FHA is assumed in the analysis to come from assemblies with peak inventory. To simulate this assumption, the inventory is scaled up by the maximum power Radial Peaking Factor (RPF). This represents the maximum achievable operational power history immediately preceding shutdown. Based on prior experience with GNF2 cores, a representative RPF value of 1.70 is assumed for this analysis.

Activity Released from the Fuel

All particulate isotopes are retained by the water in the fuel pool or reactor cavity water. Thus, only the noble gases and the gaseous form of iodine are available to escape the water. The activity released from the fuel shown in Table 15.5-38. The release is the mathematical product of the core power (E_f), RPF, gap fractions (Table 15.5-37), and core damage fraction. With this information, the release from the fuel is calculated as shown in Table 15.5-38.

Pool Scrubbing (Decontamination)

Credit is taken for retention of some of the released fission gas in the water where it is released from the damaged fuel rods. Because the depth of water above reactor cavity pool and SFP is greater than 7.01 m (23.0 ft), the simple Decontamination Factor (DF) model from USNRC RG 1.183, Appendix B is used.

The activity released from the surface of the reactor cavity pool, shown in Table 15.5-39, is calculated by applying the RG 1.183 DFs to the activity released from the fuel in Table 15.5-38.

Transport in the Reactor Building

The radioactivity released from the reactor cavity pool is assumed to mix instantaneously with the free air volume of the refueling outage floor and crane area, which is the intermediate volume between the FHA release from the water and the environment. The refueling outage floor and crane area has an approximate air volume of 9,620.25 m³ (339,736 ft³) taking into account the equipment footprint.

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The analysis conservatively ignores mixing to account for the possibility of inadequate mixing over the short term. Therefore, crediting for mixing is reduced by 50% by reducing the volume of airspace. There is no mechanical means to ensure the refueling outage floor and crane area airspace is well mixed. To account for the 50% volume reduction of airspace:

$$\text{Credited Refueling Area Volume} = (50\%) \times 9620.25 \text{ m}^3 = 4,810.13 \text{ m}^3 (169,868 \text{ ft}^3)$$

This doubles the concentration of contamination in the leakage from the RB to the environment.

Release Assumptions

To implement this in the model, a single leakage rate from the reactivity cavity or fuel pool ensures the entire release is transported to the environment from the RB in two hours.

To simulate the leakage to the environment, the following equation is used to calculate a corresponding leakage rate.

This rate is calculated using the equation below, and setting to time (variable "t") to 120 minutes as follows:

$$\frac{C(t)}{C_{ss}} = e^{-Qt/V}$$

Where:

$C(t)$ = evacuation volume transported to the environment at time t

C_{ss} = evacuation volume at time t=0

$C(t)/C_{ss}$ = fraction of evacuation volume remaining at time t (unitless)

Q = constant flow rate out of space (m^3/s)

V = volume of air transported to the environment (m^3)

t = time duration of dilution period in minutes (minutes)

As $C(t)$ approaches steady state or complete transport of the evacuation volume (assume a value of 0.1 %), $C(t)/C_{ss}$ approaches a value of zero and the previous equation can be simplified to:

$$0.001 = e^{-Qt/V}$$

Substituting the known values and solving for Q:

$$-\ln(0.001) \times \frac{V}{t} = Q$$

The flow rate that transports 99.9% of the contamination released to the RB refueling outage floor and crane area can be transported to the environment in two hours at a flow rate of $267.9 \text{ m}^3/\text{min}$ ($9778.4 \text{ ft}^3/\text{min}$).

Dispersion in the Environment

Transport to the EAB and Automatic Action Zone (AAZ) from the BWRX-300 is simulated using atmospheric dispersion factors (χ/Q values) that are established for the Darlington site. Dispersion and deposition is determined in the ADDAM code based on 5 years of DNNP site-specific meteorological data. The meteorological data set used complies with the requirements of CSA N288.2:19. The DBA χ/Q s are set equal for the OPG 350 m Emergency Zone (EZ).

Dose Calculation

Because the inventory of isotopes released from the surface of the reactor cavity pool is already determined in Table 15.5-39. The release from the pool to the RB in Table 15.5-39 is modeled at time $t=0$.

Breathing Rates

The postulated FHA breathing rates used are consistent with CSA N288.2. The deterministic calculations conservatively used to demonstrate compliance with CSA N288.2 are in the 95th percentile of the breathing rates for the representative person.

Decay and Daughtering

This analysis assumes a decay time of 24 hours prior to the removal of spent fuel during refueling, and credit for this decay period is taken in the initial core inventory Table 15.5-36.

Decay and daughtering of nuclides during the FHA are assumed in the ADDAM model for the duration of the event (30 days).

Dose Conversion Factors

The 30-day committed whole-body dose is compared to the CNSC REGDOC-2.4.1 acceptance criteria. Consistent with REGDOC-2.4.1, the dose contributions include:

- External radiation from cloud and ground deposits
- Inhaled radioactive materials
- Skin absorption of tritium

The ADDAM code uses dose conversion factors that comply with CSA N288.2.

Results

The BWRX-300 FHA, which is classified as a DEC, meets the Darlington site DBA dose acceptance criteria with considerable margin for the most critical group as demonstrated in Table 15.7.6-1.

15.5.9 Analysis of Radioactive Releases from a Subsystem or a Component

15.5.9.1 Analysis of Liquid and Gaseous Radioactive Waste System Scenario Releases

The radioactive liquid tank and offgas system (OGS) failures are addressed as process system failures. The analysis for these releases is described in Chapter 11, Subsections 11.2.9 and 11.3.13 and Table 11.3-4.

Liquid Tank Failure

To prevent tank leakage from exiting the Radwaste Building (RWB), the collection and sample tanks are enclosed with a concrete wall barrier. If a tank failure occurs, the concrete wall containment area prevents any liquid radioactive waste from exiting the RWB. The contents of the tank drain to the nearest radioactive drain system, and any venting is collected in the buildings ventilation exhaust system and processed accordingly.

Offgas System Failure

The limiting BWRX-300 OGS failure is an inadvertent bypass of the charcoal delay beds. An evaluation was performed, and the resulting public dose consequences are well below the REGDOC-2.5.2, Section 4.2.1 limits as shown in Chapter 11, Table 11.3-4.

15.5.9.2 Analysis of Loss-of-Coolant Accidents Outside Containment

As discussed in Subsection 15.5.4.6, the scram and RPV isolation trips occur for the large breaks outside containment within the same time as breaks inside containment.

For large breaks, timing of the break detection is less than 1 s for breaks outside containment that is the same as the time to reach the drywell high pressure setpoint. Because reactor scram and isolation valve closures for breaks inside containment also occur for breaks outside containment, the event progression is no different for breaks outside containment than inside containment. For main steam pipe breaks, the break flow rate calculated for breaks inside containment is also used for breaks outside containment because the MSCIV closure is not credited in the main steam pipe break inside containment. For FW pipe breaks, the only difference between the pipe breaks inside and outside of containment is the closure of the FWCIV. For a FW pipe break outside containment, break flow includes flow from the reactor as well as the pipe inventory.

The isolation condenser pipe break outside containment is limited to the flow passing through the orifices in the steam distribution pipes. During normal operation prior to the break, condensate return valves are closed and remain closed. Isolation steam supply pipe has a guard pipe outside containment so that break flow in the supply pipe upstream of the orifice is not discharged outside containment. A break in the isolation condenser is followed by a discharge of the subcooled water in the isolation condenser into the pool, followed by steam break flow passing through the orifice in the steam supply pipe until the isolation condenser RIVs close on break detection.

Since the small break analyses inside the containment do not credit containment back pressure, the mass and energy release calculated for breaks inside the containment are bounding for breaks outside the containment.

15.5.9.2.1 Main Steam Line Break Outside Containment

The thermal-hydraulic response of RPV and containment for a MSLB outside containment is bounded by the response to a break inside containment as presented in Subsection 15.5.4.6.1.

Event Description

The MSLB accident is assumed to occur in the steam tunnel which is the interface between the RB and Turbine Building (TB). The postulated event assumes that a main steam line instantaneously and circumferentially breaks downstream of the outermost MSIV. The plant is designed to immediately detect such an occurrence, initiate isolation of all main steam lines including the broken line. The release of reactor steam and water from the break is assumed to blowdown into the TB airspace where it is released to the environment instantaneously as a ground-level release, with no building holdup credited.

Source Term

There is no fuel damage as a result of an MSLB outside containment. The only activity available for release from the break is that which is present in the reactor coolant prior to the break. Radiation concentrations in BWRX-300 reactor coolant and steam adequate for use in design basis calculations (such as shielding, equipment design, etc.) are determined based on ANSI/ANS-18.1.

Two cases are considered for conditions when the postulated accident occurs: (1) the maximum equilibrium iodine concentration permitted for continued full power operation, and (2) the iodine concentration corresponding to the conditions of an assumed pre-accident spike. For both cases, the release to the environment is assumed instantaneous and without holdup from the TB coolant radiation concentrations.

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For postulated accidents the design basis BWRX-300 reactor coolant concentrations are adjusted to account for two conditions that may exist when the accident begins:

1. The concentration that is the maximum equilibrium value (typically 0.2 $\mu\text{Ci/gm}$ Dose Equivalent I-131) permitted under plant Operating Limits and Conditions (OLCs).
2. The concentration that is the maximum iodine spike value (typically 4.0 $\mu\text{Ci/gm}$ Dose Equivalent I-131) permitted under plant OLCs.

The plots for RPV are shown on Figures 15.5-100 through 15.5-103 and for containment parameters Figures 15.5-104 through 15.5-105.

Radiological Consequences

The dose consequences are calculated for the MSLB accident at the proposed BWRX-300 exclusion zone.

The dose consequences of the MSLB are calculated using the ADDAM computer code (refer to Section 15.5.1.2).

Mass Release

The total mass of coolant released is the amount in the line and connecting lines at the time of the break plus the amount that passes through the RPV isolation valves and the outboard containment isolation valve prior to closure. The masses of coolant and steam released from the postulated MSLB based on preliminary thermal-hydraulic analysis are:

Liquid Release = 15,878 kg

Steam Release = 2,400 kg

Table 15.5-44 provides the activity release from this coolant volume.

This analysis conservatively assumes that all of the liquid released from the break flashes to steam and is available for transport to the environment along with the steam released from the break. The mass release duration from the MSLB is equal to the maximum closure time of the containment isolation valve of 10 seconds.

No Holdup Release to the Environment Flow Rate

No holdup release to the environment is assumed with a total of 99.9% transport of the TB airspace over a period of 10 minutes after the event. *Breathing Rate*

The postulated MSLB breathing rates used are consistent with CSA N288.2. The deterministic calculations conservatively used to demonstrate compliance with CSA N288.2 are in the 95th percentile of the breathing rates for the representative person.

Decay and Daughtering Nuclides

Decay and daughtering of nuclides during the MSLB are credited in the dose model for the duration of the event.

Dose Conversion Factors

The 30-day committed whole-body dose is compared to the CNSC REGDOC-2.4.1 acceptance criteria. The ADDAM code uses dose conversion factors that comply with CSA N288.2.

Results

This radiological consequence is calculated for the 30-day whole-body dose at the exclusion zone. The results are listed in Table 15.7.7-1 for comparison to the radiological acceptance criteria.

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The MSLB accident dose consequences are bounded by the Darlington site DBA dose acceptance criteria with considerable margin as shown in Table 15.7.7-1 for comparison against the radiological acceptance criteria.

15.5.9.2.2 Large Feedwater Pipe Break Outside Containment

Flow from the RPV side of the break is bounded by feedwater breaks inside containment because of the longer pipe length. The higher pressure losses occur for a break outside containment. No operator actions are required to mitigate the event.

Event Description

The FWLB accident occurs in the BWRX-300 steam tunnel that interfaces between the RB and TB. An instantaneous circumferential break of a feedwater line is postulated. The plant is designed to immediately detect such an occurrence and initiate FW line isolation. The energetic release of reactor water from the break is assumed to blowdown into the TB airspace where it is instantaneously released to the environment as a ground-level release, with no building holdup credited.

Source Term

There is no fuel damage as a result of an FWLB outside containment in the BWRX-300. The only activity available for release from the break is that which is present in the reactor coolant prior to the break.

Radiation concentrations in BWRX-300 reactor coolant and steam adequate for use in design basis calculations (such as shielding, equipment design, etc.) are determined based on ANSI/ANS-18.1.

For DBAs, the design basis BWRX-300 reactor coolant concentrations are adjusted to account for two conditions that may exist when the accident begins:

1. The maximum equilibrium value concentration (typically 0.2 $\mu\text{Ci/gm}$ Dose Equivalent I-131) permitted under plant OLCs for continued full power operation.
2. The maximum Dose Equivalent I-131 concentration (typically 4.0 $\mu\text{Ci/gm}$) permitted under plant OLCs and corresponds to the conditions of an assumed pre-accident spike.

Mass and Energy Release.

The total mass of coolant released is the amount in the line and connecting lines at the time of the break plus the amount that passes through the RPV isolation valves and the outboard containment isolation valve prior to closure. The masses of coolant and steam released from the postulated FWLB based on preliminary thermal-hydraulic analysis are:

Liquid Release = 18,090 kg

Steam Release = 733 kg

Break Flow Enthalpy = 1289.1 kJ/kg

The mass release vs. time is shown on Figure 15.5-167.

Flashing of Reactor Coolant to Steam

The FW break fluid enthalpy varies at saturation conditions at differing pressures. As the coolant exits the break into the steam tunnel, some fraction of the coolant released from the break flashes to steam and becomes available for release to the environment.

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The flash fraction (FF) is determined using a constant enthalpy process:

$$FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$$

Where:

h_{f1} = The enthalpy of fluid within the FW line at normal operating conditions in kJ/kg

h_{f2} = The specific enthalpy of water at atmospheric pressure in kJ/kg

h_{fg} = The heat of vaporization at 100 °C in kJ/kg

The flashing fraction applied to the liquid release is 0.390.

No Holdup Release to the Environment Flow Rate

No holdup release to the environment is assumed with a total of 99.9% transport of the TB airspace over a period of 10 minute after the event.

Breathing Rates

The postulated FWLB the breathing rates used are consistent with CSA N288.2. The deterministic calculations conservatively used to demonstrate compliance with CSA N288.2 are in the 95th percentile of the breathing rates for the representative person.

Decay and Daughtering Nuclides

Decay and daughtering of nuclides during the FWLB are credited in the dose model for the duration of the event.

Dose Conversion Factors

The 30-day committed whole-body dose is compared to the CNSC REGDOC-2.4.1 acceptance criteria. The ADDAM code uses dose conversion factors that comply with CSA N288.2.

Radiological Consequences

The dose consequences were calculated for the FWLB accident outside containment at the proposed BWRX-300 exclusion zone.

Two cases are considered for conditions that may exist when the postulated accident occurs: (1) the maximum equilibrium iodine concentration permitted for continued full power operation, and (2) the iodine concentration corresponding to the conditions of an assumed pre-accident spike. For both cases, the release to the environment is instantaneous and without TB holdup.

Radiation concentrations in BWRX-300 reactor coolant and steam are determined based on ANSI/ANS-18.1.

The dose consequences of the FWLB are calculated using the ADDAM computer code (refer to Section 15.5.1.2).

This radiological consequence is calculated for the 30-day whole-body dose at the exclusion zone. The results are listed in Table 15.7.7-2 for comparison to the radiological acceptance criteria.

Results

The FWLB accident dose consequences are bounded by the Darlington site DBA dose acceptance criteria with considerable margin as shown in Table 15.7.7-2.

15.5.9.2.3 Shutdown Cooling System Pipe Break Outside Containment

The shutdown cooling system (SDC) (Chapter 9, Subsection 9A.2.3) provides decay heat removal for refueling or maintenance. The SDC provides decay heat removal at normal and lower reactor operating pressures. SDC is subjected to high energy conditions for a short time (less than 2% of the plant operating conditions). The system piping is assigned a medium energy line due to the short time that it is subjected to high energy conditions.

The SDC connects to the feedwater system between the feedwater containment isolation and feedwater isolation control valve outside containment. Due to the smaller SDC piping diameter, the feedwater pipe break outside containment discussed previously in Subsection 15.5.9.2.2 bounds the SDC pipe break outside containment.

15.5.9.2.4 Large Isolation Condenser Pipe Breaks Outside Containment

The isolation condenser pipe configuration outside the containment is under development. For the worst-case scenario, the large isolation condenser pipe break outside containment may be larger than the large isolation pipe breaks inside containment. The mass and energy release from the isolation condenser pipe breaks outside containment are still bounded by the isolation condenser pipe breaks inside containment. Breaks inside containment remain bounding because the isolation condenser pipe breaks do not utilize any DL3 functions that depend on containment parameters, and the containment back pressure is not credited in any of the isolation condenser pipe breaks inside containment.

As is the case for isolation condenser pipe breaks inside containment, core response is not a concern since the break is isolated rapidly. The consequences of large isolation condenser pipe breaks outside containment require an evaluation for the loads, pressures, and temperatures outside containment, and radiological consequences resulting from normal operation coolant activity.

15.5.9.2.5 Small Breaks Outside Containment

Since the small break analyses inside containment do not credit containment back pressure, the mass and energy release calculated for breaks inside containment are bounding for breaks outside containment.

However, dose analyses are performed for breaks outside containment for two line break cases:

- ICS line break
- Instrument line break

ICS Line Break Dose Consequences

Event Description

The ICS line break accident analyzed for offsite dose consequences is a postulated break of an ICS steam supply pipe in the ICS pool on the operating deck of the RB. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line within 5 seconds, and fully isolate the break in 10 seconds. Blowdown steam from the break is directed to the two heat exchangers in one ICS unit and the liquid mass in the heat exchangers is expelled to the ICS pool where it mixes with the pool water without flashing. The energetic release of reactor steam from the break is assumed released to the environment instantaneously as a ground-level release, without holdup.

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Source Term

There is no fuel damage as a result of an ICS line break outside containment. The only activity available for release from the break is that which is present in the reactor coolant prior to the break.

Radiation concentrations in reactor coolant and steam are adequate for use in design basis calculations (such as shielding, equipment design, etc.) and are determined based on ANSI/ANS-18.1.

For postulated accidents the design basis reactor coolant concentrations are adjusted accounting for two conditions that may exist when the accident begins:

- The concentration that is the maximum equilibrium value (typically 0.2 $\mu\text{Ci/gm}$ Dose Equivalent I-131) permitted under plant OLC.
- The concentration that is the maximum iodine spike value (typically 4.0 $\mu\text{Ci/gm}$ Dose Equivalent I-131) permitted under plant OLC.

Mass Release

The mass of steam released from the postulated ICS line break is 400.8 kg and is based on preliminary GEH thermal-hydraulic analysis.

Release Duration

The mass release duration from the ICS line break is equal to the maximum closure time of the isolation valves that is 10 seconds.

The mass of steam released from the break is transported to the environment over a 10 minute period which adequately simulates instantaneous transport.

Breathing Rates

The postulated ICS line break breathing rates used are consistent with CSA N288.2. The deterministic calculations conservatively used to demonstrate compliance with CSA N288.2 are in the 95th percentile of the breathing rates for the representative person.

Decay and Daughtering Nuclides

Decay and daughtering of nuclides during the ICS line break are credited in the dose model for the duration of the event.

Dose Conversion Factors

The 30-day committed whole-body dose is compared to the CNSC REGDOC-2.4.1 acceptance criteria. The ADDAM code uses dose conversion factors that comply with CSA N288.2.

Results

The ICSLB accident dose consequences are bounded by the Darlington site DBA dose acceptance criteria with considerable margin as shown in Table 15.7.7-3.

Instrument Line Break Dose Consequences

The dose consequences of an Instrument Line Break Accident (ILBA) at the exclusion zone considers two cases that may exist when the postulated accident occurs: (1) the maximum equilibrium iodine concentration permitted for continued full power operation, and (2) the iodine concentration corresponding to the conditions of an assumed pre-accident spike. For both cases, the line break release to the environment from the RB is instantaneous and without holdup.

The dose consequences of the ILBA are calculated using the ADDAM computer code.

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Event Description

A circumferential rupture of an instrument line connected to the primary coolant system is postulated to occur outside primary containment in the RB. The ILBA analysis assumes that the event cannot be isolated, and no fuel damage occurs. The resulting activity is released to the environment directly from the RB with no credit for holdup or filtration. Primary coolant flows at the maximum rate for a typical instrument line that has a 1/4" flow restricting orifice. Saturated water flows in the instrument line into containment that flashes to steam, resulting in the maximum iodine release.

Source Term

There is no fuel damage as a result of this accident. The only activity released from the break is that present in the reactor coolant prior to the break.

Radiation concentrations are determined using ANSI/ANS-18.1-2020 for the reactor coolant and steam used in design basis calculations for shielding and equipment design.

The design basis reactor coolant and steam concentrations are adjusted to account for two conditions that may exist when the postulated accident occurs:

1. The maximum equilibrium value concentration (typically 0.2 $\mu\text{Ci/gm}$ Dose Equivalent I-131) permitted by the plant OLCs.
2. The maximum iodine spike value (typically 4.0 $\mu\text{Ci/gm}$ Dose Equivalent I-131) concentration permitted by the plant OLCs.

The ILBA Airborne Release Source Term for equilibrium and iodine spike is provided in Table 15.5-43A and 15.5-43B, respectively. Only the equilibrium concentrations of iodine and iodine spike activity releases differ. The concentrations of the remaining isotopes are not impacted by iodine behavior.

Coolant Mass Release

The mass of steam released 408.0 kg from the postulated ICS line break is used for the ILBA based on preliminary GEH thermal-hydraulic analysis. The release duration is equal to the maximum closure time of the isolation valves in the ICS break outside containment that is 10 seconds. The mass of released break steam is transported to the environment over a 10-minute period that adequately simulates instantaneous transport.

Release Duration

The steam mass released is transported from the RB to the environment over a 10-minute period that adequately simulates instantaneous transport with no holdup. As a result, the steam release duration to the environment follows the coolant mass release duration.

No Holdup Release to the Environment Flow Rate

No holdup release total transport of 99.9% from the TB airspace to the environment occurs over a period of 10 minute after the event.

Breathing Rates

The postulated ILB breathing rates used are consistent with CSA N288.2. The deterministic calculations conservatively used to demonstrate compliance with CSA N288.2 are in the 95th percentile of the breathing rates for the representative person.

Decay and Daughtering

Decay and daughtering of nuclides during the MSLB event are credited in the dose model for the duration of ILBA event.

Dose Conversion Factors

The 30-day committed whole-body dose is compared to the CNSC REGDOC-2.4.1 acceptance criteria. The ADDAM code uses dose conversion factors that comply with CSA N288.2.

Results

The BWRX-300 ILB dose consequences are bounded by the Darlington site DBA dose acceptance criteria with considerable margin as shown in Table 15.7.7-4.

15.5.9.3 BWRX-300 Out of Core Criticality Analysis

A representative Out Of Core Criticality (OCC) accident scenario has been analyzed in the BWRX-300 Out of Core Criticality Safety Analysis Demonstration that shows that the dose consequences at the site boundary do not exceed the generic criteria to trigger a public evacuation (see Canadian Guidelines for Intervention During a Nuclear Emergency - Reference 15.5-12). Criticality analyses assess fuel handling activities outside the core for the GE High Density Fuel Storage Racks (HDFS) and the RAJ-II Inner Container (IC). The HDFS rack assessment for the storage of GNF2 fuel analyses resulted in a storage rack maximum k_{eff} ($k_{\text{max}}(95/95)$) less than the USL of 0.95 for normal and credible abnormal operation. The OCC analysis demonstrates compliance to REGDOC-2.4.3, Section 2.3.2.2, Part 3 (Reference 15.5-13) by computationally investigating a representative accident scenario that complies with REGDOC-2.4.3, Section 16.4. The representative OCC accident scenario is postulated to occur inside the RB on the refueling floor due to the unsafe stacking of RAJ-II-ICs.

Method of Analysis

The estimated total number of fissions follows the use of a simplified models approach described in BSI ISO 16117:2013 (Reference 15.5-14):

- The estimate of the number of fissions should be based on simplified options providing “order of magnitude” values
- This estimate should rely on the collective experiences from past criticality accidents (Annex B) and criticality experiment results (Annex C) and the possible use of simplified formulae (Annex D)
- When a simplified model is used, the consistency of its area of applicability with the chosen assumptions of the postulated criticality accident shall be justified and documented

The parameters extracted from ISO 16117:2013, Table B.2 (Reference 15.5-14) for the purpose of this analysis is provided in Table 15.5-45. The total number of fissions allows for determining the neutron and photon source term magnitudes that are used subsequently in developing MCNP-06P models that are used to assess the dose consequence.

Computational Models

Computational models have been developed in MCNP-06P to assess the neutron and photon dose consequence of the postulated OCC accident scenario. MCNP-06P is the GEH controlled version of the Los Alamos National Laboratory code MCNP6 (Reference 15.5-15). MCNP6 is a general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron/photon/electron Monte Carlo transport code. All models employ ENDF/B-VII.0 cross section libraries. Compliance with CSA standards is demonstrated and confirmed.

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The first computational model is developed for the purpose of tracking the particle flux/dose as a function of distance in air from the accident location (see Figure 15.5-168). The second computational model is developed to study particle flux attenuation as a function of concrete thickness present to predict shielding behavior in the refueling floor area where the accident occurs. The relative decrease in neutron/photon radiation dose rates is utilized to approximate the dose consequence as a result of shielding present.

Source Term

The total number of fissions shown in Table 15.5-46 is used to determine the neutron and photon source term magnitudes utilized in the MCNP-06P calculations. The energy distribution of the neutron source is governed by the Watt energy spectrum. The photon source term is discretized into two energy groups with even probability. The first group is the average energy emitted per prompt fission photon, and the second group is the maximum prompt fission photon energy.

Radiation Dose Rates

For all computational models, the neutron and photon dose rates are computed by utilizing the ANSI/ANS-6.1.1-1977 (Reference 15.5-11) particle flux to dose conversion coefficients in the F4 tallies to yield units of mSv/H. The time-integrated dose is calculated by scaling the total dose rates by the accident duration to yield time-integrated dose units of mSv. The resultant doses at the site boundary are then directly compared to the generic criteria from Canadian Guidelines for Intervention During a Nuclear Emergency (see Table 15.5-47) to show that no protective actions are necessary to ensure the safety of the public in the event of the postulated representative OOC accident.

Results

The results for the total integrated dose provided in Table 15.5-47 are compared directly to the generic criteria for implementing actions to protect the public provided in Table 15.5-48. The total integrated dose is plotted on Figure 15.5-169 with respective error bands to highlight dose behavior as a function of distance from the OOC accident location, with and without the attenuation provided by concrete shielding.

The total integrated doses provided in Table 15.5-46 was conservatively based upon the total number of fissions actually occur during the accident excursion timeframe. These results are deemed to be conservative based on the following:

1. The unsafe stack of RAJ-II ICs is originally confined in space by the gravitational, normal, and static friction forces. The fission force magnitude that occurs during the accident within the critical system will overwhelm the gravitational and normal and frictional forces holding the bundles in their original locations ultimately displacing the fuel far enough apart to yield a subcritical system.
2. There are established RAJ-II IC stacking limits and fuel handling patterns in place to avoid unsafe ICs stacking. In order for an OOC scenario to occur, workers would violate operating procedures or have inadequate training.

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In the event that workers violate established operational procedures, there are natural and physical barriers in place mitigate the likelihood of an OOC event sequence. The following physical barriers are DL1 design measures that prevent an OOC occurrence:

1. The time it takes to obtain and arrange RAJ-II ICs in an 8x8 stack on the refuel floor is significant. The ICs must each be handled via crane and cannot easily be displaced. It is quite unlikely that other workers in the reactor building would ignore the workers unsafely stacking ICs during this entire duration. There are safety checks in place and verifications made during the process that prohibit unsafe stacking.
2. The new fuel receipt and handling plan schedule is provided to the plant that contains the number of fuel bundles that arrive on site for the first 14 years of plant operation. There is a limit to how many ICs are allowed on the refuel floor at one time.
3. The criticality analysis was based on an average bundle fuel enrichment of 5 wt% U-235 for each RAJ-II IC in the stack. This is conservative and unlikely as the average enrichment for the BWRX-300 core is less than 5 wt% U-235.

Conclusions

The dose consequence of a representative criticality accident scenario in accordance with REGDOC-2.4.3, Section 16.4.1 shows that the dose consequence at the site boundary does not exceed the generic criterion that would trigger a public evacuation in accordance with Canadian Guidelines for Intervention During a Nuclear Emergency (Reference 15.5-12).

15.5.10 Analysis of Internal and External Hazards

The internal and external hazards portion of the PSA (refer to Subsection 15.6.2.4) explicitly analyzes radionuclide release accidents initiated during power and shutdown operation for internal and external hazards. Malevolent acts are addressed in the Security Annex.

The internal and external hazards is presented in Chapter 3. Sections 3.4 and 3.3, respectively. The design features to mitigate these hazards are provided for the specific hazards in the following subsections.

15.5.10.1 Analysis of Internal Hazards

The description of internal hazards and design features are described in the following sections:

Fires – Chapter 3, Section 3.4.1 hazard identification, design features Chapter 9A, Section 9A.6

Explosions – Chapter 3, Section 3.4.1 hazard identification, design features in Chapter 9B, Section 9B

Toxic Gas – Chapter 3, Subsection 3.4.1 hazard identification, MCR and SCR design features in Chapter 6, Section 6.4

Internal Flooding – Chapter 3, Subsection 3.4.2 hazard identification, structural design in Chapter 9B, Section 9B

Internal Missiles – Chapter 3, Subsection 3.4.3 hazard identification, structural design in Chapter 9, Section 9B, equipment and pipe design features in Chapter 3, Section 3.6, and Chapter 5

High Energy Line Breaks (HELB) – Chapter 3, Subsection 3.4.4 hazard identification, HELB evaluation in Chapter 3, Subsection 3.6.7

15.5.10.2 Analysis of Natural External Hazards

The description of external hazards and the structural design features are described in the following sections:

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Seismic – Chapter 3, Subsection 3.3.1 hazard identification, structural design in Chapter 9, Section 9B

Extreme Weather (Hurricanes, Tornadoes, Wind, etc.) – Chapter 3, Subsection 3.3.2 hazard identification, structural design in Section 9B

Hydrogeological (Floods) – Chapter 3, Subsection 3.3.3 hazard identification, structural design in Chapter 9, Section 9B

Aircraft Impact – Chapter 3, Subsection 3.3.4 hazard identification, structural design in Section 9B

Missiles – Chapter 3, Subsection 3.3.5 hazard identification, structural design in Section 9B

Fires – Chapter 3, Subsection 3.3.6 hazard identification, design features in Chapter 9, Section 9A.6

Explosions – Chapter 3, Subsection 3.3.6 hazard identification, structural design in Chapter 9, Section 9B

Toxic Gas – Chapter 3, Subsection 3.3.6 hazard identification, design features in Chapter 6, Section 6.4

15.5.10.3 Analysis of External Human-Induced Hazards

The external human-induced hazards is described in Chapter 3, Subsection 3.3.7. Malevolent acts is addressed in the PSAR Security Annex.

15.5.11 References

- 15.5-1 NEDE-32177, "TRACG Qualification," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.5-2 NEDO-33922-A, "BWRX-300 Containment Evaluation Method," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.5-3 NEDC-33987P, "TRACG Application for BWRX-300," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.5-4 GOTHIC Thermal Analysis Package Technical Manual, Version 8.3(QA), November 2018.
- 15.5-5 GOTHIC Thermal Analysis Package Qualification Report, Version 8.3(QA), November 2018.
- 15.5-6 CSA N286.7, "Quality Assurance of Analytical, Scientific, and Design Computer Programs," CSA Group.
- 15.5-7 NEDC-32082P SH 0001, "BWR Steady State Thermal Hydraulic Methodology (ISCOR)," GE-Hitachi Nuclear Energy Americas, LLC.
- 15.5-8 CANDU Owner's Group, Inc., SQAD-20-5065, "ADDAM 1.4.2 User's Manual," April 2021.
- 15.5-9 CSA N288.2-19, "Guidelines for calculating the radiological consequences to the public of a release of airborne radioactive material for nuclear reactor accidents," CSA Group
- 15.5-10 ANSI/ANS-18.1-2020, "Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society.
- 15.5-11 ANSI/ANS-6.1.1-1977, "American National Standard: neutron and gamma-ray flux-to-dose-rate factors," American National Standard.

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- 15.5-12 Government of Canada, "Canadian Guidelines for Intervention During a Nuclear Emergency," Minister of Health.
- 15.5-13 CNSC Regulatory Document REGDOC-2.4.3, "Nuclear Criticality Safety."
- 15.5-14 BS ISO 16117, "Nuclear criticality safety - Estimation of the number of fissions of a postulated criticality accident," British Standards Institution.
- 15.5-15 LA-UR-17-29981, "MCNP User's Manual^(R) - Code Version 6.2," Los Alamos National Security, LLC.
- 15.5-16 NEDC-33939P, "BWRX-300 Darlington New Nuclear Project (DNNP) GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology," GE-Hitachi Nuclear Energy Americas, LLC.

15.6 Probabilistic Safety Assessment

A principal element of the design process is the development and results of the Probabilistic Safety Assessment (PSA). The PSA provides an integrated review of the plant design, operational safety, and complements the results of the deterministic analyses. The PSA measures how the safety of plant design and operation prevents the risk of releasing radionuclides to the environment. The PSA supports risk-informed design development and with the DSA, demonstrates the success of the design in achieving the design objectives. The PSA assesses design vulnerabilities and optimizes the design using a graded approach consistent with CNSC REGDOC-2.4.2, "Probabilistic Safety Assessment (PSA) for Nuclear Power Plants," (Reference 15.6-1).

CNSC REGDOC-2.4.2, Requirement 3.2 (Reference 15.6-1) requires a Level 1 and Level 2 PSA. The BWRX-300 PSA meets the performance requirements of ANSI/ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment For Advanced Non-Light Water Reactor Nuclear Power Plants Publication," (Reference 15.6-2).

Other important sources of PSA guidance are in USNRC Regulatory Guide (RG) 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 15.6-3), USNRC RG 1.206, "Applications for New Power Plants" (Reference 15.6-4), International Atomic Energy Agency (IAEA) SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants" (Reference 15.6-5), and IAEA SSG-4, "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Plants," (Reference 15.6-6) provide guidance for performance of Level 1 and Level 2 PSAs. Canadian Standards Association (CSA) N290.17-17, "Probabilistic safety assessment for nuclear power plants" (Reference 15.6-7) provides high-level guidance and guidelines for performing a PSA. These PSA regulatory guides align with CNSC REGDOC-2.4.2 (Reference 15.6-1). The PSA level of detail expands as the design details are finalized in compliance with CNSC REGDOC-2.4.2, Requirement 3.8 (Reference 15.6-1) that states that the level of detail must be consistent with testing, maintenance, configuration management programs, and the intended PSA uses.

CNSC REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility," Section 4.4.5, (Reference 15.6-8) requires that "the PSAR includes a deterministic safety analysis, a PSA and a hazards analysis that demonstrate all levels of D-in-D" are addressed and confirms that the facility's design is capable of meeting applicable dose acceptance criteria and safety goals.

CNSC REGDOC-1.1.2 (Reference 15.6-8) and CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 15.6-9), require performance of Hazard Analysis, PSA, and DSA. Section 15.6 is a summary of the work performed for the Hazard Analysis Screening and PSA. The PSA is updated for the operating licence application.

The PSA is updated as additional design and site-specific information becomes available for the operating licence application.

15.6.1 General Approach to Probabilistic Safety Assessment

The PSA risk assessments are divided into two levels:

- Level 1 calculates Core Damage Frequency (CDF) inside containment
- Level 2 calculates Small Release Frequency (SRF) and Large Release Frequency (LRF), with the associated release magnitude

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The Level 1 PSA consists of:

1. Initiating Events (IE) –events that may challenge fuel cooling. The initiating events may occur when the unit is at-power, low power, or shutdown.
2. Event Tree Analysis – plant response analysis for mitigating systems credited for maintaining fuel cooling. Mitigating systems failures result in fuel cooling failures and subsequent core damage.
3. Fault Tree Analysis – models mitigating system failure. Fault tree analysis requires human reliability inputs and component failure including CCF.
4. Accident Sequence Quantification – determines frequency of the core damage end states.
5. Uncertainty and Sensitivity Analyses – assesses the statistical and phenomenological uncertainties. Sensitivity analyses determines initiating events, mitigating systems, or phenomena that has major impact on the results.

Level 2 PSA consists of:

1. Reactor Coolant System (RCS) / Containment Response Analysis
2. Developing an interface between Level 1 and Level 2 PSA
3. Identifying and modeling safety functions and operator actions
4. Performing containment performance analysis
5. Developing Containment Event Tree
6. Phenomena Analysis – consisting of accident progression analysis
7. Source term analysis
8. Quantifying the model and results for LRF and SRF and consequences
9. Performing uncertainty and sensitivity analysis for SA release category, character, and quantification.

The PSA is divided into internal and external events. The internal events occur within the plant (e.g., loss-of-coolant, loss of feedwater, internal fire, etc.). External events occur from outside the plant (e.g., seismic, high wind, flood etc.).

Fuel Pool events address spent fuel damage.

The PSAR PSA scope includes:

A. Level 1 PSA hazards:

1. Internal Events are events at-power, low power and shutdown, internal flood at-power, and internal fire at-power
2. External Events are seismic event at-power, and high wind at-power

The development of BWRX-300 PSA models are conducted with Electric Power Research Institute (EPRI) Integrated Risk Technology suite of software packaged under the Phoenix Architecture, which includes the following:

- CAFTA for event tree, fault tree and cut set development and viewing purposes
- Phoenix Application Programming Interface (API) for model integration and one-top development
- PRAQuant for quantification processes, as well as for sensitivity runs

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The PSA model is integrated and quantified using the computer codes CAFTA, PRAQuant, and Fault Tree Reliability Evaluation eXpert (FTREX). These computer codes have been demonstrated throughout the industry to produce appropriate results. No method-specific limitations have been identified with regard to the software tools or the methodology implemented to quantify the model.

Using CAFTA, the BWRX-300 PSA model is developed by merging all model event trees, system fault trees, IEs, passive feature failure results, and associated basic event databases. Using PRAQuant, a top logic fault tree is created. All system fault trees are merged with the top logic fault tree. The model is quantified to arrive at event sequence frequencies for the various accident sequence end states and release categories.

Figure 15.6-1 demonstrates the principal steps in PSA model development.

The BWRX-300 PSA methodology approaches each PSA element. The level of detail for some technical areas does not include all steps but may refer to specific guidance documents or reports. For example, in the fire PSA, the description refers to USNRC NUREG/CR-6850 "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (Reference 15.6-20) and USNRC NUREG/CR-6850 Supplement 1 "Fire Probabilistic Risk Assessment Methods Enhancements" (Reference 15.6-27) that provides details of each fire PSA step. Not all the described methods are applied early in the design.

Hazard and scenarios that are estimated as lower risk for the final design may be initially analyzed early in the design with simplified scoping evaluations, with a detailed PSA performed later in the design phase or during the site-specific PSA. Where these simplifications are applied, this is generally noted in the document. The DNNP PSA evolves with the design and is subject to IAEA-TECDOC 1106, "Living Probabilistic Safety Assessment (LPSA)" (Reference 15.6-10). The PSA is maintained and updated as the design matures and procedures are developed, when the plant(s) is built, Operating Experience (OPEX) is gathered and applied for any design or operating modifications.

External hazards analysis addresses possible combinations of external hazards. For example, high wind causing loss of the external grid is considered in the high wind analysis. Other site-specific combinations (such as snowstorm blocking the diesel generator combustion air intake and frazil ice blocking sea water cooling) are assessed.

The design phase PSA covers both at-power (full and reduced power levels) and low power and shutdown operations, which include outages and transitions between each plant operating state (POS) in accordance with the guidance of CNSC REGDOC-2.4.2, Section 3.10 (Reference 15.6-1). The design phase PSA includes main systems and components (including the reactor and turbine I&C logic), main operator actions, and the relevant event and system dependencies, interconnections, and CCF relationships.

The development of DNNP PSA models is conducted with the EPRI Integrated Risk Technology Phoenix Architecture. The computer codes CAFTA and PRAQuant, and FTREX are used to integrate and quantify results into PSA model. These computer codes are known throughout the industry to produce appropriate results. No method-specific limitations have been identified regarding the software tools or the methodology implemented to quantify the model. Chapter 3, Appendix 3G provides additional discussion about all the computer programs used in the PSA.

The BWRX-300 PSA model is developed by merging all model event trees, system fault trees, initiating events, passive feature failure results, and associated basic event databases. A top logic fault tree is created. All system fault trees are merged with the top logic fault tree. The model is then quantified to arrive at event sequence frequencies for the various accident sequence end states and release categories.

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At a minimum, the following areas are addressed in the design phase PSA:

- DNNP risk results compared to Safety Goals:
 - CDF
 - SRF
 - LRF
- Initiating event selection, grouping, and frequency for each analyzed plant operating state
- Event sequence modeling and quantification for accident sequence end state and release category
- System analysis and system models
- Success criteria
- Component reliability data (including CCF data and modeling principles)
- Human Reliability Analysis (HRA)
- Uncertainty and sensitivity analysis
- Containment analysis
- Mechanistic source term analysis for each release category
- Model documentation

The design phase PSA includes the PSA models as well as documentation describing plant characteristics important to safety, modeling assumptions and techniques, model structure, data values sources used in the model, and the analysis results are explained making it possible to review and replicate the analyses.

CNSC REGDOC-2.4.2, Section 3.3 (Reference 15.6-1) requires that the PSA be conducted under the management system or quality assurance program established in the licensing basis. This section shows how PSA output (in concert with the DSA) provides input into the design that may affect the PSA models. All work for the design phase PSA is performed in accordance with the requirements of the BWRX-300 Quality Assurance Program Plan.

Regulatory PSA Requirements

A PSA is required to be conducted in accordance with the defined requirements for a licence to construct or operate a reactor facility per CNSC REGDOC-2.4.2, Section 1.1 (Reference 15.6-1). CNSC REGDOC-1.1.2, Section 4.4.1 (Reference 15.6-8) states that the PSA is required to support the PSAR.

In reviewing CNSC REGDOC-2.4.2 (Reference 15.6-1) PSA objectives, six of the eight objectives are being met. The two outstanding objectives require detailed Emergency Operating Procedures (EOPs) and Severe Accident Management Guideline (SAMG) procedures that are developed during the final safety analysis. Per CNSC REGDOC-2.4.2, Section 2.0 (Reference 15.6-1), the PSA objectives are:

- A. Providing a systematic analysis in order to give confidence that the reactor facility's design will align with the fundamental safety objectives as established in IAEA No. N-SF-1, Fundamental Safety Principles, including to protect people and the environment from harmful effects of ionizing radiation.

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- B. Demonstrating that a balanced design has been achieved; this can be demonstrated as achieved if no particular feature or postulated IE makes a disproportionately large or significantly uncertain contribution to the overall risk.
- C. Providing confidence that small changes of conditions that may lead to a catastrophic increase in the severity of consequences (cliff-edge effects) are prevented.
- D. Providing assessments of the quantitative safety goals (the probabilities of occurrence for severe core damage states, and the assessments of the risks of radioactive releases to the environment) as defined in CNSC REGDOC-2.5.2 (Reference 15.6-9) or as established in the licensing basis for the facility.
- E. Providing site-specific assessments of the probabilities of occurrence and the consequences of external hazards.
- F. Identifying facility vulnerabilities and systems for which design improvements or modifications to operational procedures could reduce the probabilities of severe accidents or mitigate their consequences.
- G. Assessing the adequacy of EOPs throughout the nuclear power plant lifetime.
- H. Providing insights into the severe accident management program used in the development, implementation, training, and optimization of accident management strategies and measures.

This section provides the hazard analysis summary presenting the methodology, hazard identification and the screening results in compliance with CNSC REGDOC-1.1.2, Section 4.4.1 (Reference 15.6-8).

Regulatory Safety Goals

The CNSC defined safety goals for core damage, small release, and large release frequencies are defined in CNSC REGDOC-2.5.2 (Reference 15.6-9), as follows:

Core Damage Frequency

The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 10^{-5} per reactor year.

Small Release Frequency

The sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{15} becquerels of iodine-131 shall be less than 10^{-5} per reactor year. A greater release may require temporary evacuation of the local population.

Large Release Frequency

The sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{14} becquerels of cesium-137 shall be less than 10^{-6} per reactor year. A greater release may require long-term relocation of the local population.

The DNNP PSA is performed iteratively with the design development to evaluate and improve the risk aspects of the design.

The PSA and severe accident evaluations specific objectives demonstrate that the BWRX-300 is designed with state-of-the-art safety features, incorporating highly reliable and available passive safety functions with significant redundancy and diversity meeting all established safety goals with margin.

The DNNP design PSA uses information available from the BWRX-300 plant design and procedures that meets the intent of CNSC REGDOC-2.4.2, Section 3.4 (Reference 15.6-1), for the PSA to reflect the as-built and as-operated plant. Periodic updates of the PSA (CNSC REGDOC-2.4.2, Section 3.5 (Reference 15.6-1)) are required after the plant begins operation and every five years thereafter. Component failure data and IE frequencies are based on generic U.S. industry data compared to the BWRX-300 design. As required by CNSC REGDOC-2.4.2, Section 3.7 {Reference 15.6-1}, this data and the assumptions used in developing the PSA are selected realistic as practical. Because site or programmatic information is not available during the design, the PSA analyses contains conservative elements (e.g., human error probabilities, maintenance delay or postponement, component failure rates, flood and fire initiation, propagation, and effects).

15.6.1.1 Principal Steps in the Probabilistic Safety Assessment Model Development

A design process that is “risk-informed” incorporates the results of risk evaluations and PSA where risk is evaluated for both the frequency and consequences of a possible event. In the nuclear industry, where design, operation, and maintenance are generally conducted using a prescriptive set of requirements, risk evaluations provide information on potential vulnerabilities of the design that may not be revealed by the deterministic design process. In using risk evaluations to assess a design, the risk results add to or supplement the design resulting in an increase in its overall reliability and safety. Using the PSA during the design process ensures that no particular feature or postulated IE makes a disproportionately large or significantly uncertain contribution to the overall risk. The application of risk information and PSA to supplement and improve a design during the design process is to risk-inform the design.

A design process that is “performance-based” relies on a process or equipment measurable outcomes as evidence of meeting a requirement or objective. One advantage of a performance-based approach to design is the flexibility available to meet the outcome requirement or objective. This contrasts with a prescriptive requirement where the adherence to the requirement infers an acceptable outcome. A process that is both risk-informed and performance-based uses both the process to inform and improve the design and the design process.

The full benefits of risk-informed, performance-based methods applied to the design for a new reactor are achieved through a disciplined and quantitative evaluation of options considered during the design process. The effectiveness of risk-informed, performance-based methods rely on the presence of four process elements that ensure a minimum necessary structure is available for these evaluations. These four process elements are:

- Design criteria or requirements
- Systems engineering process
- D-in-D
- Sequence-based risk assessment (using PSA)

The design process uses the PSA results at each stage in the design to confirm the design adequacy and to evaluate alternatives being considered during the system engineering and design process. The PSA interfaces with the design requirements, the system engineering process, and D-in-D evaluations. The PSA process and use is described throughout this section.

The internal events at-power model is developed first and covers these principal steps. The internal and external hazard models and the low-power and shutdown models are developed based on the internal events at-power model files.

By using the ASME/ANS standards for LWR PSA, the DNNP PSA is also required to follow what is normally referred to as a Level 2 assessment, which accounts for the response of containment

following core damage events. IAEA SSG-4 (Reference 15.6-6) provides guidance on assessing safety for low power and shutdown states. Where applicable, these guides are used to inform the DNNP PSA efforts.

15.6.1.2 Internal and External Events and Level 1 PSA

The internal and external hazards portion of the PSA analyzes radionuclide release accidents initiated during power and shutdown operation for the following hazards:

- Internal Hazards:
 - Internal floods
 - Internal fires
- External Hazards:
 - High winds
 - Seismic events
- Other hazards (internal and external). Risk assessment is based on the results of screening analyses that identifies potential significant contributors.

The internal and external hazard analyses are initially bounding assessments that show significant design margin for these hazards. The frequencies of internal events are based on generic industry data or bounding site data and are applied in a bounding manner. The fault trees and event trees developed for the internal events evaluations are used in the internal and external hazards analyses to the extent possible, using logic flags or additional fault tree modeling that account for the failures induced by the internal or external hazard events.

The scoping Seismic PSA used the seismic hazard evaluated at the Darlington site. The scoping Seismic PSA used generic bounding seismic capacities for SSCs that are set conservatively low. Using conservative SSC generic seismic capacities reveals those SSCs that contribute significantly to the seismic safety profile. With this information, the detailed design is performed demonstrating that sufficient hardening already exists (Chapter 9, Section 9B provides additional information regarding hardening) and provides focus for additional design enhancements to achieve the BWRX-300 safety goals through a risk-informed design process.

The internal events PSA analyzes and quantifies initiating internal events. Internal events generally include events occurring within the plant boundary, other than those involving internal hazards such as internal fires or floods. Historically, the internal events PSA includes the analysis of loss of offsite or preferred power, which has potential contributors both on-site (e.g., from the switchyard or switchgear) and offsite (e.g., grid or weather-related losses not involving other external hazards).

15.6.1.2.1 Internal Hazards

This section summarizes the screening for internal hazards for the BWRX-300 plant. Internal Hazards were screened for both consideration in the Probabilistic Safety Assessment (PSA) and for inclusion in the fault list. In addition, potential non-reactor sources of radioactivity are screened in this section.

From IAEA SSG-64, "Protection against Internal Hazards in the Design of Nuclear Power Plants," (Reference 15.6-11):

"Internal hazards are those hazards to the safety of the nuclear power plant that originate from within the site boundary and are associated with failures of facilities and activities that are under the control of the operating organization."

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This section documents the screening of internal hazards for the BWRX-300 Standard Plant Design. Because the BWRX-300 PSA is an all-hazards PSA, internal hazards are evaluated for potential PSA development. NEDC-33946P, "BWRX-300 Darlington New Nuclear Project (DNNP) Probabilistic Safety Assessment Methodology" (Reference 15.6-25) provides guidance for screening external hazards; however, the screening approach may be adopted for the screening of internal hazards as well. The PSA Standard, ANSI/ASME/ANS RA-S-1.1-2022, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 15.6-12), Sections 1 and 6 provide several screening criteria.

The general approach screens internal hazards using the following steps:

1. Develop a list of candidate internal hazards.
2. Apply qualitative screening criteria to each of the internal hazards.
3. For those internal hazards remaining unscreened (i.e., pose a credible threat to nuclear safety) from Step 2, consider application of quantitative screening.
4. Any hazards left unscreened after Step 3 are retained as candidates for explicit treatment in the PSA.

In addition, on-site radioactivity sources are screened for potential treatment in the PSA.

Scope

The internal hazard scope assessment applies screening criteria to all hazards originating within the site boundary. Several sources are consulted for developing of the internal hazards list. A list of on-site radioactivity sources is considered for treatment in the PSA. This subsection does not assess external hazards, sabotage/terrorism, nor is it used in the screening of external events.

Because the BWRX-300 uses a standard plant design and because internal hazards apply to those that originate within the site boundary, this screening section applies to both standard plant design and the DNNP.

Assumptions

Administrative controls are implemented to preclude compressed gas cylinders from becoming missiles in areas containing risk-significant mitigating equipment.

1. Valves are designed to prevent removable parts from becoming missiles in the event of failure in accordance with guidance in SSG 64 (Reference 15.6-11).
2. Rotating equipment (excluding the main turbine) is designed such that potential failure-generated missiles are prevented from impacting risk-significant equipment through spatial or engineered means.
3. Administrative controls are placed to ensure stored combustibles are not collected in sufficient quantities to impact nuclear safety if ignited.
4. No risk-significant mitigating equipment resides in the RB Auxiliary Bay Plant Services Area truck bay or the service bay area beneath the truck bay.
5. The on-site radwaste system does not contain radioactivity in sufficient form or quantity to pose a public health hazard to the level of a small or large release as defined by the PSA.
6. Dry casks containing spent fuel, when stored or handled outside the cask pit, is capable of passive cooling and are not vulnerable to potential hazards.

Hazard List

The preliminary list of BWRX-300 internal hazards is generated from industry guidelines, past studies, and a plant-specific review.

The internal hazards list of sources includes:

1. BWRX-300 Safety Strategy.
2. ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12). This source is an aggregation of hazards based on review of industry studies such as USNRC NUREG/CR-2300, USNRC NUREG-1407, IAEA SSG-3, USNRC NUREG/CR-5042, EPRI 1022997, EPRI 3002005287, and ASAMPSA_E List of External Hazards.
3. CSA N290.17-17 "Probabilistic safety assessment for nuclear power plants," (Reference 15.6-7).
4. IAEA SSG-3 (Reference 15.6-5), "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants".
5. IAEA SSG-64 (Reference 15.6-11), "Protecting against Internal Hazards in the Design of Nuclear Power Plants".

The initial hazards screening results are provided in Tables 15.6-1 through 15.6-2.

Qualitative Screening

The following qualitative screening criteria are provided for internal and external hazards. Consideration of internal hazards is not fundamentally different than external hazards from a risk-impact perspective; therefore, these same criteria apply in screening internal hazards:

1. The event is of equal or lesser damage potential than events for which the plant is designed. This requires an evaluation of the plant design basis to estimate the resistance to a particular (internal) event.
2. The event has a significantly lower mean occurrence frequency than similar events included in the PSA and could not result in worse consequences.
3. The event cannot occur close enough to the plant to effect it.
4. The event is included in the definition of another event.
5. The event frequency is sufficiently low when compared to the probabilistic limits defined for the release category frequencies and is included in the PSA.

The following additional qualitative screening criteria are also considered:

1. The event does not result in a plant trip (manual or automatic) or require a plant shutdown.
2. The event develops slowly (DSA and PSA fault sequence figures show event progressions) and it is shown there is demonstrably conservative time margin available to eliminate the source or to provide adequate response.

The following events were assessed qualitatively:

- Release of chemicals from on-site storage
- Turbine-generated missiles
- Other internally-generated missiles from pressure vessels, valve failures, control rods

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- High speed rotating equipment (e.g., fan blades, turbines, pump impellers, fly wheels, coupling bolts)
- Internal fires are treated in the same manner as external events for BWRX-300 and are thus “screened-in” as part of this analysis alongside external hazards

Most explosions sources are within the Internal Fires PSA scope. Ignition of the following components with explosion potential is considered:

- Batteries
- Diesel generators
- Switchgear
- Hydrogen tanks
- Miscellaneous hydrogen sources
- Offgas/hydrogen recombiner
- Transformers
- Transient combustibles
- Boilers
- Turbine auxiliaries

Explosions are qualitatively screened based on Criteria 1.

The quantitative screening criteria from RA-S-1.4-2021 (Reference 15.6-2) are applied. A hazard screens quantitatively if:

1. Based on absolute risk contribution, an event sequence family subject to screening does not exceed the selected risk significance criteria and has mean occurrence frequencies less than $1\text{E-}7/\text{plant-year}$, as estimated using a bounding or demonstrably conservative analysis.
2. The total contribution of all screened out event sequence families does not exceed 1% of the cumulative risk targets included in the absolute risk significance criteria.

The qualitative assessment of the above hazards is summarized in Table 15.6-1.

Quantitative Screening

The quantitative screening criteria for external hazards are also used for internal hazards.

To determine the CDF from turbine-generated missiles, the IE frequency, and Conditional Core Damage Probability (CCDP) are determined.

The CCDP was calculated by assuming a LOPP and failure of diesel generators and was estimated to be approximately $3.4\text{E-}7$. Multiplying the CCDP by the IE frequency, $1\text{E-}4/\text{yr}$, yields a conservative CDF of $3.4\text{E-}11/\text{yr}$. The $1\text{E-}4/\text{yr}$ frequency is conservative because it does not credit the fractional probability of the turbine missile striking and then damaging the target equipment.

Turbine-generated missile hazards meets the screening criteria, so this hazard screens quantitatively.

Internal flooding hazards are treated explicitly in the PSA. No quantitative screening is performed for internal flooding.

The results of the Internal Hazard quantitative screening analysis are presented in Table 15.6-2.

Hazard Combinations

Combinations of External Hazards and Correlated Hazards

The combinations of external hazards or correlated hazards (referred to below as combinations of hazards) are analyzed for those hazards that have the potential to combine, e.g., heavy winds causing the loss of the external grid, snowstorm drifts blocking the diesel generator combustion air intakes and frazil ice blocking water cooling.

Categories of Hazard Combinations

Consequential Hazards: Combinations of consequential (or subsequent) events are defined consistent with IAEA, SSG-64, Appendix I (Reference 15.6-11):

An initial event, e.g., an external or internal hazard, results in at least one further hazard, e.g., another external or internal hazard. Event chains of three or more hazards subsequent to each other are possible.

For the assessment of external hazards combinations, the initial event impacting a nuclear installation is either a natural hazard such as an earthquake or an external flooding, or a human-induced one, such as an accidental aircraft crash or an explosion pressure wave (blast) originating from outside the plant boundary.

The following hazard combinations involving external hazards are possible to occur:

1. Natural hazard and consequential external hazard covering:
 - a. Natural hazard and consequential natural hazard (e.g., earthquake and consequential tsunami)
 - b. Natural hazard and consequential human-induced hazard (e.g., extreme wind and consequential aircraft or helicopter crash)
 - c. Natural hazard and consequential internal hazard (seismic and fire)
 - d. Human-induced hazard and consequential internal hazard (aircraft crash and internal explosion)
2. Correlated Hazards: Combinations of correlated hazards are defined consistent with IAEA, SSG-64, Appendix I (Reference 15.6-11):
 - a. Two or more external and/or internal hazards occur as a result of a common cause initiator. The common cause can be any anticipated event including an external hazard or might be due to an unanticipated dependency. The two or more hazards correlated by this common cause could occur simultaneously.
3. The following hazard combinations involving external hazards may occur:
 - a. A natural hazard or hazard source as common cause for other external and/or internal hazards and/or internal events. Typical examples include an earthquake as the common cause for landslide, external fire, internal fire, internal explosion, and station blackout, as potentially correlated events, or tsunami as the common cause for external flooding, internal flooding, and internal fire as potentially correlated events.
 - b. A human-induced hazard (or hazard source) as common cause for other human-induced and/or internal hazards. Examples are an accidental aircraft crash of a large commercial passenger aircraft as the common cause for an external

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explosion (outside the plant boundary), a plant internal fire an internal explosion, and a heavy load drop correlated by the aircraft crash, or an electromagnetic interference outside the plant boundary as the common cause for a plant internal electromagnetic interference and internal fire as the two correlated hazards.

4. Independent Hazards: Combinations of unrelated (or independent) events are defined as follows consistent to IAEA, SSG-64, Appendix I (Reference 15.6-11):
 - a. An initial event (e.g., an external or internal hazard) occurs independently from, but simultaneously with another hazard without any common cause. The term 'simultaneously' in this case does not mean that the hazards occur exactly at the same time but rather that the second hazard occurs before the consequences of the previous hazards have been completely mitigated.
5. The following event combinations involving external hazards may occur:
 - a. Natural hazards and independently occurring natural hazards. Typical examples include longer duration riverine flooding and independently occurring extreme weather conditions, such as extreme wind.
 - b. Natural hazard and independently occurring human-induced hazard. Examples are a longer duration hydrological hazard and an independently occurring industry accident (such as an explosion, release of hazardous substances, etc.).

Identification and Screening of Hazard Combinations

Event combinations used to develop a list of potential combined hazards is initially based on the identified individual hazards. The number of individual hazards identified to occur at a given plant location is already high. Building combinations of all possible individual hazards result in a number of combined hazards, which is not manageable without applying a systematic approach.

Identification of consequential hazards begins with the unscreened single hazards and determining those hazards that may occur as a result of the initial hazard. This can be performed using a matrix approach, listing the hazards graphically and showing the combinations of two hazards (so-called first order combinations) where the possible combinations are identified. The combined hazards can be further reviewed based on this initial identification to determine additional hazard combinations possible (second-order combinations) or additional identified combinations.

For combinations of correlated hazards, it is not directly possible to build a matrix for these combinations. The analyst starts the combined hazard screening by identifying potential common causes for those individual (single) hazards not screened out for the plant/site and builds a tree-type structure (or similar) for all hazards correlated by such common causes representing the roots of the branches. With such a tree-type structure, higher order combinations of correlated hazards can be built and undergo a screening process.

For the third category of combinations of independent hazards, the same two-dimensional matrix structure is used. The possibility of an event combination is demonstrated by a color change or other designation in the corresponding matrix field. The matrix involves the identification of hazards that may impact the plant/site for a time period longer than 3 days. The likelihood is included in the matrix based on the duration of the first hazard and the likelihood of the second hazard occurring during that time period. Combined events are identified by analyzing the correlations and the effects on the plant. The analysis of possible correlations (dependency) between events is made by assessing the physical bases of the phenomena, observed data, actual operating events, and general knowledge of local conditions.

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Expert judgment and rough quantitative analysis are used for estimating correlations. The observed data for intensities relevant to external events PSA is sufficient for order of magnitude correlation estimates.

For example, high wind often results in high water level and organic material in water (e.g., seaweeds, kelp, algae, Asiatic clams, mussels, etc.). The combination may simultaneously endanger offsite power and the diesel generator cooling water intake; therefore, it is analyzed as a combined event.

Extreme wind velocities are measured in winter and could be associated with snowfall resulting in possibility of simultaneous LOPP and loss of the diesel generators due to snowdrift blockage of combustion air intake.

Simultaneous high air and seawater temperature could also endanger plant equipment, e.g., instrument room cooling at units with diverse cooling system heat sinks. Identified hazard combinations are screened similar to single hazards using the process discussed previously.

Evaluation of Risks Associated with Unscreened External Hazards

For unscreened external hazards and identified combinations of external hazards, a similar model development and quantification processes described previously are adapted to evaluate external hazard risks. The process utilized for hazards combinations is discussed below.

Several external hazards from transportation and nearby facilities are evaluated. These external hazards are evaluated with bounding assumptions and can be modified with plant-specific inputs:

- Airports and airways hazards
- Industrial accidents
- Pipeline accidents
- Hydrogen storage failures
- Transportation accidents

Process for Selection, Screening and Analysis of Combinations of Hazards

Figure 15.6-2 provides an overall process for selecting, screening, and analyzing combinations of hazards. The quantification of the hazards is performed similar to the quantification of the individual hazards with consideration of the combined hazard on the SSC fragilities, the likelihood of the combined hazards, and the inclusion of these factors in the PSA model.

The process for the specific modeling depends on the hazards, and the supporting single hazard PSA modeling. During the PSA quantification, the quantification steps may be slightly different if the combined hazards PSA results are calculated together with or separately from the single hazards PSA model quantification. For example, if a seismically-induced flooding event PSA model is developed and quantified, it can be quantified as a part of the Seismic PSA or separately where only the accident sequences involving seismically-induced flooding are quantified. There are no requirements in either IAEA-TECDOC-1804, "Attributes of Full Scope Level 1 Probabilistic Safety Assessment for Applications in Nuclear Power Plants" (Reference 15.6-13) or the ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12) to quantify single and combined hazards separately or together. There are advantages of long-term model maintenance in developing and analyzing the combined hazards using a single model. The quantification of each single and combined hazard avoids double counting the results. For example, if the CDF of a single hazard PSA model such as a seismic PSA provides a value of $1\text{E-}05$ / reactor year, and the combined hazards PSA model for the combination of consequential hazards seismically-induced internal flooding provides a CDF of $5\text{E-}06$ /reactor year, there is likely some double counting if the final

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results are equal to the sum of the two CDF values. In this case, a realistic assessment of the single hazard PSA is based on accident sequences where a flooding event does not occur.

The quantification of the combined hazard PSA is performed in a similar manner as the individual hazard PSA, accounting for the conditional probability/frequency of the combined hazard magnitudes. As previously discussed, this can be applied in a variety of approaches, but often specified magnitude ranges or bins are used. The analysis of hazard combinations magnitude ranges or bins include variation of the various hazards and the conditional probability of each hazard in combination with the others.

Internal Hazard Combinations

Heavy load drop-induced floods and structure damage are addressed as part of the heavy loads analysis in the low power/shutdown PSA evaluation.

Turbine-generator missile-induced fires/explosions are addressed in the fire PSA. Pipe whip and jet effects are addressed in the internal flooding PSA.

External Hazard Combinations

The external hazards combinations were identified and screened. The high wind combinations are covered by the high wind PSA.

The BWRX-300 PSA scope includes considerations of radioactive sources outside the core. Since one goal of the PSA is to ultimately evaluate the level of safety to the surrounding populace, only those radioactive sources containing sufficient radioactivity that, if released, would pose a significant health hazard outside the site boundary.

The only significant radiation source in the primary containment is from the reactor core. The reactor core is treated in the PSA. The FMCRD system also contains a radiation source but is not expected to pose an offsite threat.

In the RB, the fuel pool is assumed to contain during some portion of the plant lifetime sufficient radioactivity to pose a public health hazard outside the site boundary. This radiation source is treated in the PSA.

Radioactivity deposited on systems in the plant is not liberated in sufficient quantities to pose a public health hazard. This includes systems such as the fuel pool cooling, reactor water cleanup, main steam, feedwater, and Heating, Ventilation, and Air Conditioning (HVAC) systems.

The radioactive waste system does not contain radioactivity in sufficient form or quantity to pose a public health hazard in the form of a small or large release defined by the PSA.

The potential radionuclide source term from severe damage to spent fuel casks was examined in USNRC NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants." (Reference 15.6-14). USNRC NUREG/CR-6451 (Reference 15.6-14) considered tornado missile impact the limiting hazard and the tornado wind speed required to generate a missile with enough force to compromise a cask has never been observed. The judgment that tornado impacts are the limiting physical hazard to spent fuel dry casks is adopted for the BWRX-300 design. Regardless of the ruggedness of the casks, a conditional radionuclide source term was developed for a cask event where all rods of a fuel assembly are compromised. The release fractions for iodine and cesium are less than $1.5\text{E-}5$ and $2.25\text{E-}5$ respectively, from the damaged spent fuel assembly in the cask. The Level 2 Probabilistic Safety Assessment estimates that the release fraction for I-131 constitutes a small release of $1.14\text{E-}3$ and release fraction for Cs-137 constitutes a large release of $5.20\text{E-}4$ in terms of fractional inventory of the entire core.

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Because hazards to spent fuel in dry casks do not result in fuel damage or significant radionuclide releases, dry casks are screened from treatment in the PSA.

15.6.1.2.2 External Hazards

The BWRX-300 PSA is an all-hazards PSA where external hazards are evaluated for potential PSA development.

The methodology examines the hazards from Appendix 2 "Questionnaire on External Events PSA (Other Than Seismic) of NEA/CSNI/R(2009)4," "Probabilistic Safety Analysis (PSA) of Other External Events Other Than Earthquake," (Reference 15.6-15) and applies screening to each hazard according to the rules of ANSI/ASME/ANS RA-S-1.1-2022, Appendix 2 (Reference 15.6-12):

1. A hazard qualitatively screens if demonstratively conservative assessments find the hazard does not impact the plant or is subsumed into a more frequent or more impactful event. Any one of the following qualitative screening criteria may be used:
 - a. The event is of equal or lesser damage potential than events for which the plant is designed. This requires an evaluation of the plant design basis in order to estimate the resistance to a particular (external) event.
 - b. The event has a significantly lower mean frequency of occurrence than similar events included in the PSA and could not result in worse consequences in those events.
 - c. The event cannot occur close enough to the plant to effect it.
 - d. The event is included in the definition of another event.
2. The frequency of the event is sufficiently low in comparison with the probabilistic limits defined for the release category frequencies so that it does not need to be included in the PSA. Hazards that do not screen qualitatively are subjected to quantitative screening. A hazard screens quantitatively if:
 - a. Based on absolute risk contribution, an event sequence family subject to screening that does not exceed the selected risk significance criteria and has mean occurrence frequencies less than $1\text{E-}7/\text{plant-year}$, as estimated using a bounding or demonstrably conservative analysis, and.
 - b. The total contribution of all screened out event sequence families may not exceed 1% of the cumulative risk targets included in the absolute risk significance criteria
3. Hazards that screen neither qualitatively nor quantitatively are addressed by detailed PSA. A decision can be made to bypass screening of a hazard and go directly to detailed PSA development.

The NK38-REP-03611-10043 R003, "Hazards Screening Analysis – Darlington" (Reference 15.6-26) site environmental conditions were used for external hazards screening:

1. For the screening assessment of high water level, a 100-year flood is conservatively assumed to flood the site and allow water to enter the RB.
2. The BWRX-300 Diesel Generators (DGs) are air-cooled, and thus operation may degrade or cease in extreme ambient temperature conditions.
3. Given the advent of weather extremes due to global climate change, an extreme high temperature condition of 37.8°C or more is assumed to occur at the site once per 20 years.

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4. Safety SSCs exposed to ambient environmental conditions and required to function in local extreme meteorological events are assumed to be designed for -40°C.
5. For extreme high or low ambient temperatures, a LOPP is assumed to occur due to high energy demand.
6. The DGs, which are air-cooled, are assumed to be unavailable given extreme temperatures at or below -40°C that may lead to station blackout.
7. A northern climate, Toronto Canada, is selected as a representative location for snowfall at the Darlington site. Historical heaviest single day snowfall data for Toronto for the past 169 years indicate that the maximum snowfall episode was 48.3 centimeters on December 11, 1944. Heavy snowfall did not extend beyond or prior to this single day. It is assumed that 48 centimeters is below the design snow load for non-Seismic Category-I (non-SC-I) structures and below the air intakes for the DGs.
8. It is assumed that if heavy snowfall is forecast for multiple days, operator action will initiate shutdown procedures if the integrity of plant structures due to heavy loading is threatened.
9. BWRX-300 structures are designed to withstand credible accumulations of hail stones.
10. It is assumed that all outdoor structures of the BWRX-300 are designed to withstand white frost.
11. It is assumed that the plant is not susceptible to sand accumulation that may affect SSCs.
12. The BWRX-300 plant design protects electrical systems functions from voltage transients, including those caused by lightning.
13. For explosions, it is assumed that the Darlington BWRX-300 site is situated beyond the safe screening distance for explosion 700m and vapor cloud explosion 460m to the Saint Mary's Cement Plant.
14. If an explosion occurs in the vicinity of the site, a loss of all structures other than the reactor building is assumed including the MCR as well as LOPP. The DGs are assumed to remain available as they are in a bunkered area and air-cooled (assumed). Electrical cables and control cables are assumed to not be impacted (e.g., they run underground).
15. For marine transportation accidents, a restricted zone is assumed to exist surrounding the plant intake channel and diffuser that does not allow commercial ships to approach the site shoreline. Therefore, marine vessel explosions will not harm the site.
16. It is assumed that the Darlington site is a safe distance from blasting.
17. The Darlington site is assumed situated beyond the screening distances for explosions after pipeline accidents.
18. Chemical release can occur due to the rail line accidents near the site. The control room HVAC has design provisions to address a chemical release.
19. It is assumed that the site is situated beyond the screening distances for chemical releases after pipeline accidents, and chemical releases to water. There are three pipelines (2 natural gas and one petroleum) that are located beyond the screening distance for the Darlington CANDU reactors.
20. BWRX-300 I&C systems are designed to prevent magnetic disturbance from significantly impacting them.

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21. The RB is designed to withstand large aircraft impact. The RB utilizes steel bricks construction, and the design parameters for the Steel Bricks™, such as steel thickness, steel ductility, and concrete thickness, is selected to withstand RB perforation from aircraft and aircraft engine impact.
22. The potential exists for a direct hit from an aircraft engine to one of the RB exterior doors and subsequent penetration. The CCDP given for an airplane crash is 1.0.
23. It is assumed that the BWRX-300 at Darlington site includes an evaluation that demonstrates no historical evidence of subsidence.
24. It is assumed that BWRX-300 building foundations are designed to address potential soil frost and that pipes are buried below the frost line.
25. The Darlington site is not located near volcanoes. There is no concern for ash to fall on the plant.
26. Darlington topography is relatively flat surrounding the site. No avalanche potential exists.
27. The Darlington site includes consideration of waterline topography with relatively flat topography.
28. The land coverage surrounding Darlington is expected to have minimal vegetation, limited to bushes, and mostly paved in the areas proximal to site buildings. It is assumed there is limited numbers of trees, and that the area is mostly paved and cleared to preclude the possibility of external fires or tornado missiles, causing damage to equipment, or impacting control room operations. As a result of the sparse vegetation, external fire propagation to the site is not credible.
29. It is assumed that plant procedures are in place to preclude damage to underground equipment, such as piping or tanks, from excavation work.
30. It is assumed that plant procedures are in place to preclude damage to underground equipment, such as piping, from heavy transportation within the site.
31. Missile impacts from military activity are assumed to be bounded by the airplane impact hazard.
32. It is assumed that the Darlington BWRX-300 plant will not be contiguous with another existing plant.
33. Lake Ontario that is part of the Darlington site boundary is assumed not to have strong lake water currents.
34. Low water level slowly develops. It is assumed that the plant would be shut down and reach a safe stable state prior to the loss of circulating water or plant cooling water.
35. It is assumed that the Darlington BWRX-300 site waterline topography allows water to drain away from the plant.
36. Surface ice will not affect plant operation. The coolant intake and discharge lines at the intake structure are assumed to be located on the water body floor.
37. It is assumed that the Darlington BWRX-300 site is located a safe distance from shipping lanes.
38. For quantitative screening in which offsite power is lost, the CCDP calculations assume that the offsite power loss is not recovered that is used in the internal events PSA modeling.

OPG has used drones for the Darlington CANDU Vacuum Building inspections. The impact of drones hitting the BWRX-300 Structures Systems and Components (SSCs) is bounded by small aircraft crash.

The USNRC has reviewed impact of drones on U.S. Nuclear Power Plants (Backgrounder – Drones and Nuclear Power Plant Security (Reference 15.6-16)). Their assessment states:

“The technical analysis concluded that U.S. nuclear power plants do not have any risk-significant vulnerabilities that could be exploited by adversaries using commercially available drones to result in radiological sabotage, theft, or diversion of special nuclear material (essentially the reactor fuel)”. Thus, drones are screened out.

Results

The external hazards qualitatively screened out from the PSA are shown in Table 15.6-3. The external hazards quantitative assessment results are shown in Table 15.6-4.

15.6.1.3 Level 1 PSA

15.6.1.3.1 Level 1 PSA At-Power

One of the first and basic steps in a PSA is the identification and quantification of the IEs used in the sequence analysis. IEs have historically been broadly classified as either “internal” or “external” hazards. An IE may result from human causes, equipment failure from causes internal to the plant (e.g., hardware faults, floods, or fires), or external to the plant (e.g., earthquakes or high winds), or combinations thereof. Internal events PSA assesses events that are caused by systems or components located within the site structures. An exception of this is LOPP that is analyzed in the internal events PSA. Internal and external hazard induced IEs (e.g., seismic events, internal floods) and IEs during shutdown excluding sabotage are discussed in other sections in this report:

- Probabilistic Fire Analysis (internal fire events)
- Probabilistic Flood Analysis (internal flooding events)
- High Wind Risk (including tornado events)
- Seismic PSA (seismic events)
- Shutdown Risk (shutdown IE for internal events)
- Spent fuel damage

15.6.1.3.2 Initiating Events Identification

To develop a comprehensive accounting of postulated IEs, a systematic approach is used to identify events that challenge normal plant operation and require successful mitigation to prevent radionuclide release. IAEA-TECDOC-1804 (Reference 15.6-13) provides the following IE definition: “an event which could directly lead to core damage or challenges normal plant operation and requires successful mitigation to prevent core damage”.

A comprehensive identification of IEs includes a review of existing BWR PSAs, including the ESBWR, and generic sources. Generic lists of IEs are included in multiple reference documents such as IAEA-TECDOC-719, Purpose of Probabilistic Safety Assessment,” (Reference 15.6-17), USNRC NUREG/CR-5750 and USNRC NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants.” (Reference 15.6-19) update 2015.

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IEs identified typically include transients of various types, LOCAs, and support system initiators. IEs under low power conditions are also identified and are listed in Table 15.6-5.

All system failures that disturb plant operation (e.g., front-line and support systems) are reviewed, except for those already identified as the source IE based on other analyses.

In addition, planned and unplanned manual shutdowns which seldom place demands on any standby safety equipment are treated as IEs because of their high frequency and because they represent changes in operating states that result in demands on available equipment to reach a safe shutdown condition.

Support system initiators are defined as unplanned normal shutdown resulting from loss of a support system with dependent failures. Related to support system initiators are CCF that are equipment failures that cause an IE and simultaneously degrade equipment credited for mitigating the IE (e.g., failure of balance of plant equipment that could cause a plant trip and preclude use of the normal power conversion system). CCFs are systematically identified and treated in the PSA.

Human error IE is considered for these analyses.

An example list of the general damage definitions of internal IE are shown in Table 15.6-6, which is taken from IAEA-TECDOC-719, Table 3.9 (Reference 15.6-17). This list is supplemented with the other generic sources previously discussed in developing a comprehensive IE list.

15.6.1.3.3 Initiating Events Screening

The requirements for screening potential IE candidates from the PSA are discussed in IAEA-TECDOC-1804 (Reference 15.6-13) and the ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12).

Based on the references in Subsection 15.6.1.3.2, the IE candidates identified are excluded from further consideration if they meet one of the following criteria:

1. The event does not lead to the IE as defined in the PSA.
2. The event does not correspond to the scope of the PSA.
3. The frequency of the event is less than the truncation value related to the accident sequence frequency, and the event does not involve an interfacing systems LOCA, containment bypass, or RPV rupture. For these events, the truncation value is at least one order of magnitude lower than the truncation value accepted in the PSA.
4. The resulting reactor shutdown during at-power is not an immediate occurrence. The event does not require the plant to transfer to shutdown conditions until a defined amount of time has elapsed, the condition is detectable before plant systems are required to respond, and there is a high degree of certainty (based on supporting calculations) that the condition can be detected and corrected before normal plant operation is affected (either administratively or automatically).

The IE screening is limited to those events that do not lead to an IE defined in the PSA, and those events where a shutdown occurs prior to the conditions being corrected. Screening is carefully and conservatively applied, especially early in design where the impact for an event may not be fully understood without design details that are fully developed.

15.6.1.3.4 Accident Sequence Analysis

The event trees developed for BWRX-300 are based on the IEs developed in the IE analysis. The event tree models include the set of safety functions needed to mitigate each IE. The accident

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sequence analysis ensures that the response of the plant systems and operators to an IE is reflected in the risk assessment so that:

1. Operator actions, mitigation systems, and phenomena that influence or determine the course of sequences are appropriately included in the accident sequence model and sequence definition.
2. Plant-specific dependencies due to IEs, human interfaces, functional dependencies, environmental, and spatial impact, and CCFs are reflected in the accident sequence (event tree) structure.
3. The individual function successes, mission times, and time windows for operator actions for each critical safety function modeled in the accident sequences reflects the success criteria evaluated in accordance with the acceptance criteria attributes following.
4. End states are clearly defined as either core/fuel damage or successful prevention with the capability to support the interface between Level 1 and Level 2 PSA.
5. The accident sequences are defined for the selected set of IEs, POSs, and times that a POS can occur.

The resulting accident sequence analysis addresses:

- General assumptions relating to all event tree development
- Sequence end states
- Definition of the type of models produced (e.g., small event trees and large fault trees) and level at which the event tree headings are defined (safety function, system, train)
- The BWRX-300 event trees are generally small event trees developed using the CAFTA software discussed previously. Event tree headings are largely based on top-level system functions with functional fault trees added to connect the models to system and train failure events
- Requirements for developing event sequence diagrams
- Interface between the IE, success criteria, human reliability, system modeling and data analysis tasks

The event sequence analysis meets the ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12).

Methodology

Acceptance Criteria

A formal definition of core damage is defined and justified before the function overall success criteria is established.

Core damage is defined in terms of physical processes, phenomena, and failure mechanisms. The accident analyses and past experiences for similar plants are used in deriving the core damage definition. The physical plant parameters (e.g., highest node temperature) and associated acceptance criteria limit values (e.g., percentage of cladding thickness oxidized) used in determining the core damage definition. The parameters selected in determining core or fuel damage is practical, consistent, and realistic. Best practices and OPEX, conservatism in computer codes, sophistication of models, and sufficient margin for uncertainties are parameters considered in defining core or fuel damage.

The success criteria supporting each event tree and key safety functions are the minimum requirements necessary to achieve safe, stable conditions (i.e., to protect the fuel and prevent

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radionuclide releases to the environment). Safe conditions are determined by meeting the following key safety function success criteria, and stable conditions are determined by maintaining each key FSFs for long-term operation:

1. Reactivity Control: Achieve subcriticality and maintain the reactor in a subcritical state (combined with adequate core cooling and/or containment heat removal functions) so that core damage and containment failure are avoided.
2. Core Cooling: Adequate core cooling is provided to prevent core damage and maintain the reactor in a safe-and-stable condition. An example of the overall success criteria for core cooling is ensuring the PCT does not exceed 1200°C and cladding oxidation does not exceed 15%.
3. Containment System Integrity: Ensure that the containment pressure does not exceed the ultimate containment failure pressure, and containment isolation occurs when required.

Success in achieving the FSFs is defined in Chapter 15, Section 15.2.

Other safety functions are developed as needed. For example, the RCPB success criteria is developed and modeled based upon the analyzed maximum pressure increase in the reactor coolant.

Accident Sequence Development

The role of an event tree is graphical expression of IE groups, success criteria, and sequence end points. The basic concept is preparing an event tree for each IE group defined in IE analysis task.

IE group is studied and organized based on following policy based on IAEA-TECDOC-1804 (Reference 15.6-13): "For each IE group for internal events, internal hazards, and external hazards for each POS the accident progression for all sequences is identified and justified. For each IE group the accident sequence models are developed. As models explicitly address realistic plant behaviour in response to IE in terms of normal plant systems operation, operator actions, and mitigation systems that support the key safety functions necessary to achieve a stable safe state."

End States of Event Sequences

The event trees identify the potential sequences that may lead to radionuclide release. Many of the sequences have common characteristics that challenge the containment fission product barrier. These sequences are grouped into damage classes that are analyzed in the release portion of the PSA. The end states of the event sequences developed are defined to facilitate the containment performance analysis and provide the link between core damage and a release category.

Event sequences are then qualitatively analyzed to determine which sequences lead to core damage end states.

The core damage sequences are grouped together based upon the overall challenge to the containment barrier and defined as:

1. OK: The core is successfully cooled, and the containment is intact. There is no core damage in these events.
2. CD I: The containment is intact when core damage occurs and the RPV is at low (or controlled) pressure.

3. CD II: The containment is breached, either due to overpressurization or venting, while the core is successfully cooled. Core damage results from failure of long-term heat removal core cooling.
4. CD III: The containment is intact when core damage occurs and there is high RPV pressure at the time of RPV failure.
5. CD IV: Core damage results from an accident sequence with initial failure of effective reactivity control (CCF failure of hydraulic scram or FMCRD run-in, control rod binding, etc.). This potentially affects containment more severely than CD I or CD III because more energy is deposited into the containment prior to RPV failure. All CD IV end states are treated as CD I or CD III depending on the RPV pressure without affecting the results of the containment analysis. This end state has been retained in this scoping analysis to readily allow sensitivity analyses for reactivity control.
6. CD V: The containment is bypassed at the time of core damage.
7. CD VR: Core damage occurs due to RPV ruptures in the lower or mid-vessel regions.

Sequences where core damage is expected to occur very late are appended with an 'L'. An example of this occurrence is when CRD flow is being relied upon to avoid core damage, but eventually fails due to inventory depletion.

15.6.1.3.5 Success Criteria Analysis

The main objective of success criteria formulation is determining for given IEs what represents a successful or unsuccessful plant response and translating this information into detailed plant system and operator action success criteria accomplished for at-power or low power/shutdown operation. The BWRX-300 success criteria analysis is integrated with the accident sequence analysis.

Success criteria are defined for several different levels: The highest level of success criteria development determines what mitigates an IE (i.e., aversion of core damage). Using the qualitative description of core damage (e.g., gross fuel/cladding degradation due to failure to adequately remove fission and/or decay heat), a parametric description (e.g., peak core nodal temperature) generated.

Success criteria are developed supporting the functions required to mitigate core damage. An assessment of front-line systems identifies mitigating functions and determines the requirements to fulfill those functions. Best estimate analyses use computer codes or other acceptable calculations in performing this task. Examples include determining how many trains of a system performs a function for IE sub-groups.

Supporting success criteria is determined once the functional success criteria is identified. This includes mission times calculations and human actions, inventory availability determination, and other system or train-level success criteria development.

Methodology

Definition of Core Damage

The definition of core or fuel damage is a key aspect in determining the functional success. Core damage is defined by the two key physical parameters: PCT and fuel cladding oxidation.

The above definition and criteria are consistent with the definition in PSA Standard ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12) as "uncovery and heat-up of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects."

Functional Success Criteria

The functional success criteria defines the minimum performance that differentiates the end state when that function succeeds from the end state reached when the function fails. With the parameters that determine core damage and decay heat levels defined, specific success criteria are added defining the necessary capability of SSCs and operator actions in accident sequences.

A mission time is determined to achieve a stable end state after an IE for the accident sequences. As a first approach, a general mission time of 24 hours is assumed for Best Estimate (BE). A longer mission time is used for some accident sequences to achieve stable plant conditions.

Success criteria analysis considers the available inventories needed to support each system or function. The available inventories of fuel, water, or air in tanks etc., are compared with those required to support each success criterion and the model reflects the comparison results.

Analysis models and computer codes are used to model the conditions and phenomena that determine the success criteria. The plant model and parameters used for thermal-hydraulic analyses provides sufficient resolution and reflects the actual design and operational features of the plant.

For each accident sequence condition, the applicability ranges and conditions for each analysis code are reviewed. An assessment is performed ensuring that all accident sequences are within the applicability of the codes used in determining the success criteria. The use of all codes is justified based on this comparison.

Each success criterion in the analysis is based on either a plant design parameter or a thermal-hydraulic calculation. When a success criterion is used to characterize a range of conditions, the limiting parameters are chosen to represent all cases. In addition, representative thermal-hydraulic analyses and offsite dose consequence analyses are run to validate the appropriateness of the success criteria.

15.6.1.3.6 System Analysis

The system analysis is performed for each plant system represented in the IE analysis and accident sequence analysis for each POS in such a way that:

1. Each safety function in accident sequence models and system models are developed that account for the success criteria.
2. System-level success criteria, mission times, time windows for operator actions, different initial system alignments and assumptions provide the basis for the system logic models reflected in the model. A reasonably complete set of system failure and unavailability modes for each system is represented.
3. Human errors and operator actions that influence the system unavailability or the system contribution to accident sequences are identified for development as part of the HRA element.
4. Intra-system dependencies and intersystem dependencies including functional, human, phenomenological, and CCFs that influence system unavailability or the system contribution to accident sequence frequencies are identified.

Other objectives include:

- Considering credible failure modes
- Modeling each failure mode impact on system performance
- Including support system failure modes in the front-line system fault trees

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- Creating linked fault tree models that allow resolution
- Creating fault tree models where cutsets are clear and concise

Methodology

The BWRX-300 system analysis includes systems modeled in the BWRX-300 PSA, which correspond to the functional headings described in the event sequences plus any support systems needed to accomplish those functions. The system modeling methodology follows the guidance in USNRC NUREG/CR-2300, USNRC NUREG/CR-4550 and USNRC NUREG/CR-4551 series. USNRC NUREG-0492 provides additional guidance on fault tree modeling, current information for the system studies, and latest results.

The analysis for each system begins by defining the system functions, system operation modes and alignments, establishing system-level success criteria (in conjunction with the event tree development that addresses operability and survivability), and defines the system boundaries. All system components and system connections that are modeled are described in simplified drawings.

For systems that operate in several different modes or POSs, each mode and the functions performed, are explained. Only those functions that contribute to plant risk defined by the accident sequences are included. The interfaces and system connections of the system with other systems are described. This also includes the support systems and relevant system components required for the operation of the system. Any other relevant aspects of the system operation including system trips, interlocks and procedural restrictions during accident conditions are also addressed.

Necessary operator actions are identified for systems where manual initiation is necessary or manual backup is credited in the PSA. The post-initiator HFEs (Type C) are identified based on operational procedure. Significant pre-initiator HFEs (Type A) are identified by investigating test and maintenance procedures related to the SSCs modeled in fault trees and are identified as part of the system analysis.

The success criteria of front-line systems for each IE are identified in the success criteria analysis task. System success criteria of support system are obtained by following success criteria for front-line systems. Based on these success criteria, necessary system functions are identified. Operability relative to trip setpoint limits and adverse environmental effects impacting all system components survivability are addressed. Any operator interventions that mitigate adverse environmental effects is proceduralized.

Systems models are developed that include all component failure modes and unavailability factors that lead to failing to achieve the system function defined by the system success criteria. Component failures that would be beneficial to system operation are not included in the model.

Database for component failure probability has two general types of failures. One is a demand failure, and the other is a rate failure based on the amount of time a component needs to operate. The reliability models considered to support BEs are not discussed in detail here, but involve exposure-rate for demand failures, exposure-rate or periodic test models for rate failures, and models that account for monitoring and repair. The data that supports the parameters listed are discussed in the data analysis task.

The documentation of the system analysis describes the processes used in modeling and quantification, generic modeling assumptions and is constructed considering future updates and uses in risk-informed applications.

15.6.1.3.7 Fault Trees

One fault tree for each system function is included in the PSA model. If multiple trains perform the function, then each train is modeled. All the logic for a particular system is contained in a single fault tree file.

The fault trees are constructed using the gate and basic event naming conventions. The fault tree database is stored in the CAFTA database file format.

15.6.1.3.8 Data Analysis

The data analysis objective is providing estimates for the reliability parameters for the reliability models specified under the systems analysis. The reliability models are used in determining the BEs probabilities of specific equipment failures and unavailability factors.

Data analysis is used in deriving the parameter values that estimate system failure probabilities and accident sequence frequencies. These BEs used in the PSA model are outputs from the IE analysis, accident sequence modeling, and system fault tree modeling discussed previously. The PSA BEs include the IE frequencies, component unavailability, component/train unavailability factors (resulting from testing or maintenance), CCF events, and other types of events. At this stage in the design process, there is no plant-specific reliability data available. Generic data from the nuclear industry and associated uncertainty are used.

Where plant-specific reliability data are not available, mainly generic reliability data are used as discussed below under component reliability database.

In the absence of actual operating experience for the passive plants, Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) recommends using failure data for components that are most similar to those used in passive plants. Additional adjustments to the generic data are introduced after analyzing the test and maintenance intervals and the environmental factors.

The data and basis for test and maintenance unavailabilities are based on bounding generic values and OLC allowed outage times. Because of limited operating experience and the lack of plant-specific data, the development of failure rates for equipment reflects appropriate characterizations of the associated uncertainties.

The level of redundancy in passive plant design and few critical support systems leads to increased focus on CCFs. CCFs are modeled for the components of the same size and in the same operating environment. The CCF data from the USNRC CCF (Reference 15.6-18) is used in this analysis. Generic CCF factors are used when component-specific data are not available.

Methodology

Component Reliability Database

The parameters are estimated using data and other information that is compatible with the definitions of the BEs. The task procedure either contains the definitions of the events or does so by task procedures for systems analysis and IEs. The component boundaries and the interfaces among connected components are identified. Because a component is required for different operating conditions, BEs representing different failure modes is included in the system model.

Components are grouped into population groups for parameter estimation. The rationale for grouping components into a homogeneous population for parameter estimation considers the design, environmental, functional, and operational conditions of the components in the as-built

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and as-operated plant. For parameter estimation, components are grouped according to type and to the detailed usage characteristics.

In this analysis, generic reliability data-based USNRC NUREG/CR-6928 (Reference 15.6-19) are used. When data from this USNRC NUREG/CR-6928 (Reference 15.6-19) is not available for a modeled failure mode, data is used from the following resources in this order:

- EPRI TR-016780-V3R8, "T-Book: Reliability Data of Components in Nordic Nuclear Power Plants, 6th Edition"
- USNRC NUREG/CR-4550, "Analysis of Core Damage Frequency From Internal Events: Methodology Guidelines"
- USNRC NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide"
- USNRC NUREG/CR-1740, "Data Summaries of Licensee Event Reports of Selected Instrumentation and Control Components at U.S. Commercial Nuclear Power Plants"
- EGG-SSRE-8875, "Generic Component Failure Data Base for Light Water and Liquid Sodium Reactor PRAs"
- Engineering judgment

Dependent Events and Common Cause Failure Analysis

Dependent events challenge redundancy or diversity and ultimately increase the unavailability of a system. Four dependency types are analyzed in the PSA analysis:

1. Plant-level functional dependencies: Plant-level functional dependencies lead to functional failures as a direct result of a shared IE.
2. Intersystem dependencies: The dependencies of BWRX-300 systems are carefully analyzed in the PSA analysis. Physical and functional intersystem dependencies are incorporated into the system fault trees by using transfer gates. Physical dependencies include spatial interactions, influences for common environment, or impacts from adjacent SSCs. These physical dependencies are important when modeling the impacts from internal and external hazards(fire and flood).
3. Human action dependencies: Dependencies due to human actions (testing, maintenance, diagnostic errors, incorrect calibration of sensors or instrument), which could affect manual actions or actuations of redundant systems, are reviewed in the system PSA analysis and modeled as separate events in the fault trees where necessary. Modeling of human action dependencies is discussed further in Section 15.4.
4. Intra-system dependencies: Intra-system dependencies (CCFs) can lead to multiple component unavailabilities from shared causes. The remainder of this PSAR section discusses the analysis approach for this type of dependency.

Developing the Common Cause Failure Database

Once the CCF basic events are identified, the model utilizes the CCF tool in CAFTA to automatically identify all the CCF combinations for a given CCF group and places the appropriate CCF BEs into the fault trees. CAFTA is also used to automatically quantify each of the CCF BEs.

As recommended by the CAFTA developers, common cause groups containing more than four components are treated simplistically. CCF events are developed for all combinations of one, two, and three components within the group, with all remaining failure probability assigned to a CCF event of all components failing. This is a conservative representation of the event modeling

all components failing, because the CCF probability of all components is overstated, but greatly simplifies the fault tree modeling.

When this treatment is too conservative, the calculations are completed outside of CAFTA and entered manually. In these cases, the CCF combinations are manually input to the fault trees using the parameters for the actual CCF group size and the CCF equations are directly input into the database file.

15.6.1.3.9 Human Reliability Analysis

The HRA is performed as part of the BWRX-300 PSA. Human actions play a key role in the risk assessment and may point to plant vulnerability early in the design phase that can be corrected before the design is finalized.

The HRA describes operator failure in a systematic and reproducible manner to facilitate a Human Error Probability (HEP) that is applied to the PSA model. The methodology is described below. Individual failure events generate reports that are presented in appendices to the PSA model.

Scope

The HRA covers three types of human actions:

- Those performed by maintenance or operations before an event
- Those performed by operations or maintenance to cause an event
- Those performed by operations after an event has occurred

Currently only post-initiator actions are documented given the limited plant design information.

Inputs, Requirements, and Assumptions

The primary input to the current HRA comes from a review of the success criteria and event sequences documents. These documents discuss the operator actions that are considered for the safe shutdown of the plant. The second resource is examination of the system fault trees to find actions that backup automatic functions. The third resource and the primary resource for pre-initiator actions comes from system engineers and any developed system maintenance schedules. Currently maintenance has not been considered in many of the systems and therefore pre-initiators are not included in the model.

Requirements

The requirements for the HRA analysis primarily come from the PRA Standard ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12). Due to the PSAR design development stage and emergency procedures that are developed for the final safety analysis, this analysis is not expected to meet all the requirements of the standard at this time.

Assumptions

Generic assumptions used in the PSA:

1. All human actions are driven by procedure. No actions are considered that are outside operations documentation (e.g., actions based on knowledge, or mitigating actions outside the procedure).
2. Errors of commission are screened out from the analysis because they have negligible probability compared to errors of omission.
3. All persons performing operations or maintenance are adequately trained by an accredited training program.

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4. Post-initiator actions are all conducted from the control room unless otherwise specified in the human failure event report.
5. In the current phase of design, procedures for actions are based on engineering judgment and experience.
6. The pre-initiator HRA for a component that is made unavailable due to testing or maintenance, a restoration error is always possible.
7. The control room displays, and alarm annunciators are well-designed with critical alarms for all human failure events within the safety case identified and shown on the appropriate displays, with color-coding, lighting, and flashing to guide alarm management. Information is displayed in a manner appropriate to the type of display (e.g., large screen versus control desk versus local control panel, etc.).
8. Software-based display screens and control actuations are well-designed with consideration of human factors good practice principles including grouping, labeling, and spacing. In particular, potential selection errors either in screen navigation or in equipment icons on a particular screen are minimized through effective layout, screen item design, and group demarcations.
9. The time required to perform the steps in a task is estimated in most cases based on input from personnel that have BWR PSA human reliability analysis experience, have interacted with nuclear operators and is familiar with BWRs operator actions. Wherever possible, worst-case plant conditions and competing task workload are assumed ensuring conservative estimates. In some cases, the time taken to carry out task steps that are part of a task but are not necessary for successfully completing the specific action identified in the PSA are included in the overall task required time. This is done in cases where members of the HRA team indicate that these actions are likely to be attempted. Task timings are further examined and verified in later iterations of these analyses.
10. Assumptions related to individual dependencies include:
 - a. Although many actions occur over a long period of time, it is assumed that there are no intervening successes between actions
 - b. The same crew is assumed to perform the actions throughout the event
 - c. Cognitive demand is assumed different except where the actions are similar, e.g., alignment of cooling
 - d. Simultaneous actions are assumed when operators are working with multiple, immediate concerns (e.g., pump power and isolation)
 - e. The required resources to perform a task (operators, maintenance personnel, health physics, etc.) are estimated based on experience and engineering judgment
11. Timing overrides are implemented as necessary to place actions into chronological order. In the current model, no overrides are necessary.

HRA Discussion

There are several types of human actions during normal plant operation:

- Those related with the failure to restore equipment to its normal condition following a test or maintenance action
- Those causing an IE

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Human actions during an accident are:

- Those performed by operators following established procedures for mitigating the consequences of an accident
- Those performed in the mistaken belief that they are the appropriate actions as indicated in the procedures, but in reality, can worsen the conditions of the accident, thereby complicating the mitigation process
- Those that are not explicitly included in the procedures and are used to recover the operability of failed equipment or use alternative means

Treatment of human actions in the PSA leads key to realistic understanding of event sequences and their relative importance to overall risk. The HRA is performed in a structured framework that allows development of the analysis in an orderly fashion while considering the general boundary of the PSA.

There are three main types of human actions in the HRA:

Type A: Pre-Initiating Event Human Actions

Before an IE, plant personnel affect availability of standby systems by inadvertently disabling equipment or failing to restore the correct position during the performance of operational activities in the plant. Example: maintenance staff incorrectly perform a valve lineup after maintenance on a pump. When the pump is needed during an event, it cannot run or is damaged.

Type B: Initiating Event Induced by Human Actions

Plant personnel initiate an event through interactions with plant equipment. These typically occur due to a mispositioning or incorrect operation of equipment that inadvertently trips equipment or inserts false control signals. Example: One of three pumps providing service water to a condenser fails and operators do not align a backup in a timely manner. The plant trips on loss of service water.

Type C: Post-Initiating Event Human Actions

After an initiating event has occurred, plant operations attempt to place the plant in a safe, stable state using emergency operating procedures. There are two types of errors associated with Type C actions:

1. Commission Errors – By following procedures during the course of an accident, plant personnel operate standby equipment that aggravates the accident. This action, called a commission error, gives rise to a situation in which the operators place the plant in a worse condition than if they had done nothing (i.e., an omission error). The risk of commission errors is considered insignificant when the plant has EOPs that have a symptom-based orientation and the EOPs cover all the possible scenarios in the event sequences analysis.
2. Omission Errors – Failure to perform a step in the procedure, or failure to act in a timely manner results in either equipment damage or core damage. It is imperative that operations perform the correct steps within the time frame before damage occurs.

Review of Plant Practices

Once the pre-accident errors are identified as part of the systems analysis, the human reliability analyst examines the pertinent procedures to determine which critical components are placed in off-normal configurations and if not restored to their normal alignment, could render the component, train, or system inoperable. Examples of errors that render components, trains, or system inoperable include:

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- Prior to performing maintenance on a pump, the protective tagging order typically involves isolating the pump by closing both the suction and discharge isolation valves, racking out the pump breaker, and placing the pump control switch in pull-to-lock. Failure to return these components to their normal position will result in unavailability of the pump.
- In the case of testing, failure to close the test line isolation valve can result in unacceptable degraded performance of the train or system.
- During instrument calibrations, a I&C technician errantly calibrates an instrument that automatically actuates, automatically trips, or causes some other actuation.
- While performing maintenance on a transmitter, the instrument isolation valves are left closed resulting in erroneous operation or failure to detect system parameter changes such as pressure or level changes.

Plant practices for operating BWRs are used until plant-specific maintenance procedures are developed.

15.6.1.3.10 Event Sequence Frequency Quantification

The BWRX-300 PSA model consists of event trees and fault trees that are quantified using a fault tree linking process. The calculation of core/fuel damage and release category frequencies is performed as single gate for each sequence and release category. The top gates include all sequences but uses a sequence marker to identify the event tree sequences in the cutsets generated by the single gate. The use of a sequence marker results in retaining all the minimal cutsets for that specific sequence but prevents subsuming of non-minimal cutsets among sequences. This is addressed later in the recovery process. The contribution to risk from these non-minimal sequences is generally small, although this can affect the results for specific hazards, such as earthquakes (seismic).

The sequence logic is set up to exclude any combinations associated with the success branches in the specific sequence. The individual sequence results are then combined for reporting, analyzing, or used as input for the L2 or L3 portions of the PSA.

Methodology

The purpose of the event sequence frequency quantification is to obtain the Boolean equation corresponding to the radionuclide release. The quantification is developed in terms of minimal cutsets representing the minimal combinations of events that result in radionuclide release.

The following key aspects characterize the release category frequency quantification process:

- Event trees model plant response to each group of IE
- Fault trees model the behaviour of front-line and support systems
- Integration of event tree and fault tree structures into a single linked model
- Quantification of the linked Boolean model with the probabilistic database and boundary condition files (flag files)

Use of Computer Aided Fault Tree Analysis for Solving the PSA

The EPRI Integrated Risk Technologies User Group offers the suite of risk software tools (formerly known as R&R Workstation) that are used for the Level 1 PSA model development and quantification. These risk software tools include CAFTA for event tree and fault tree construction and PRAQUANT for master model integration and quantification. Accident sequence frequencies

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are derived from the quantification results of a single top master fault tree by post-processing using the sequence markers.

The FTREX code is used in generating Minimal Cutsets (MCS) from fault tree logic models. FTREX uses a zero-suppressed binary decision diagram method for interpreting large fault tree models when developing MCSs. FTREX is built for PSA applications and is applicable to the BWRX-300 PSA without specific limitations.

Chapter 3, Appendix 3G discusses the computer programs used in the PSA.

The reliability databases supporting the BWRX-300 PSA include the IE frequencies, component failure frequencies and probabilities, CCF data, component repair times, test intervals and durations, mission times, maintenance unavailability, unavailability due to testing, and human error data. Uncertainties for data values are also included. All the data sources are justified and documented.

Additional Quantification Steps

The quantification process used for the PSA is iterative and includes numerous stages where internal and external review is performed before final results are documented. Key steps performed during the quantification include:

1. Circular Logic: It is possible to generate a master fault tree that has circular logic. This is where a gate is used as input to a second gate that happens to also be an input somewhere in the tree of the first gate.
2. Basic Event Consistency Check: The IE frequencies and the probabilities associated with BEs of the model are consistent with the definitions of the events in the context of the logic model. As the PSA cutsets are reviewed, they are reviewed for consistency in terms of the applied assumptions, specific design, and operational experience.
3. Mutually Exclusive Event Review: The model is solved to verify the event combinations are removed.
4. Human Error Dependencies: One example of how human failure error dependencies are determined is summarized:
 - a. All post-initiator HFE probabilities are set to one (or a high value).
 - b. The single top quantification is performed again. The most critical human failure error dependencies align, and this is apparent in the cutset results.
 - c. The PSA then uses the results generated in the cutsets to generate recovery rules that change the probability of these joint failures to a value that better reflects the dependency between these human failure errors taken from the HRA.
5. Recovery Actions: For a design level PSA, equipment recovery is not credited other than operation of backup systems that is addressed in the Type C operator actions. In the site-specific PSA, recovery is possible for longer term accident sequences where equipment recovery is possible.
6. Truncation Justification: To justify an appropriate truncation value is used, a test for CDF convergence is performed. Additionally, the last two convergence runs are used to develop a list of risk-significant accident sequences (e.g., contributing greater than 1%) to ensure no new significant accident sequences are identified by lowering the truncation limit further.

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7. Importance Analysis: This analysis documents the overall contribution of basic events to each phase of the PSA including the Level 1, Level 2, and Level 3 PSA. Separate importance results are provided for basic event types such as IE, human errors, event class, release categories, etc. Together with the uncertainty and sensitivity analyses, this satisfies CNSC REGDOC-2.4.2, Requirement 4.10 (Reference 15.6-1).
8. Uncertainty and Sensitivity
9. Review of the results: The U.S. PSA Standard and the IAEA-TECDOC-1804 (Reference 15.6-13) require the review of the results in detail to ensure the results are consistent and justified. For the BWRX-300, the comparison of these results with similar plants is not possible. As a result, a general comparison with the ESBWR or other BWRs using similar components (e.g., ICS) is made.

15.6.1.4 Internal Fire Hazards

The probabilistic fire analysis was performed with simplifying assumptions because the specifics of cable routings, ignition sources, or target locations in each zone of the plant are still in the design phase. Because of this limitation, a simplified, conservative, and bounding approach is used in this analysis.

The current scope of the analysis is for at-power accident scenarios.

Methodology

The BWRX-300 internal Fire PRA is performed according to the guidance in USNRC NUREG/CR-6850 (Reference 15.6-20). Other key references may be used:

- USNRC NUREG/CR-6850, EPRI 1019259, Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements" (Reference 15.6-27)
- USNRC NUREG-2169, EPRI 3002002936, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database" (Reference 15.6-28)
- USNRC NUREG/CR-7150 Vol 1, BNL-NUREG-98204-2012, EPRI 1026424, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE) Volume 1: Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," (Reference 15.6-29)
- USNRC NUREG/CR-7150 Vol 2, BNL-NUREG-98204-2012, EPRI 3002001989, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE) Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Fire," (Reference 15.6-30)

USNRC NUREG/CR-6850, EPRI 1011989 (Reference 15.6-20) and USNRC NUREG/CR-6850 Supplement 1, EPRI 1019259 (Reference 15.6-27), document state-of-the-art methods, tools, and data for the conduct of a Fire PRA for a commercial nuclear power plant application. The methods have been developed under the fire risk re-quantification study. This study was conducted as a joint activity between EPRI and the U.S. NRC Office of Nuclear Regulatory Research under the terms of an NRC/EPRI memorandum of understanding and an accompanying fire research addendum.

For the BWRX-300 Fire PRA model development, the following USNRC NUREG/CR-6850 (Reference 15.6-20) tasks are applicable:

- Task 1: Plant Boundary & Partitioning

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- Task 2: Fire PRA Component Selection
- Task 3: Fire PRA Cable Selection
- Task 4: Qualitative Screening
- Task 5: Fire-Induced Risk Model
- Task 6: Fire Ignition Frequencies
- Task 7: Quantitative Screening
- Task 12: Post-Fire HRA
- Task 14: Fire Risk Quantification
- Task 15: Uncertainty and Sensitivity Analyses
- Task 16: Fire PSA Documentation

Some of the above tasks are simplified while others are omitted at this stage of the plant design. Tasks not addressed in this study include:

- Scoping Fire Modeling (Task 8)
- Detailed Circuit Failure Analysis (Task 9)
- Circuit Failure Mode Likelihood (Task 10)
- Detailed Fire Modeling (Task 11) including multi-compartment analysis
- Seismic Fire Interaction (Task 13)
- Support Task A Plant Walk Downs is not performed in the design phase

The Fire PRA database is created with an Access®-driven database (FRANX) that includes all tables that are necessary to develop a scoping-level Fire PRA model. Enhancements to the Fire PRA database are possible in future updates once more details are available for cable tray routes and their contents as well as more detailed ignition source locations.

Fire ignition frequencies for power operation at each plant area (physical analysis unit) are estimated using the USNRC NUREG/CR-6850 (Reference 15.6-20) and USNRC NUREG/CR-6850 Supplement 1 methodology (Reference 15.6-27) and data from USNRC NUREG-2169 (Reference 15.6-28). Fire frequencies for shutdown conditions are not developed at this time.

For a postulated fire, a list of impacted components is generated with the mapping defined in the Fire PRA database. A list of impacted components is also generated with the assumed cable routing. The cable routing is assumed based on the modeled PRA components, their supports, and the general room layout of the BWRX-300 design. Fires are conservatively assumed to propagate unmitigated in each fire area (no suppression is credited) and damage all functions in the fire area. The internal events PRA accident sequence structures and system fault trees and success criteria are used in the calculation of the fire CDF.

The BWRX-300 Fire PSA employs the following EPRI software:

- Fault Tree Reliability Evaluation eXpert. This software generates cutsets from the fault trees produced in CAFTA.
- CAFTA. This software is used to build the logic model of the plant, producing all the fault trees and event trees.

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Assumptions

1. A fire ignition in any fire area may grow into a fully developed fire.
2. The analysis does not take credit for any fire suppression (i.e., self-extinguishment, installed suppression systems, nor manual firefighting activities). All fires disable all potentially affected equipment in the area in the analysis.
3. Unless otherwise stated, a fire causes failures of all fire-susceptible components in the subject fire area and detailed fire modeling is not performed in this revision of the BWRX-300 Fire PRA.
4. Recovery of the failed components or cables after the postulated fire is not credited.
5. Unless otherwise stated, the analysis does not credit the distance between fire sources and targets.
6. All fire-induced equipment damage occurs at time zero in the scenario progression.
7. Cable routing is postulated for PRA purposes using plant general arrangement drawings with major electrical component locations because cable routing is provided later in the design. A list of cables is generated that includes all modeled supports for PRA components included in the current PRA model. This list captures most cables, especially for expected risk-significant components.
8. Fires in the MCR and SCR are currently modeled as impacting components associated with any assumed cable routings that go through the rooms but do not terminate in the MCR or SCR (i.e., the assumed cable route for a bus that goes through the MCR between the TB and the B train MCC room in the CB). The MCR and SCR are not assumed to impact components where control or visualization cables may be in the rooms. This is a realistic treatment because plant design could be similar to the ESBWR where the control room controls are connected to the Distributed Control and Information System (DCIS) rooms (which would be unaffected by a MCR fire), via fiber cables. The loss, including melting, of the fibers or visual display units (VDUs) will not cause inadvertent actuations, nor affect the automatic actuations associated with safety and non-safety equipment. In addition, fires in the MCR and SCR are assumed to fail all modeled human failure events.
9. All finalized details of cable route information are not yet available. Individual routes are assumed based on location of PRA modeled power/signal sources and the component.
10. The cable routings use general separation criteria because of the number of different signal and power divisions where there are physical analysis units with more than one division and power and control cables postulated. As the BWRX-300 design matures, cable routings are refined to include any credited fire barrier(s) within an overall physical analysis unit that serve to separate divisions.
11. Fire ignition frequencies remain constant over time and are based on industry generic fire frequencies. At operating plants, the total ignition frequency is the same among plants for the same equipment type, regardless of differences in the quantity and characteristics of the equipment type that exists among the plants. This is conservative because the BWRX-300 design has significantly fewer pumps, motors, and other active components than earlier plant designs upon which the current plant generic ignition frequencies are based.

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12. All ignition source type bins are applicable to the BWRX-300 plant with the following exceptions:
 - a. Bins 02 and 03 are not applicable since they are used for pressurized water reactor plant designs.
 - b. Bin 22 for scram motor generator sets is not applicable to BWRX-300.
 - c. Bin 04 is not applicable to the BWRX-300 MCR or SCR because the BWRX-300 design is completely digital as opposed to traditional electro-mechanical designs. However, to ensure a conservative analysis is produced, twenty-five percent (25%) of the traditional Bin 04 ignition frequency is assumed to be applicable and is assigned to the BWRX-300 MCR ignition frequency.
13. Although the BWRX-300 plant may be located with existing NPPs, the plant location weighting factors are one because the BWRX-300 plant is designed as a single-unit plant with no shared buildings.
14. Because the BWRX-300 plant is in the preliminary design phase, the count of components is performed with the modeled PRA components as well as preliminary design layout drawings.
15. Because the plant has not operated and no history of maintenance activity is available, the weighting factor evaluation is simplified. It is assumed that all compartments (physical analysis units) have the same transient fire influencing factors with a value of one is used for all influencing factors.
16. All fires result in a manual reactor shutdown regardless of location or potential induced failures.
17. Full details of electrical cabinet locations are not yet available. Preliminary design information notes the division or train of equipment is used in the development of this revision of the BWRX-300 Fire PRA for electrical cabinet and high energy arching fault counts.
18. The containment is always inerted during power operation. The possibility exists that the containment could be de-inerted in preparation for an outage, or still de-inerted when coming out of an outage. Typically, de-inerting occurs around 2-3 days before an outage that can result in a fire during this time. This time window is not included in the current BWRX-300 Fire PRA but is included in the future shutdown operation BWRX-300 Fire PRA.
19. Cables associated with display or visualization component status or function may be routed between a component or instrumentation location and the MCR or SCR are not impacted by any automatic function of the component.

Modeling for Full-Power Condition

Detailed fire modeling is not performed for this revision of the BWRX-300 Fire PRA for at-power operation. Detailed fire modeling is undertaken when the BWRX-300 design is more mature and additional details on equipment locations and cable tray routes are known with certainty.

For this revision of the BWRX-300 Fire PRA, full room (physical analysis unit) burnout scenarios are used where any PRA-related component and cable within the particular physical analysis unit is failed at the total physical analysis unit ignition frequency during Fire PRA quantification.

Evaluation of Multi-Component Scenarios

Fire propagation cases (multi-compartment scenarios) from one physical analysis unit into another physical analysis unit are not currently postulated for the fire areas listed in the at-power Fire PRA. As the BWRX-300 design matures, this analysis is undertaken.

Evaluation of Potential Smoke Damage

Detailed HVAC system design information is not yet available for the BWRX-300 plant design. Consequential failures from potential smoke damage cannot be assessed. This is provided in future updates to the BWRX-300 Fire PRA as the plant design matures.

Detailed Fire Modeling for Shutdown Condition

The shutdown PRA model is not yet constructed but is expected to be based on the Level 1 internal events PRA model. Detailed fire modeling follows the same process described previously with the exception that the evaluation of the applicability of shutdown conditions in the Fire PRA model is provided in future updates to the BWRX-300 Fire PRA.

15.6.1.5 Internal Flooding Hazard

The BWRX-300 internal probabilistic flood analysis objective is identifying and providing a quantitative assessment of the radionuclide release risk due to internal flood events. It models potential flood vulnerabilities in conjunction with random failures modeled as part of the internal events PSA. Through this process, flood vulnerabilities that could jeopardize core integrity and containment integrity are identified.

The floods may be caused by large leaks due to rupture or cracking of pipes, piping components, or water/fluid containers such as storage tanks. Another possible flooding cause is the operation of fire protection equipment.

Excluded from this analysis is flooding associated from external sources that is considered under the External Flooding Hazard Analysis for localized flooding events and intense weather events. These external flooding events are reasonably precluded from the BWRX-300 probabilistic flood analysis because the standard plant design prevents external flooding conditions affecting the safe operation of the plant.

Methodology

The internal PSA flooding analysis identifies and classifies potential flooding sources and events. Component location and data is compiled generating a frequency of occurrence that represents the effects associated with each of the potential flooding events. In addition, an evaluation is performed to identify, screen, and quantify specific plant effects/failures associated with each flooding event. Finally, the BWRX-300-specific flooding frequencies and plant effects are applied to the PSA model to obtain risk results. For the BWRX-300 Flooding PSA model development, the following tasks are performed:

- Identification of flood sources
- Development of flooding scenarios
- Development of flooding frequencies
- Analysis of flooding scenarios

The internal PSA flooding analysis is based on the design basis BWRX-300 SSCs. Critical to the flooding analysis is the location of these features and their interaction with other BWRX-300 SSCs. The current list of system components and location of equipment is based on the current

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design and plant layout drawings and relies on the component location analysis collected for the Fire PSA step (e.g., common spatial information).

The development of BWRX-300-specific flooding scenarios requires a detailed analysis of data including plant component location, system capacity, and potential failure mechanisms. Following the identification of potential flooding scenarios, characteristic scenarios are selected as representative of flood areas and subject to quantitative analysis.

Data is collected from industry sources for the BWRX-300 equipment and system components. Failure data for consideration in the flooding analysis includes piping runs, pumps, valves, tanks, heat exchangers, and circulating water expansion joints to develop the severity and effect of potential flooding scenarios. These failure rates in combination with types and capacity of system components located within specific flood zones are used to develop the flooding frequencies and frequency uncertainties. Flooding frequencies for both large break and small leak scenarios are developed for each flooding scenario.

The flood scenarios for each flood area are quantified to calculate a probabilistic risk value and summed to provide an overall risk analysis. USNRC NUREG/CR-4639 and NSAC-60 provide additional guidance for flooding analysis. The requirements for a flooding PSA are discussed in Section 3 of the PSA Standard ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12).

Walkdowns are a critical task for a mature flooding PSA and is performed and documented after the plant construction is complete.

Key Inputs and Assumptions

The following are inputs for the BWRX-300 flooding analysis:

- At-power internal events PSA model
- BWRX-300 design features for protection against flooding
- Flood zone spatial information

The following assumptions are used in support of the BWRX-300 flooding analysis:

1. Flooding resulting from component ruptures
2. For each tank rupture, the entire tank inventory is drained
3. Non-qualified submerged equipment (motors or solenoids for valves, control cabinets and circuitry) fails if the water level in the flood zone reaches a level of 1 foot above floor elevation
4. If equipment fails, the equipment fails at the start of the flood
5. The expected effect of flooding electrical equipment such as motor control centers, electrical cabinets, and terminal boxes, is a short to ground, removing power from the loads served by the component (all electrical equipment failures are treated as ground shorts)
6. Flooding and/or spraying of a motor-operated valves (MOVs) causes the valve to fail as-is
7. Passive components, such as pipes and tanks are not considered vulnerable to flooding effects
8. Concrete walls are considered flood barriers, capable of withstanding the expected maximum flood loading, and remain intact throughout a flooding event

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9. Electrical circuit fault protection provides protection for plant electric circuits via protective relaying, circuit breakers, and fuses and does not result in the loss of the bus that supplies power to the affected component
10. For floor drains, appropriate precautions such as check valves, back flow preventers, and siphon breaks prevent back flow and any potential flooding into connected plant areas (propagation of flooding through the drain system via failed check valves is not considered)
11. Doors in electrical equipment rooms within the RB are evaluated for watertight doors for flooding up to the ground level elevation. For the flooding analysis, the watertight doors are normally closed at-power and opening of these doors generates an alarm in the control room with procedures that direct their immediate closure upon alarm receipt
12. Dry pipe systems (such as a pre-action Fire Protection System) with closed sprinkler headers are not modeled as flood sources due to the low frequency failure of the dry pipe coincident with spurious opening of the actuation valve
13. Equipment located in the yard is not considered susceptible to internal flooding damage
14. Human-induced mechanisms such as overfilling tanks or diversion of flow created by maintenance are not modeled because operating and maintenance procedures have not been developed, and frequency of maintenance and duration of maintenance have not been determined. They are included in the PSA updates when these inputs are available
15. The flood volume retained by the drain system is not credited in the flooding analysis for large flooding scenarios. Therefore, the capacity of the drain system has not been estimated
16. The amount of water retained by sumps, berms, dikes, and curbs is negligible. This is conservative because retained water provides additional time for flooding response
17. Flooding initiator frequencies remain constant over time
18. Among the plant buildings, flooding frequency is the same for the same equipment type, regardless of difference in the quantity and characteristics of the equipment type that may exist among the plant

15.6.1.6 High Wind Hazard

The BWRX-300 high wind analysis quantifies event sequences and containment releases initiated by both tornado and straight winds. Straight winds are lesser velocity winds that pose minimal challenges to the plant design. Straight-line winds of lesser velocities are not included in the BWRX-300 high wind risk analysis. The consequences of these winds is addressed as part of the weather-related LOPP IE in the at-power internal events PSA model.

Methodology

The high wind risk analysis involves the following major steps:

- Tornado high wind event frequency
- Tornado-induced plant effects
- Analysis of tornado-induced release category frequencies
- Analysis of straight-line wind effects

The high wind PSA analysis identifies and classifies potential high wind events. Data is then compiled to generate an occurrence frequency that represents each potential high wind events.

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In addition, a qualitative evaluation of the design basis BWRX-300 SSCs is performed identifying specific plant effects/failures associated with each high wind event. A BWRX-300-specific high wind event frequency and plant effects are applied to the at-power model to obtain risk results. High wind hazards are characterized by their impacts (e.g., dynamic load from gusts, averaged loading, rotation velocity, pressure differential, tornado path, missile impact potential).

The BWRX-300 is designed for a tornado wind load that is the maximum wind speed such that it does not challenge safety structures.

ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12), including the guidance in the non-mandatory appendix, provides requirements and guidance on performing detailed high winds PSA. The changes in Part 7 of ASME/ANS PSA Standard are the most significant changes and reflect advancements in high winds PSA. Included in the standard requirements is the consideration of correlated hazards and hazard effects, such as the potential for local flooding associated with high winds or the impact of high wind-driven rain. This standard is utilized for both the screening of potential hazards and the analysis of any unscreened hazards.

The BWRX-300 High Wind PSA uses the following EPRI software:

1. Fault Tree Reliability Evaluation eXpert - this software generates cutsets from the fault trees produced in CAFTA.
2. CAFTA- this software is used to build the logic model of the plant, producing all the fault trees and event trees.
3. ACUBE 2.0 -this software provides a more accurate solution to the cutsets than conventional solutions.

Key Inputs

Site-specific data are inputs for the external hazards PSA analyses. Site-specific wind hazard analysis generally requires the use of regional data. The size of the region requires judgment and depends on the regional climatology and type of wind hazard, the number of years accurate records are available, the extent and quality of the data, and the hazard's spatial variability within the region. In the design phase, it is possible to select a bounding region for a high wind hazard; however, a bounding region for tornadoes may not be the bounding region for hurricanes.

The high winds equipment list is evaluated based on the internal events PSA required functions and supporting SSCs. The internal events PSA is used to quantify the high winds PSA with modification of the damaged or potentially damaged SSCs.

Key Assumptions

The following assumptions were used in conducting the BWRX-300 high wind risk analysis:

1. The winds classification used in the BWRX-300 high wind analysis is based on the Saffir-Simpson scale.
2. The tornado winds classification for the BWRX-300 high wind analysis is based on the recommended wind speed ranges of the Enhanced Fujita scale (EF-scale).
3. All at-power BWRX-300 high wind analyses include high winds assuming the plant is operating. This approach is conservative for the high wind analysis where sufficient advanced warning and procedures allow the plant to be placed into a safe condition (shutdown operations) prior to the high wind event.

4. For the BWRX-300 high wind analysis, the frequency of occurrence data for tornadoes high wind events is compiled from sources based on the Enhanced Fujita scale (EF-scale) for classification of tornadoes. This data is correlated to the BWRX-300 EF-scale of tornado classification.

15.6.1.7 Seismic Hazard

The scoping Seismic PSA (SPSA) for the BWRX-300 Standard Plant Design describes the methodology used:

- Presents the seismic hazard curve used
- Lists the generic structure and component fragilities and sources
- Presents results by ground motion level and important fragilities
- Documents seismic risk insights
- Identifies plant design requirements

The model used is a preliminary scoping model that provides important insights into plant design. The SPSA activities include:

- Identifying IEs applicable to seismic events
- Developing a seismic equipment list
- Assigning representative fragilities applicable to SSCs
- Adjusting for human error probabilities to reflect increased distractions and stress from the seismic events
- Modeling seismic Level 2 PSA in the updated SPSA

The scoping SPSA is a full-power model. Seismic-induced flooding and fire, contact chatter, building-to-building interactions, and soil liquefaction are addressed qualitatively in the scoping stage.

Although for low ground motion levels offsite power may be available, offsite power is assumed unavailable in all modeled accident sequences.

The plant operational and emergency systems are represented by event and fault trees. The analysis gives insights into the dominant contributors to seismic risk.

Methodology

The BWRX-300 scoping SPSA development involves four major activities:

1. Probabilistic seismic hazard analysis
2. Seismic fragility analysis
3. Seismic plant response analysis
4. SPSA quantification

The probabilistic seismic hazard analysis identifies ground motion bin IE frequencies that are based on the seismic hazard curve for the Darlington site. This curve is then partitioned into several ground motion bins.

The BWRX-300 SPSA results are reported based on truncating the seismic hazard at the 1E-6/yr earthquake frequency, consistent with IAEA-TECDOC-1791, "Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants" (Reference 15.6-21).

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For some external hazards, the IAEA Safety Requirements states that “it may not be practical or even possible to demonstrate that the occurrence of a hazard of such severity could cause extensive plant damage leading to a large or early radioactive release and therefore needing to be practically eliminated, is below a threshold of frequency such as 10⁻⁶/year”. Seismic events with exceedance frequencies less than 1E-6/yr are attended by significant uncertainty. The IAEA guideline is applied to the SPSA results by truncating the seismic hazard. Applying the seismic hazard truncation at the 1E-6/yr earthquake frequency results in the sum of the CDFs for seven ground motion bins.

The seismic fragility analysis produces a seismic equipment list that includes a list of plant SSCs related to seismic-induced IEs and SSCs that mitigate plant responses to IEs. Fragilities for the SSCs on the seismic equipment list are selected in the seismic fragility analysis.

The SPSA model is developed from the seismic plant response analysis. The seismic plant response analysis uses a seismic IE tree to model the plant structures that are required to remain intact and the probability of a seismically-induced LOCA, main feedwater line break and main steam line break. A LOPP is assumed to occur for all SPSA modeled sequences. The end states of this event tree that do not go directly to core damage are linked to the appropriate internal event trees.

An updated internal events top logic model is used as input to the SPSA top logic model development update. This updated internal events top logic model is used to update the systems fault trees and PSA event trees reflecting the current evolution of the BWRX-300 design. The internal events top logic addresses:

- Updated system fault trees to reflect current system designs
- Updated Level 1 event trees
- Updated model continues to credit containment vent for decay heat removal
- Updated model continues to credit CRD injection, to control inventory during failure to scram and LOCAs
- Updated model continues to credit ultimate pressure relief for transients, small LOCA and failure to scram

For the SPSA quantification, the FRANX code is used to integrate the seismic hazard with the SSC fragilities to produce the SPSA CDF results.

The BWRX-300 SPSA uses the following EPRI software:

- FRANX uses the internal events model and adds seismic initiators and hazard curves that results in the CDF for each ground motion level
- Fault Tree Reliability Evaluation eXpert generates cutsets from the fault trees produced in CAFTA
- CAFTA is used to build the logic model of the plant, producing all the fault trees and event trees
- ACUBE 2.0 provides a more accurate solution to the cutsets than conventional solutions

Assumptions

The following principal assumptions are used in developing the seismic PSAs:

1. A LOPP is modeled to occur for all seismic accident sequences. Although offsite power may be available for low ground motion level earthquakes, this assumption is made to simplify the model. A consequence of this assumption is that power conversion system is unavailable. This assumption is made in this scoping model, but in the final model, the offsite power system is modeled explicitly.
2. The seismic-induced failures of the RB, reactor support structure or RPV are assumed to result in core damage and containment bypass.
3. Fragility group MV-MSIV consists of the MSIVs and RIVs. It is conservatively assumed in this assessment that the two sets of valves are similar in design response to be seismically correlated, meaning that if one fails, the other will fail as well. This assumption is revisited in the final model using the methodology discussed in USNRC NUREG/CR-7237, "Correlation of Seismic Performance in Similar SSCs (Structures, Systems, and Components)" (Reference 15.6-22), these valves have uncorrelated fragilities due to dissimilarities. The outboard MSIV and MSIVs are treated as fully correlated and with further analysis may be treated as partially correlated.
4. Chatter of electrical contacts, such as relay contacts, is not addressed. Contact chatter is not expected to contribute significantly to BWRX-300 seismic risk. The DL4a ICS actuation and the DL4 scram actuation are expected to fail-safe as a result of chatter and that do not include seal in circuits that would interfere with actuation. Contact chatter assessment is addressed in a future revision of the SPSA.
5. The SPSA utilizes fragilities primarily from the ESBWR. Where ESBWR data are not available, representative fragilities from EPRI Report 3002000709, "Seismic Probabilistic Risk Assessment Implementation Guide" (Reference 15.6-23) are used. For risk-significant fragility groups, representative fragilities developed for the BWRX-300 are used.
6. The electrical equipment that supports DL2 and DL3 actuation signals is seismically correlated and is modeled by a fragility group. This modeling is conservative because the equipment is not collocated. DL2 electrical equipment is located in the control building and the DL3 electrical equipment is located in the RB. Although DL3 is designed to fail in the fail-safe condition, DL3 ultimately relies upon digital electrical equipment to function and DL3 actuation is assigned to a fragility group.
7. In this scoping model, the fractions of the small LOCA frequency that occur in the CUW and the CRD systems are the same for seismic events as for random failures. This simplification is compared to plant-specific data in the final model.
8. Dependencies of multiple human failure events in the same cutset are not modeled. Because there are few human failure events in the model, human actions are not as important in the BWRX-300 as they are in most operating plants, and the nominal HEPs are screening values where dependency is not likely to be important.
9. The diesel generators located in the TB are air-cooled. The seismic failure of the TB is modeled to fail both diesel generators. The seismic failure of the intake structure does not impact the diesel generators.

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10. At this stage of the analysis, equipment interaction concerns (e.g., equipment falling on other equipment) are not evaluated. As equipment location is finalized, equipment interaction is addressed.
11. All redundant components in the same system are correlated. This is reasonable for components that are identical, of the same orientation, and on the same floor. The MSIVs and the MSRIVs correlation may be partially correlated as the design evolves. As more detailed designs are developed, other instances of partial correlation may be found and modeled using the USNRC NUREG/CR-7237 methodology (Reference 15.6-22) and re-evaluated in the final model.
12. Generic fragilities are used in the preliminary SPSA. This is sufficient for the scoping study because the fragilities are not site-specific.
13. The DNGS seismic hazard curve is used as input to the BWRX-300 SPSA. In the final model, a site-specific hazard curve is used.
14. A break in the CUW piping could result in a break outside containment if the RWB that is not a safety structure (it is a hardened structure that meets OBE or 1/2 SSE earthquake) is damaged or the CUW heat exchangers are damaged. This scenario is evaluated when the design is finalized.
15. The polar crane is designed, or its position administratively controlled by plant procedure during normal operations so that seismic damage of the crane does not impact safety class RB contents, such as the ICS pools and heat exchangers.
16. Equipment anchorages comport with the assigned fragilities, and this is verified as the design progresses.
17. The existing DNGS site has no impact on the SPSA for the BWRX-300 and is a single site.
18. In this current scoping model, equipment screened out of the internal events PSA due to low failure likelihoods, such as ductwork, dampers, piping, and cable trays, are not added and are included in the fragilities that are modeled. In the final model, as more plant-specific fragilities are developed, these components are added.
19. If a building fails, all equipment in that building also fails.
20. If the CB fails, all human failure errors except the human failure errors to align and operate diverse and flexible mitigation strategies (FLEX) fails. The final model addresses whether the CB failure also fails FLEX due to unavailability of operators or lack of a path to the FLEX equipment. Since the HEP for aligning FLEX is 1.0 at the most risk-significant ground motion levels, allowing FLEX with CB failure is of little consequence.
21. A failure of the TB fails both steam lines and feedwater lines because these lines extend into the TB.
22. The FLEX system allows water from Lake Ontario pumped to the reactor, fuel pool, or ICS.
23. Unisolable LOCA is not modeled by the SPSA at this time.
24. Fragility group CR-RODS, Failure of Control Rods to Insert is completely correlated. If control rod scram function is damaged, all control rods are assumed to fail to fully insert.
25. For RPV rupture (fragility group CONRPV), damage occurs in the lower portion of the RPV.

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As part of the scoping Seismic PSA, a sensitivity analysis is performed to identify the safety-significant seismic fragility groups. This sensitivity analysis identifies the BWRX-300 SSCs that contribute significantly to the seismic safety profile.

Nine fragility groups are identified by the sensitivity analysis to have seismic safety significance. Of these nine, one group (the reactor building polar crane) is expected to be mitigated by administrative controls rather than seismic design/analysis refinement. The other eight fragility groups are:

- Control Rods
- RIVs/MSIV
- IC HX
- ICS Actuation Valves
- RB
- RPV
- RPV Support Structure
- Scram HCUs

As the detailed design progresses, updated seismic capacities are developed. These seismic capacities are then fed back into the seismic PSA and new risk metrics are recalculated in an iterative process. This risk-informed design process ensures that the DNNP BWRX-300 safety goals are met.

15.6.1.8 Level 1 PSA – Low Power and Shutdown Risks

The Low Power Shutdown (LPSD) Risk PSA is developed following the requirements of the ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12). Currently, the ASME/ANS standard for LPSD is issued for trial use and pilot application. After the trial use, feedback was provided to the PSA Standard committee and the LPSD standard is being revised.

A detailed PSA is performed to determine the radionuclide release risk during shutdown. Loss of CUW, SDC, Plant Cooling Water System (PCW) and LOPP are all evaluated for radionuclide risk during shutdown. The approach for the LPSD PSA is similar to the Full-Power PSA, involving fault trees and event trees used in determining the shutdown risk for each IE analyzed.

The evaluation encompasses plant operation in hot shutdown, cold shutdown, and refueling modes (Modes 3, 4, 5, 6) while Modes 1, 2, 3, 4 are bounded by the at-power PSA model. During these modes, the plant is transitioned through several POSs that are distinguished in the LPSD PSA by different plant conditions and configurations. The LPSD PSA addresses POSs where there is fuel in the reactor vessel. It includes all aspects of the NSSS, the containment, and all systems that support operation of the NSSS and containment.

The event sequences in the Shutdown PSA are classified according to whether the core is damaged. All shutdown core damage sequences lead directly to a radionuclides release to the environment (containment is assumed to be open at the time of the IE). In the final PSA, this assumption is revisited to accurately assess the PSA results.

The critical safety functions credited in the shutdown model are decay heat removal and inventory control. The containment function is credited for POSs where containment is not open. Because the containment is open for many of the POSs, containment integrity is not relevant to any modeled functions. Reactivity control is assumed to have no significant effect on the shutdown

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model. Electrical power availability, as well as the availability of other support systems, are modeled for the effect on decay heat removal. LOPP is evaluated as an IE, and power dependencies for systems are included in the model in a similar manner to the at-power PSA.

The LPSD PSA includes fuel and heavy load movements where the PSA analyzes heavy load drops that can cause fuel damage, core damage, or large releases. The PSA results for heavy load drops considers the following events:

- Fuel pool leak due to rupture At-Power
- RPV leak due to rupture (below top of fuel) during Low Power and Shutdown
- RPV leak due to rupture (at feedwater nozzle) during Low Power and Shutdown
- Fuel Pool leak due to rupture during Low Power and Shutdown
- Loss of ICS Train A or SDC during Low Power and Shutdown
- Loss of ICS Train B or SDC during Low Power and Shutdown

Methodology

ANSI/ASME/ANS RA-S-1.1-2022 (Reference 15.6-12) provides input guidance documents for the Low Power and Shutdown PSA model development, including the LPSD PSA Standard, and IAEA-TECDOC-1144, "Probabilistic Safety Assessments of Nuclear Power Plants for Low Power and Shutdown Modes" (Reference 15.6-24).

Plant Configurations in Low Power and Shutdown Conditions

Differences between the Low Power and Shutdown PSA and the Power Operation PSA are attributed to:

- Plant operating mode
- Plant operating state including configuration
- Time after shutdown
- Reactor vessel and containment status
- Vessel and core temperatures
- Fuel location
- Availability of required systems and support systems

This analysis addresses the BWRX-300 risk associated with a refueling outage. The systems modeled are evaluated based on anticipated activities associated with refueling operations.

To develop a suitable shutdown model, multiple bounding plant configurations/POSSs are defined with similar characteristics in relation to the residual heat and the availability of systems.

The outage plant operating mode and POS are used to define the initial plant condition for individual event sequence quantification.

Once the outage POSSs are defined, the duration of each is estimated to determine its contribution to the overall calculation of annual CDF. The duration is expressed in hours per refueling outage.

The Shutdown PSA considers outage plant configurations representative of the possible plant configurations during shutdown.

Initiating Events

The IEs that challenge normal operation including the critical safety functions (e.g., heat removal, inventory control) during shutdown operations are determined. A shutdown IE is defined as any event that challenges normal operation and requires action to prevent core damage.

Event Trees

The LPSD event tree construction considers the following aspects:

- Chronological order of system actuation
- Grouping of mitigating systems by safety functions

The event sequence analysis and end state nomenclature are the same as the at-power PSA. The associated success criteria analysis is performed similar to the at-power success criteria analysis. However, success criteria analysis for LPSD may be approximated by hand calculations.

System Analysis

The unavailability of a system to perform its safety function on demand is evaluated by fault tree analysis similar to the fault tree analysis approach.

The necessary fault trees are identified following construction of the event trees. These fault trees represent the nodes included in the event trees and any required support system fault trees

Maximum use is made of the fault trees developed for the Full-Power PSA. Potential differences between the at-power and the shutdown fault tree models may result from differences in:

- Maintenance unavailabilities
- Success criteria between at-power and shutdown condition
- Initial system configuration between at-power and shutdown condition
- Human actions

Maintenance events used in the at-power model are not adjusted for the LPSD PSA, other than for POSs where some systems are entirely unavailable (for the POS). Additional maintenance events may be possible at LPSD that do not normally occur at full-power. The evaluation of maintenance for each POS is justified and documented.

Quantification

The shutdown event sequence analysis models the effects on the following two critical safety functions during shutdown:

- Decay heat removal
- Containment during configuration changes in the containment boundary and reactor head during refueling

IE, associated frequencies, are identified that challenge the above critical safety functions. Event trees are developed specific to the shutdown configurations and the system fault tree analysis is based on at-power fault tree models adjusted to match shutdown conditions. The model development and quantification are performed using CAFTA and FTREX in a similar manner to the at-power PSA.

Start-up and Shutdown Transition Risks

In the BWRX-300 PSA shutdown models supporting the design, the start-up and shutdown risks are included in the at-power conditions.

Planned shutdown and start-up transition states are difficult to model with PSA for several reasons. The reactor is under continuous transition, e.g., main circulation flow. Temperature and pressure in the reactor vary significantly during the operation. Several manual operator actions and decisions are required. In addition, the operation may require the use of many systems and/or components that are not used during power operation. Identification of risks is also more challenging as it requires detailed understanding of the systems related to process and control. For these reasons, certain assumptions and simplifications are performed to create a usable and practical PSA model.

The new event trees include the same IEs as the previous PSA model, but it also considers transients prior to control rod insertion and failure possibilities related to maneuvering of the control rods. Modeling and quantification of transition modes is similar to the at-power PSA, if quantified.

Key Inputs and Assumptions

- At-Power BWRX-300 PSA models (note external events low power and shutdown models are developed in the external events sections)
- Representative BWRX-300 Refueling Outage Plan Key Assumptions
- Containment is open during the outage

15.6.1.9 Spent Fuel Damage

Spent fuel damage evaluation is required for the plant-specific BWRX-300 PSA because fuel damage frequency is an important contributor to release risk relative to other low risk contributors. For accidents, where the spent fuel is damaged outside the reactor core, the term spent fuel damage is applied. In particular, the fuel damage frequency may be an important contributor to release risk. The term “fuel damage” represents damage to the fuel outside the reactor vessel, while “core damage” is used for damage inside the vessel.

Methodology

A separate PSA analysis investigates the event sequences leading to spent fuel damage. The analysis estimates the related event sequences and their frequencies, and it is documented separately as part of the PSA. The analysis covers both power operation and outages. Sabotage is explicitly excluded from the calculation of the spent fuel damage frequency.

During the performance of the BWR PSA, the Fuel Pool PSA is performed using the process and requirement of the LPSD PSA including the requirements in the LPSD PSA Standard. In this PSA, the fuel pool is analyzed as a separate POS, and the associated IEs are those that result in an event that challenges the normal operation of the fuel pool cooling or inventory requiring mitigation to prevent fuel damage. The overall process of event tree analysis, success criteria analysis, fault tree analysis, is the same process used in the LPSD PSA. Generally, the time available for responding to a fuel pool IE is much longer than the LPSD IE response.

Event sequences are developed and quantified that credit potential recovery actions taken by the operator.

Key Inputs

- Site-specific external events

- Site-specific spent fuel handling equipment design
- Plant procedure for heavy load lifting and spent fuel handling
- Task outputs and preliminary results

This task develops the Fuel Damage PSA models. A number of IEs are expected to be screened with qualitative assessments.

15.6.1.10 Systems Credited in the PSA

The following design features are modeled in the PSA.

Isolation Condenser System

The ICS function modeled in the PSA is removing decay heat from the reactor by condensing steam in the ICS heat exchangers.

Reactor Isolation Function

The reactor isolation function mitigates the effects of large and medium sized pipe break LOCAs.

The term "RPV isolation system," is used to refer to RIV closure. The system model is a surrogate for valves that are part of the NBS.

Ultimate Pressure Regulation Function

The ultimate pressure regulation function of the RPV provides emergency pressure relief in the event of a severe pressure transient.

Control Rod Insertion Function

The CRD system performs several insertion functions and consists of these design features: FMCRDs, HCU, and the CRD Hydraulic Subsystem.

The PSA-credited function of the control rod insertion is to insert negative reactivity into the reactor core rapidly upon a scram signal. Subcriticality is achieved with the negative reactivity insertion and terminates fission heat generation.

The CRD system also provides an RPV inventory makeup function.

Feedwater Flow Reduction Function

The feedwater flow reduction provides negative reactivity in a failure to scram condition by reducing feedwater flow resulting in increased core voiding (the BWRX-300 has a negative reactivity void coefficient).

Containment Isolation Function

The containment isolation function isolates containment in the event of accidents or other conditions and prevents the unfiltered radioactive release before they exceed allowable limits.

Boron Injection System

The BIS provides a separate, diverse means, D-in-D backup system to the CRD system for manually inserting negative reactivity into the reactor core for BDBAs. All equipment is located outside primary containment to allow easy access for testing and inspection activities during all plant operating conditions.

The BIS utilizes an aqueous solution of highly enriched sodium pentaborate decahydrate for reactivity control. The sodium pentaborate solution temperature is maintained above the solubility temperature due to the placement of the system within the reactor building.

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CRD Injection Function

The focus of this system for PSA is the CRD Hydraulic Subsystem that is used for inventory makeup and flow to the SDC pumps.

The CRD Hydraulic Subsystem provides clean, demineralized water that is regulated and distributed to provide charging to the scram accumulators and purge water flow to the FMCRDs during normal operation. The CRD Hydraulic Subsystem is also the source of purging water to the SDC system pumps and the NBS reactor water level reference leg instrument lines. Additionally, the CRD Hydraulic Subsystem provides high pressure injection to the reactor. This makeup water is supplied to the reactor via the drives.

Shutdown Cooling System

The SDC system provides long-term decay heat removal after planned shutdowns. SDC consists of two independent trains designated as Train A and Train B. Each train suction is independently connected to an ICS condensate return line outside containment, downstream of the containment isolation valves.

Power Conversion Function

There is only one function for the power conversion function modeled in the PSA. This function is providing water from the condenser, via the condensate pumps, feedwater heaters and feedwater pumps, to the reactor where it is heated, and steam is produced. Steam is transferred to the condenser via the turbine bypass valves.

Because the PSA model assumes a scram has occurred or is warranted, the full-power function of providing steam to the turbine for power generation is not considered.

Cooling Water Systems

The function of PCW removes heat loads in the RB and TB.

PCW is a closed cooling water system supported by cooling from a portion of the Circulating Water System (CWS).

DC Power

The electrical distribution system is an integrated power supply and distribution system for the power plant. Three plant systems constitute the overall electrical system: Safety Class 1 Electrical Distribution System, Safety Class 2 and 3 Electrical Distribution System and the Non-Safety Electrical Distribution System.

The electrical system powers automatic shutdown and decay heat removal functions. The Safety Class 1 portion of the electrical distribution system (includes the direct current (DC)) is limited to supplying power to Safety Class 1 SSCs within the RB rooms.

Safety Class 1 DC power three division (A, B, and C) arrangement supplies DC power to various loads. Each division has a DC battery and two redundant battery chargers powered from the Safety Class 2 and 3 Electrical Distribution System alternating current (AC) power system.

The primary load of the Safety Class 1 Electrical Distribution system is the Distributed Control and Information System (DCIS).

AC Power

AC power modeled in the PSA, is providing medium and low voltage AC power to plant components.

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The following seven AC buses are modeled:

1. Buses A1 and B1 provide 4160V AC power to BOP components such as feedwater pumps and condensate pumps. These buses are not backed by diesel generator.
2. Buses A2 and B2 provide 4160V AC power to other components, such as control rod drive pumps and reactor component cooling pumps. These buses are each backed by a standby diesel generator (SDG) in the event of LOPP.
3. Divisions 1, 2 and 3 provide 480V AC power to Motor Control Centers (MCCs), MOVs, smaller motors, and other plant equipment.

Corium Shield

The corium shield prevents core melt from damaging the containment liner once the core has broken through the bottom of the RPV.

The corium shield is a complementary design feature that includes a refractory material below the RPV. After core damage, in cases where the RPV is assumed to be at a low enough pressure to preclude direct containment heating, the core debris eventually migrates from the core region to the RPV lower head and exits the through a breach (e.g., lower head failure or CRD housing failure). The refractory material below the RPV prevents molten core concrete interaction and any potential ablation of the basemat and accompanying flammable gas generation.

Passive Containment Cooling System

The Passive Containment Cooling System (PCCS) rejects heat into the reactor cavity and/or equipment pool above containment. Supply and discharge connections from the pool are connected to closed loop piping within containment. Heat transfer occurs from the containment to the PCCS by natural convection and condensation.

Containment Filtered Venting

The containment pressure regulation function is venting pressure in the containment as a result of LOCAs, RPV pressure regulation, or core melt. By providing a vent to the containment, pressure, and temperatures are maintained. The release is filtered to reduce the amount of fission products.

Venting is required when no other containment heat removal method succeeds and in certain special situations where there is excessive containment pressure loads beyond the capacity of PCCS. Containment pressure control venting is required to prevent containment overpressurization due to non-condensable gas generation if passive containment heat removal fails. Venting requires operator action to open a pathway to the environment through a hardened vent.

Heating Ventilation and Cooling System

The HVAC system function is providing normal room atmospheric temperature and ventilation control and room cooling during accident scenarios.

Systems assumed to require HVAC for successful operation have placeholder transfer gates with an assumed failure point-estimate screening value used in PSA model quantifications.

Plant Pneumatic System

The Plant Pneumatic System function provides motive and control air or nitrogen to various plant components. The pneumatic system supports multiple systems valves.

15.6.1.11 Level 2 PSA

The Level 2 PSA confirms that the BWRX-300 plant for Darlington meets the SRF and LRF safety goals specified by the CNSC.

The development steps of the Level 2 PSA (NEDC-33946P Reference 15.6-25) are:

- Develop an interface between the Level 1 and Level 2 PSA
- Identify and model safety functions and operator actions
- Perform the containment performance analysis
- Perform the accident progression analysis and develop the Containment Event Trees (CETs)
- Perform the source term analysis
- Quantify the model and interpret results

Interface Between Level 1 and Level 2 PSA Event Tree Definition

The first step in developing the Level 2 PSA is assigning damage classes to the accident sequences resulting in core damage. The BWRX-300 PSA Level 1 Event Tree analysis provides definitions for five damage classes in addition to the successful mitigation state. Table 15.6-6 provides the Level 1 Probabilistic Safety Assessment Damage Class Definitions used in this analysis.

While not strictly required for assessing small and large release frequencies, some element of release timing is also considered in the analysis. For example, if ICS fails to operate during transients, the core can remain adequately cooled for a time by injecting water through the CRD system. If the CRD water inventory is insufficient over the long-term, and if SDC is not initiated before the CRD suction source is depleted, the core can no longer be cooled, and core damage may occur. This is expected to occur significantly later than if CRD did not succeed at all. For this reason, a 'late' designator is appended to damage classes where core damage is significantly delayed. It is expected that this early/late distinction could facilitate applications concerned with dose consequence analysis of reactor accidents.

Core Damage Sequences

Aspects of core damage sequences that are relevant to the potential for radionuclide release characterization are carried forward to the Level 2 PSA. These include:

- RPV pressure at the time of core damage
- Timing of core damage
- Availability/failure of mitigating measures
- Timing of containment failure (relative to core damage)
- Availability of containment isolation

RPV Pressure at Core Damage

If RPV pressure is not lowered substantially during the accident progression, then the potential exists for pressure-driven debris ejection at the time of vessel breach. The resultant fragmentation of the debris may increase the heat transfer to the containment quickly, resulting in a rapid containment heat-up, and pressurization. This direct containment heating is expected to result in containment failure. The potential for this phenomenon warrants development of a CET. The core damage state for high and low RPV pressure are delineated by the accident class. This interface is addressed by developing separate CETs that account for RPV pressure.

Timing of Core Damage

If the core damage occurs very late in the sequence, then these scenarios are given a 'late' designation. This interface is addressed by developing separate CETs that assume core damage occurs very late. The significance of late core damage relative to early core damage scenarios is that protective measures are taken to reduce the impact (e.g., dose to the public) of SA. No quantitative surrogate is developed for what constitutes early versus late damage. Qualitatively, late core damage is assigned for those sequences where safety functions are met initially but fail after a moderate amount of elapsed time (judged to be several hours).

Availability/Failure of Mitigating Measures

Mitigating measures (i.e., equipment and operator actions) and their failures are addressed via fault trees. This interface is addressed inherently by coupling the Level 1 sequences with the Level 2 model so that any basic events relevant to both core damage and radionuclide release will propagate to both levels of the PSA model. To the extent that is possible, the same dependency modeling and basic events are used in both models.

Timing of Containment Failure

If containment is breached or severely degraded as part of the phenomenology of sequence leading to core damage, this interface is addressed by assigning a core damage class that reflects this feature of the sequence.

Availability of Containment Isolation

If core damage occurs as part of a failure to isolate the containment, this interface is addressed by developing a dedicated CET specifically for that class of core damage.

Level 2 Mitigation

The systems and human actions used in the PSA model were taken from the Level 1 PSA (that contains aspects of containment performance, such as containment venting) and no new system or operator actions are developed. The relevant functions to the Level 2 PSA analysis are:

- Late inventory injection
- Containment isolation
- Corium shield
- PCCS
- Containment venting
- RPV depressurization

The effects these functions have on the Level 2 accident sequences is described in the Level 2 Accident Sequence Analysis because their impact on radionuclide release and retention is sequence-specific.

Containment Performance

The following containment failure modes are considered in the Level 2 PSA:

- Overpressure
- Failure of containment isolation
- Basemat melt-through
- Hydrogen deflagration
- Venting
- Steam explosion
- Containment bypass
- Direct containment heating

Containment Overpressure

No detailed ultimate containment performance assessment is available at this stage in the BWRX-300 design. However, the containment failure pressure is assumed to be 1.5 MPa (210 psig) for this evaluation. If the containment vent fails to operate, then the containment is assumed to fail.

Fail-to-scram sequences result in severe dynamic forces inside containment during RPV failure that result in gross containment failure.

Containment Isolation/Bypass

Each core damage scenario requires containment isolation. For core damage scenarios resulting from breaks outside containment (e.g., breaks in the main steam lines outside containment), that the core damage event results in a simultaneous large release due to the inability to isolate containment.

Basemat Melt-Through

BWRX-300 is equipped with a corium shield that covers the basemat. This corium shield prevents MCCI. However, for the corium shield to function, the downward heat flux between the ex-vessel debris bed must be limited by overlying water pool. If debris is not covered with water, then the limited upward heat flux and steaming results in failure to quench the molten debris pool and results in a containment basemat melt-through.

Hydrogen Deflagration

During the SA progression, cladding oxidation results in significant hydrogen generation. If the accident occurs when containment is not inerted (e.g., purging nitrogen from containment prior to a refueling outage), then it is assumed that the hydrogen will ignite, and the dynamic forces result in containment failure.

Containment Venting

In scenarios where the containment pressure is increases beyond the design pressure, the containment vent is initiated. The vent is equipped with a filter. Failure to vent when required is results in containment overpressure.

Direct Containment Heating

Direct containment heating occurs in a SA if the RPV fails while at high pressure. The resultant failure is accompanied by fragmentation of debris and entrainment in any effluent post-RPV

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rupture. The increased surface area of the debris results in rapid heat-up and pressurization of containment that leads to gross containment failure.

Steam Explosion

If an unquenched molten pool of corium comes into contact with a pool of water, the potential exists for there to be a rapid boiling event, further fragmenting the corium, increasing the heat transfer surface between debris and water. The fragmentation and entrainment of corium and rapid boiling and expansion of steam and water is a steam explosion. The dynamic effects of this event result in gross containment failure. The likelihood of this event to occur due to reflooding in-vessel is judged to be very small. The likelihood of the event to occur due to vessel failure and debris entry into an underlying pool of water is small, but more likely than in-vessel. While the dynamics of lower head failure are uncertain, it is likely that low pressure vessel failure results in fractional corium relocation to the underlying pool rather than a total "dump" of the entire corium pool. By limiting the amount of corium that relocates to the lower pool at one time, the potential for steam explosions is reduced.

This section describes the Level 2 accident sequences for each CET.

Class I/I-L

Core damage sequences that occur with the RPV at low pressure and that nominally have containment intact are Class I. There are core damage sequences that occur in Class I core damage sequences that occur "late". As a result a CET is developed for both "early" and "late" Class I core damage. The node descriptions in this section apply to the "early" Class I core damage scenarios. Late core damage sequences have the same general progression, but any early release categories are changed to late releases for late core damage events.

Class II

The containment isolation function is reassigned to the Level 2 CETs and given the definition of Class II core damage (core damage resulting from containment failure due to overpressure). In the BWR operating fleet, Class II core damage may result from injection without containment decay heat removal. The resultant overpressure of containment leads to failure and loss of the injection source due to loss of net positive suction head (NPSH), adverse equipment environments, and disruption of injection lines. Core cooling is provided by ICS, and no separate containment heat removal method is required. If core cooling is provided by CRD, then there must be a letdown path outside containment established to remove inventory heated in the RPV. So, failure of decay heat removal results in core damage due to core cooling failures before containment failure for evaluated sequences in the BWRX-300 PSA.

Class III/III-L

Core damage sequences that occur with the RPV at high pressure and that nominally have containment intact are Class III. There are core damage sequences that occur with Class III that occur "late", and as a result, CET is developed for both "early" and "late" Class III core damage. The node descriptions in this section apply to the "early" Class III core damage scenarios. Late core damage sequences have the same general progression, but early release categories are changed to late releases for late core damage events.

Class IV

Class IV core damage scenarios are failure to scram scenarios. If core damage occurs from failure to scram scenarios, SA progression results in more severe RPV and containment conditions than those scenarios with successful negative reactivity insertion. Generally, a core damage scenario for failure to scram in a BWRX-300 is expected to have core damage occurring

at elevated RPV pressure or pre-core damage energetic RPV failure due to overpressure/overpower. The containment loading resulting from these scenarios is expected to exceed the containment capacity and gross failure of containment is expected to occur.

Class V

Class V sequences are those with an initial bypass of containment. This containment bypass provides a radionuclide release path, so any Class V sequences result in large early releases.

Class VR

The VR core damage sequences involve excessive LOCAs in the mid and lower vessel regions and successful operation of the containment vent, but due to the size and location of the breaks core damage cannot be prevented.

15.6.1.11.1 Source Term Analysis

Two releases are of special interest in the source term analysis: small and large releases. Small and large releases are defined in CNSC REGDOC-2.5.2 (Reference 15.6-9). Small releases are considered those that release more than 10^{15} Bq of Iodine-131 to the environment. Large releases are those that release more than 10^{14} Bq of Cesium-137 to the environment. Sequences that meet the criteria for both small and large releases are counted as large releases, rather than both. Source terms are determined using MAAP. Release calculations are performed in MAAP for representative sequences. Release categories are then assigned to those without specific MAAP runs based on the calculations performed. This approach ensures that significant scenarios with realistic release categories are described in detail in the Level 2 PSA.

MAAP generally plots radionuclide release as a fraction of total inventory for select "groups" of fission products. The groups of fission products are subdivided by molecular compound and individual isotopic transport is not specifically tracked. However, MAAP provides the requisite information to understand the fractional distribution of all isotopes of a given element across fission product groups. It is assumed that all isotopes of a given element behave identically for the purposes of group assignment and subsequent transport. For example, if 50% of element X is in Fission Product Group 1, and all of group 1 is released, then 50% of every isotope of X is released in group 1.

Refer to Chapter 3, Appendix 3E for additional discussion regarding the MAAP computer code.

15.6.1.12 Model Integration and Quantification

The Level 2 PSA model is integrated with the Level 1 PSA model. The Level 1 accident sequences are placed under "collector gates" that group core damage sequences into classes.

The Level 2 ETs are constructed with the entry branch name identical to the applicable Level 1 Core Damage class collector gate. Functional node branches are given names identical to those in the functional fault tree file. When the event trees are converted into fault tree logic (creating accident sequence logic) and merged with functional fault tree logic, an integrated, quantifiable fault tree is developed.

The fault tree is quantified using the following PSA model development codes (see Chapter 3, Appendix 3E):

- CAFTA
- PRA Quant

In addition, FTREX 1.8 is used as the quantification/cutset generation engine and SysImp is used to develop importance measures for systems, components, and HFEs.

Intact, small release, and large release frequencies are quantified at a truncation of $1\text{E-}15/\text{yr}$. The cutsets are examined to ensure there was $<5\%$ top event frequency drop if the truncation limit are set to $1\text{E-}14/\text{yr}$. This provides evidence that no risk-significant sequences were truncated.

15.6.2 Results of the Level 1 Probabilistic Safety Assessment

15.6.2.1 Initiating Events

The IEs were derived for internal events at-power and for low power shutdown states. The IEs are tabulated with descriptions and frequencies in Table 15.6-7 and Table 15.6-8.

15.6.2.2 Mitigating Systems Modeled

The following front-line mitigating systems are modeled to mitigate the IEs. These mitigating systems or functions are:

- Shutdown Cooling (SDC)
- Isolation Condenser System (ICS)
- Containment Isolation (CI)
- Ultimate Pressure Relief
- Control Rod Insertion
- Boron Injection
- Control Rod Injection makeup
- Corium Shield
- Containment Venting
- Passive Containment Cooling System (PCCS)
- Cooling Water System
- DC Power
- Feedwater Runback
- Reactor Pressure Vessel (RPV) Isolation
- AC Power
- Heating Ventilation and Cooling
- Plant Pneumatics

Level 1 Event Tree Endpoints

The Level 1 PSA event tree end points end in one success or five core damage states. The five core damage states are described below.

Event sequences are qualitatively analyzed to determine which sequences lead to core damage end states. The core damage sequences are grouped together based upon the overall challenge to the containment barrier and defined as:

1. Containment state: intact or breached
2. RPV pressure

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3. Energy deposition into containment
4. Containment bypass

The above considerations result in the following event tree end points:

1. OK: The core is successfully cooled, and the containment is intact. There is no core damage in these events.
2. CD I: The containment is intact when core damage occurs and the RPV is at low (or controlled) pressure.
3. CD II: The containment is breached, either due to overpressurization or venting, while the core is successfully cooled. Core damage results from failure of maintaining long-term heat removal.
4. CD III: The containment is intact when core damage occurs and there is high RPV pressure at the time of RPV failure.
5. CD IV: CD IV: Core damage results from an accident sequence with an initial failure of effective reactivity control (failure to scram due to failure of RPS, control rod binding, etc.). This has the potential to affect containment more severely than the CD I and CD III because more energy is deposited into containment prior to RPV failure. All CD IV end states could be treated as CD I or CD III (depending on the RPV pressure) without affecting the results of the containment analysis. This end state is retained in this scoping analysis to more readily allow for sensitivity analyses related to reactivity control.
6. CD V: The containment is bypassed at the time of core damage.
7. CD VR: Core damage occurs due to RPV ruptures in the lower or mid-vessel regions.

Event Tree Transition – Level 1 to Level 2 PSA

The core damage Level 2 PSA accident sequence analysis is:

1. Class 1/1-L
Core damage sequences occur with the RPV at low pressure and nominally have containment intact are Class I. There are core damage sequences that occur with Class I that occur “late” and CET is developed for both “early” and “late” Class I core damage.
2. Class II
Class II core damage (core damage resulting from containment failure due to overpressure). Class II core damage often results from injection without containment decay heat removal. The resultant overpressure of the containment leads to failure and loss of the injection due to loss of NPSH, adverse equipment environments, and disruption of injection lines similarly to the Level 1 PSA Core Damage Frequencies.
3. Class III/III-L
Core damage sequences that occur with the RPV at high pressure and that nominally have containment intact are Class III. There are core damage sequences that occur with Class III that occur “late”, and a CET is developed for both “early” and “late” Class III core damage.

4. Class IV

Class IV core damage scenarios are failure to scram and failure to depressurize scenarios. If this core damage scenario occurs, SA progression results in more severe RPV and containment conditions than those scenarios with successful negative reactivity insertion. The containment loading resulting from these scenarios is expected to exceed the containment capacity and gross failure of the containment is expected to occur.

5. Class V

Class V sequences are those with core damage and an initial containment bypass. Containment bypass provides a radionuclide release path. Any Class V sequences result in large early releases.

6. Class VR

Core damage occurs due to RPV ruptures in the lower or mid-vessel regions, the containment vent functions.

Small Release Frequencies

The SRFs for the Level 2 PSA are not reported, because achieving a small release without achieving large release is within a very tight release band. If a sequence becomes SRF, it is expected the SRF sequence deteriorates to LRF.

Large Release Frequencies

The LRF for the Level 2 PSA At-Power, Shutdown and Low Power, seismic, fire, high wind and internal flood are listed Table 15.7.9-2.

15.6.3 Probabilistic Safety Assessment Insights and Applications

The Darlington BWRX-300 PSA activities are ongoing as the design progresses from conceptual to detailed. The PSA scope includes:

- Level 1 PSA
 - Internal Events at-Power and Low Power Shutdown states
 - External Events at-Power and Low Power Shutdown states, (seismic fire, high wind, and flood)
- Level 2 PSA At-Power and Low Power and Shutdown

The CNSC REGDOC-2.4.2, Section 3 (Reference 15.6-1) objectives (a thru h) are:

- a. Provide a systematic analysis
- b. Demonstrate a balanced design that no particular feature or postulated IE makes a disproportionate contribution to overall risk
- c. Provide confidence of no cliff-edge effects
- d. Provide assessments of quantitative safety goals
- e. Provide site-specific assessments of external hazards
- f. Identify facility vulnerabilities and systems where design improvements or modifications to operational procedures could reduce probabilities of severe accidents or mitigate consequences

- g. Assess adequacy of emergency operating procedures
- h. Provide insights into severe accident management

The BWRX-300 PSA assesses the design to meet the CNSC REGDOC-2.4.2 (Reference 15.6-1) objectives. PSA objectives (a to f) are assessed at this stage of the design and objectives (g and h) are assessed once the detailed design and operational procedures are available.

Concern that small changes can lead to large differences in plant effects – i.e., cliff-edge effects are largely ameliorated by the demonstrated margin between plant risk metrics and the CNSC REGDOC-2.5.2 (Reference 15.6-9) safety goals. Within the PSA accident sequence and success criteria analysis, no cliff-edge effects were discovered where plant parameters approach critical thresholds where accident sequences are expected to follow a significantly more severe pathway. However, uncertainty in these analyses will be formally explored in more detail in future PSA work.

Tables 15.7.9-1 and 15.7.9-2 show that the PSA CDF and LRF results to date meet the CNSC safety goals.

CNSC REGDOC-1.1.2, Section 4.4.4 (Reference 15.6-8) states that hazards analyses be performed. As presented in this section, internal and external hazards are identified and screened for PSA. Natural external hazards and human-induced external and internal hazards are assessed. There is limited assessment of external and internal hazards interaction. A more detailed hazard combination assessment is performed commensurate with the design progress.

CNSC REGDOC-1.1.2, Section 4.4.5 (Reference 15.6-8) states that a PSA be included in the Licence to Construct application. The PSA identifies facility vulnerabilities. The PSA application should also:

1. Verify EOPs are adequate during commissioning and future operation
2. Provide insights into the Severe Accident Management (SAM) program
3. Describe how PSA is used during commissioning and future operation

Based on the PSA work to date, CNSC REGDOC-1.1.2, Section 4.4.5 (Reference 15.6-8) requirements are not met at this time because Operating Manuals, EOPs and SAM program are not defined. The PSA is updated when the design matures and operational documents including Operating Manuals, EOPs, and SAM program are developed.

PSA is further updated for the Operating Licence Application as the design is finalized and operational information matures.

15.6.3.1 Summary

The PSA CDF, SRF, and LRF results to date meet the CNSC safety goals.

Section 15.6 is a high-level summary of the Internal and External Event Hazard Screening, and Level 1 and Level 2 PSA status supporting the BWRX-300 PSAR.

The PSA scope:

1. Level 1 PSA Internal At-Power, Low Power Shutdown States
2. Level 1 PSA for external events includes a power seismic and high wind analysis
3. Level 2 PSA At-Power

The CNSC REGDOC-2.4.2, Section 3 (Reference 15.6-1) objectives are reviewed and six of the eight objectives of CNSC REGDOC-2.4.2 (Reference 15.6-1) are met. Two other objectives (adequacy of EOPs and insights on SAM) are not assessed because the plant design is conceptual and the EOP procedures and SAM program are prepared as the design is finalized.

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CNSC REGDOC-1.1.2, Section 4.4.4 (Reference 15.6-8) requires a hazard analysis. Internal and external hazards are identified and screened. Natural external hazards, human-induced external and internal hazards are assessed.

CNSC REGDOC-1.1.2, Section 4.4.5 (Reference 15.6-8) states a PSA is conducted as required by CNSC REGDOC-2.4.2 (Reference 15.6-1) and includes:

1. Verification that EOPs are adequate during commissioning and future activities
2. Provide insights into severe accident management program
3. Describe how PSA is used during commissioning and future operation to identify design improvements, modifications to operational procedures to reduce probabilities of severe accidents and mitigate consequences.

CNSC REGDOC-1.1.2, Section 4.4.5 (Reference 15.6-8) requirements are addressed after Operating Manuals, EOPs and SAM programs are defined, and the PSA is updated.

The PSA CDF, SRF, and LRF results “to date” indicate that the CNSC safety goals are met.

The SRF is not reported, because achieving a small release without achieving large release would be within a very tight release band. If a sequence becomes SRF, it is expected the SRF sequence deteriorates to an LRF. The LRF requirement is more restrictive than SRF. Therefore, SRF is conservatively grouped with LRF.

The CDF, Small Release Frequency, and LRF results meet the CNSC safety goals as shown in Tables 15.7.9-1 and 15.7.9-2.

15.6.4 References

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Table 15.6-1: Internal Hazard Qualitative Screening Analysis

Hazard	Screening Criterion (Refer to Subsection 15.6.1.2)
Heavy Load Drop	Not screened qualitatively
Release of Chemicals from On-Site Storage	1
Turbine-Generated Missiles	Not screened qualitatively
Other Internally-Generated Missiles	1, 2
Fires	Not screened qualitatively. Covered by BWRX-300 Internal Fire Scoping Evaluation
Explosions	Not screened qualitatively. Covered by BWRX-300 Internal Fire Scoping Evaluation
Collapse of Structures	4, 5
Pipe Whips	4
Jet Effects	4
Internal Flooding	Not subject to screening

Table 15.6-2: Internal Hazard Quantitative Screening Analysis

Hazard	Quantitative Screening Results
Turbine-Generated Missiles	Screened from further PSA analysis
Internal Flooding	Covered by Internal Flooding PSA
Heavy Load Drop	Not screened out quantitatively

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Table 15.6-3: External Hazards Qualitatively Screened Out of the PSA

Hazard Type	Screen Qualitatively
Individual Hazards	Extreme Snow, Extreme Hail, Mist, White Frost, Drought, Salt Storm, Sand Storm, Lightning, Explosion Outside Plant, Explosion after Pipeline Accident, Chemical Release Outside or Inside Site, Chemical Release after Transportation Accident, Chemical Release after Pipeline Accident, Magnetic Disturbance, Land Rise, Soil Frost, Animals, Volcanic Phenomena, Avalanche, Above-Water Landslide, External Fire, Excavation Work, Direct Impact from heavy transportation within Site, Missiles from military activity, Internal Fire Spreading from Other Plant, Contamination from Chemicals, Strong Water Current, Low Water Level, High Water Temperature, Low Water Temperature, Underwater Landslide, Surface Ice, Frazil Ice, Organic Material in Water, Corrosion from Salt Water, Solid or Fluid Impurities from Ship Release, Chemical Release to Water, Direct Impact from Ship Collision.
Co-Existent Hazards	High Air and Water Temperature; Low Air and Low Water Temperature; Extreme Rain and Lightning; Drought and Low Water Level; Drought and External Fire; Explosion Outside Plant and Chemical Release; Explosion after Transportation Accident and Chemical Release after Transportation Accident; Explosion after Pipeline Accident and Chemical Release after Pipeline Accident; Chemical Release and Contamination from Chemicals; Extreme Snow and Earthquake.

Table 15.6-4: External Hazards Quantitative Assessment Summary

Hazard	Initiating Event Frequency (yr)	CCDP	CDF / LRF (yr)	Quantitative Screening
Seismic	Ground motion bin frequencies are assigned by the seismic PSA based on the seismic hazard	This hazard is selected for PSA development	This hazard is selected for PSA development	This hazard is selected to proceed directly to detailed PSA
Internal Fire	Fire scenario frequencies are assigned by the fire PSA	This hazard is selected for PSA development	This hazard is selected for PSA development	This hazard is selected to proceed directly to detailed PSA
High Wind, including high wind hazard combinations: Strong Wind and Extreme Air Pressure; Strong Wind and Ice barriers; Tornadoes and Extreme hail; Wind-Driven Precipitation	Wind speed bin frequencies are assigned by the high wind PSA based on the straight wind and tornado hazards	This hazard is selected for PSA development	This hazard is selected for PSA development	This hazard is selected to proceed directly to detailed PSA
High Air Temperature	5.0E-2	3.1E-9	1.5E-10	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.
Low Air Temperature	2.8E-3	3.1E-9	8.5E-12	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.
Meteorite	5.3E-10	1.0E+0	5.3E-10	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.
Explosion after Transportation Accident	3.0E-6	4.1E-7	1.2E-12	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.

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Hazard	Initiating Event Frequency (/yr)	CCDP	CDF / LRF (/yr)	Quantitative Screening
Satellite Crash	7.8E-8	4.1E-7	3.2E-14	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.
Airplane crash	3.4E-12	1.0E+0	3.4E-12	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.
High Water Level	1.0E-2	3.1E-9	3.1E-11	CDF and LRF are below 1E-8 and 1E-9/yr, respectively. This hazard screens quantitatively.

(1) The total contribution for quantitatively screened external hazards is also less than 5% of the total 1E-7/yr CDF and LRF assumed for the BWRX-300.

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Table 15.6-5: Generic List of Boiling Water Reactor Transient Initiating Events

No.	Initiating Event
1	Electric load rejection
2	Electric load rejection with TBV failure
3	Turbine trip
4	Turbine trip with TBV failure
5	MSIV closure
6	Inadvertent closure of one MSIV
7	Partial MSIV closure
8	Loss of normal condenser vacuum
9	Pressure regulator fails open
10	Pressure regulator fails closed
11	Inadvertent opening of a safety/relief valve (stuck)
12	Turbine bypass fails open
13	Turbine bypass or control valves cause increased pressure (closed)
14	Recirculation control failure - increasing flow
15	Recirculation control failure - decreasing flow
16	Trip of one recirculation pump
17	Trip of all recirculation pumps
18	Abnormal start-up of idle recirculation pump
19	Recirculation pump seizure
20	Feedwater - increasing flow at-power
21	Loss of feed water heater
22	Loss of all feed water flow
23	Trip of one feed water pump (or condensate pump)
24	Feedwater - low flow
25	Low feedwater flow during start-up or shutdown
26	High feedwater flow during start-up or shutdown
27	Rod withdrawal at-power
28	High flux due to rod withdrawal at start-up
29	Inadvertent insertion of rod or rods
30	Detected fault in reactor protection system
31	Loss of offsite power
32	Loss of auxiliary power (loss of auxiliary transformer)

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No.	Initiating Event
33	Inadvertent start-up of Inadvertent start-up of High Pressure Core Injection System/ High Pressure Core Spray System
34	Scram due to plant occurrences
35	Spurious trip via instrumentation, RPS fault
36	Manual scram- no out-of-tolerance condition
37	Cause unknown

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Table 15.6-6: Level 1 PSA Damage Class Definitions

Class	Definition
I	The containment is nominally intact when core damage occurs. Core damage occurs with the RPV at low/controlled pressure.
II	The containment is breached due to overpressurization or venting during the accident, but before core damage occurs. Core damage occurs due to failure of long-term decay heat removal post-containment failure.
III	The containment is nominally intact when core damage occurs. Core damage occurs with the RPV at high pressure.
IV	Failure to scram occurs with failure to insert sufficient negative reactivity prior to core damage.
V	Core damage occurs with the containment bypassed.
VR	Excessive LOCAs in mid and lower vessel regions and successful operation of containment vent but due to size and location of the breaks, core damage cannot be prevented.

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Table 15.6-7: Internal Initiating Events At-Power

IE Name Label	Description	Frequency (per RX yr.)	Comments
%TRANS	General Transient	7.4E-01	
%COND	Loss of Condenser Heat Sink	1.1E-01	Includes MSCIV and MSRIV Closure and Loss of Circulating Water
%FW	Loss of Feedwater (ALL)	6.0E-02	
%LOPP	Loss of offsite power, all categories, power operation	3.1E-02	This is the total frequency of all LOPP events and represents the other "LOOP-IEs" listed here
%LOPP-G	Loss of offsite power, grid-related, power operation	1.1E-02	
%LOPP-P	Loss of offsite power, plant-centered, power operation	2.0E-03	
%LOPP-S	Loss of offsite power, switchyard-centered, power operation	1.3E-02	
%LOPP-W	Loss of offsite power, weather-related, power operation	6.0E-03	
%MSLB-OUT	Main Steam Line BOC	3.0 E-03	MS Break Inside containment contained within the LOCA Frequency
%FLB-Out	Feedwater Line BOC	6.0E-04	FW Break Inside containment contained within the LOCA Frequency
%E-LOCA	Excessive LOCA	7.0E-09	Includes Vessel Rupture
%LLOCA-I	Large LOCA – Isolable	3.6E-06	
%LLOCA-N	Large LOCA – Non-isolable	6.1E-07	
%MLOCA-I	Medium LOCA – Isolable	3.1E-05	
%MLOCA-N	Medium LOCA – Non-isolable	1.6E-06	
%SLOCA-S-I	Small LOCA Steam – Isolable	1.6E-05	
%SLOCA-S-N	Small LOCA Steam – Non-isolable	2.6E-07	
%VSLOCA	Very-Small LOCA	3.4E-03	

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IE Name Label	Description	Frequency (per RX yr.)	Comments
%SLOCA-W-I	Small LOCA Water– Isolable	6.7E-05	
%SLOCA-W-N	Small LOCA Water – non-isolable	2.5E-07	
%A1	Loss of 4160V AC Bus A1	3.4E-03	Reactor Trip
%B1	Loss of 4160V AC Bus B1	3.4E-03	Reactor Trip
%A2	Loss of 4160V AC Bus A2	3.4E-03	Manual Shutdown
%B2	Loss of 4160V AC Bus B2	3.4E-03	Manual Shutdown
%A2-U	Loss of 4160V AC Bus A2-Unknown	3.4E-03	Manual Shutdown
%B2-U	Loss of 4160V AC Bus B2-Unknown	3.4E-03	Manual Shutdown
%DIV1	Loss of 480V AC Bus Division 1 (4 buses total)	2.3E-02	Manual Shutdown
%EB-LGA	Loss of Bus EB LGA	5.7E-03	Manual Shutdown
%DIV2	Loss of 480V AC Bus Division 2 (4 buses total)	2.0E-02	Manual Shutdown
%EB-LGB	Loss of Bus EB LGB	6.0E-03	Manual Shutdown
%DIV3	Loss of 480V AC Bus Division 3 (4 buses total)	2.2E-02	Manual Shutdown
%VDC-A	Loss of Bus 250V DC-A	1.0E-03	Manual Shutdown
%VDC-B	Loss of Bus 250V DC-B	1.0E-03	Manual Shutdown
%VDC-C	Loss of Bus 250V DC-C	1.0E-03	
%PWC-All	Loss of All PCW	2.0E-04	
%PCW-A	Loss of A Train PCW	1.8E-03	
%PCW-B	Loss of B Train PCW	1.8E-03	
%IA	Loss of All Instrument Air	7.2E-03	

Table 15.6-8: Internal Initiating Events At-Power and Shutdown

POS -->		At-Power Per Calendar Year	Hot Shutdown and Stable Shutdown	Cold Shutdown	Refueling (reactor cavity drained)	Refueling (reactor flooded to normal with the fuel pool gate installed)	Refueling (reactor flooded to normal with fuel pool gate removed)
Time in POS%		90	0.11	0.11	0.4	0.3	4.4
Initiator	Description	Mean	Mean	Mean	Mean	Mean	Mean
%LOOP-G	LOOP, grid-related, power operation	9.90E-03	1.10E-05	1.21E-04	4.40E-05	3.30E-05	4.84E-04
%LOOP-P	LOOP, plant-centered, power operation	1.80E-03	2.00E-06	2.20E-05	8.00E-06	6.00E-06	8.80E-05
%LOOP-S	LOOP, switchyard-centered, power operation	1.20E-02	1.33E-05	1.47E-04	5.33E-05	4.00E-05	5.87E-04
%LOOP-W	LOOP, weather-related, power operation	5.40E-03	6.00E-06	6.60E-05	2.40E-05	1.80E-05	2.64E-04
%MSLB-OUT	Main Steam Line Break Outside Containment	2.70E-03	3.00E-06	N/A	N/A	N/A	N/A
%FLB-OUT	FW Line Break Outside Containment	5.40E-04	6.00E-07	N/A	N/A	N/A	N/A
%E-LOCA	Excessive LOCA	7.0E-09	7.8E-12	8.6E-11	3.1E-11	2.3E-11	3.4E-10
%LLOCA-I	Large LOCA – Isolable	3.3E-06	3.7E-09	N/A	N/A	N/A	N/A
%LLOCA-N	Large LOCA – Non-isolable	5.5E-07	6.1E-10	N/A	N/A	N/A	N/A
%MLOCA-I	Medium LOCA – Isolable	2.8E-05	3.1E-08	N/A	N/A	N/A	N/A
%MLOCA-N	Medium LOCA – Non-isolable	1.4E-05	1.6E-08	N/A	N/A	N/A	N/A
%SLOCA-S-I	Small LOCA Steam – Isolable	2.3E-05	2.6E-08	N/A	N/A	N/A	N/A
%SLOCA-S-N	Small LOCA Steam – Non-isolable	1.4E-07	1.6E-10	N/A	N/A	N/A	N/A
%VSLOCA	Very-Small LOCA	3.1E-03	3.4E-06	N/A	N/A	N/A	N/A

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POS -->		At-Power Per Calendar Year	Hot Shutdown and Stable Shutdown	Cold Shutdown	Refueling (reactor cavity drained)	Refueling (reactor flooded to normal with the fuel pool gate installed)	Refueling (reactor flooded to normal with fuel pool gate removed)
Time in POS%		90	0.11	0.11	0.4	0.3	4.4
Initiator	Description	Mean	Mean	Mean	Mean	Mean	Mean
%SLOCA-W-I	Small LOCA Water-- Isolable	6.0E-05	6.7E-08	N/A	N/A	N/A	N/A
%SLOCA-W-N	Small LOCA Water -- non- isolable	2.3E-07	2.6E-10	N/A	N/A	N/A	N/A
%A2	Loss of 4160V AC Bus A2	5.2E-03	5.8E-06	6.4E-05	2.3E-05	1.7E-05	2.5E-04
%B2	Loss of 4160V AC Bus B2	5.2E-03	5.8E-06	6.4E-05	2.31E-05	1.7E-05	2.5E-04
%DIV1	Loss of 480V AC Bus Division 1 (four buses total)	3.1E-03	3.4E-06	3.8E-05	1.4E-05	1.0E-05	1.5E-04
%DIV2	Loss of 480V AC Bus Division 2 (four buses total)	3.1E-03	3.4E-06	3.8E-05	1.4E-05	1.0E-05	1.5E-04
%PCW-All	Loss of All PCW	1.8E-04	2.0E-07	2.2E-06	8.0E-07	6.0E-07	8.8E-06
%PCW-A	Loss of A Train PCW	1.6E-03	1.8E-06	1.98E-05	7.2E-06	5.4E-06	7.9E-05
%PCW-B	Loss of B Train PCW	1.6E-03	1.8E-06	2.0E-05	7.2E-06	5.4E-06	7.9E-05
%IA	Loss of All Instrument Air	6.5E-03	7.2E-06	7.9E-05	2.9E-05	2.2E-05	3.2E-04
%SDC (1 Trn)	Loss of Shutdown Cooling (Both train)	2.6E-01	2.6E-04	N/A	N/A	N/A	N/A
%SDC (2 Trns)	Loss of Shutdown Cooling (Both train)	7.54E-03	N/A	8.3E-05	3.0E-05	2.3E-05	1.8E-07

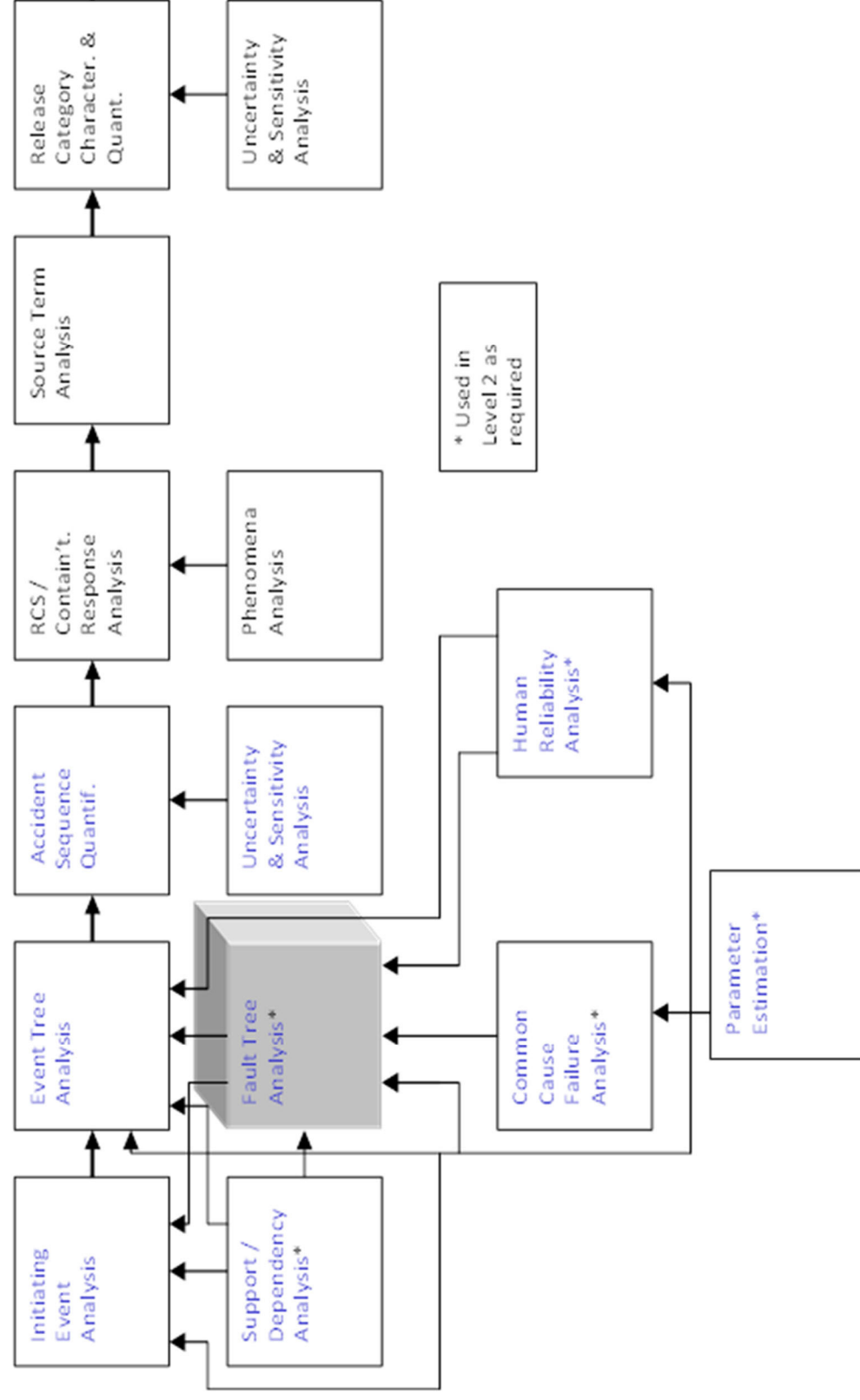


Figure 15.6-1: Principal Steps in Probabilistic Safety Assessment

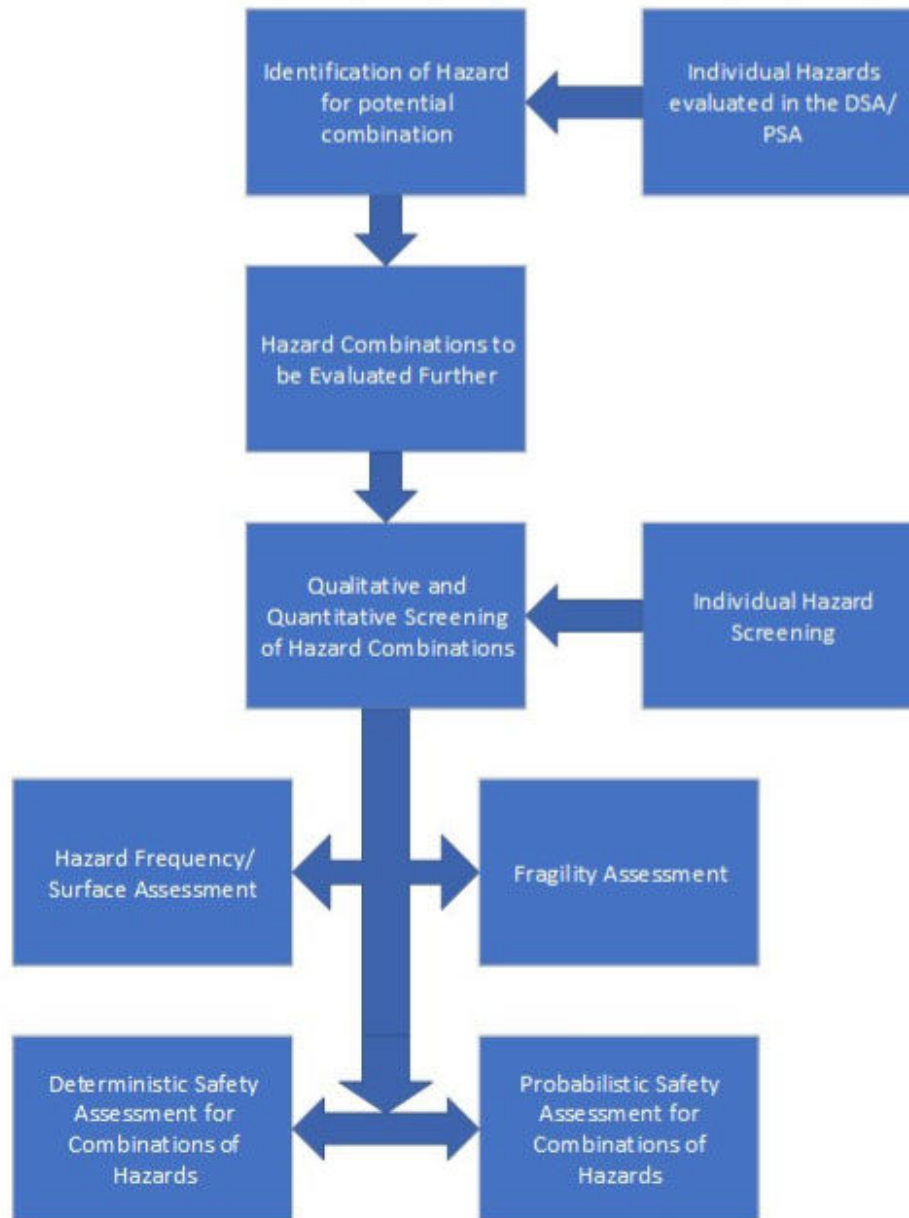


Figure 15.6-2: Overall Process for Selection, Screening and Analysis of Hazards

15.7 Results of the Deterministic Safety Analyses and Probabilistic Safety Assessment

The DSA results for the bounding BWRX-300 events meet acceptance criteria in Tables 15.3-2 and 15.3-3, respectively. Results Summary for the DSA is provided in the following tables:

Table 15.7.2-1	Results Summary of the AOO Safety Analyses
Table 15.7.3-1	Results Summary of DBA and DEC Events for Non-LOCA
Table 15.7.3-2	Results Summary of the Fuel Loading Error Event
Table 15.7.3-3	Results Summary of DBA and DEC Events for LOCA
Table 15.7.6-1	Fuel Handling Accident Dose Consequences
Table 15.7.7-1	BWRX-300 DNNP Main Steam Line Break Outside Containment – Accident Dose Consequences
Table 15.7.7-2	BWRX-300 DNNP Feedwater Line Break Outside Containment - Accident Dose Consequences
Table 15.7.7-3	BWRX-300 DNNP Isolation Condenser System Line Break Outside Containment – Accident Dose Consequences
Table 15.7.7-4	BWRX-300 DNNP Instrument Line Break Outside Containment Accident Dose Consequences

The results for the beyond design basis events evaluated in the PSA are provided in Table 15.7.9-1 “Comparison of PSA Results with CNSC REGDOC-2.5.2 Limits” indicate that the design does not exceed the safety goals in Table 15.3-3.

Implementation of the D-in-D concept ensures multiple, independent layers of protection against unacceptable radiation releases. None of the bounding AOOs, DBAs or DEC Events Without Core Damage analyzed approach the regulatory limits for radioactive releases.

15.7.1 Results of Analysis of Normal Operation

The analysis of normal operations for stability is described in Chapter 4, Section 4.8 and Section 15.5.2.4. Subsection 15.5.2 describes the normal operations DSA. The OLC (Chapter 16) provides the operating limits and parameters ensuring the plant is operating within the design basis.

15.7.2 Results of Analysis of Anticipated Operational Occurrences

The analysis of AOOs is described in Subsection 15.5.3. The resulting maximum neutron flux, dome pressure, RPV bottom pressure, simulated thermal power, and Δ CPR/ICPR for AOOs is provided in Table 15.7.2-1.

15.7.3 Results of Analysis of Design Basis Accidents

The analysis of DBAs is described in Subsection 15.5.4. The summary results of maximum neutron flux, maximum dome pressure, maximum RPV bottom pressure, maximum simulated thermal power (%NBR), and peak clad temperature from non-LOCA DBAs are provided in Table 15.7.3-1. The LOCA DBA summary results for peak cladding temperature, peak containment pressure, and peak containment shell temperature are provided in Table 15.7.3-3.

15.7.4 Results of Analysis of Design Extension Conditions Without Core Damage

The analysis description of DEC events without core melting is described in Section 15.5.5. The summary results of maximum neutron flux, maximum dome pressure, maximum RPV bottom

pressure, maximum simulated thermal power, and peak clad temperature from non-LOCA DEC are provided in Table 15.7.3-1.

The LOCA DEC summary results for peak cladding temperature, peak containment pressure, and peak containment shell temperature are provided in Table 15.7.3-3. The summary results of a fuel loading error event DEC are provided in Table 15.7.3-2.

15.7.5 Results of Analysis of Design Extension Conditions with Core Damage

The analysis description of those DEC events associated with core damage are described in Section 15.6.3 Level 2 PSA.

15.7.6 Results of Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Fuel Pool

The analysis description of those postulated events and accident scenarios associated with the fuel pool are described in Subsection 15.5.7.

15.7.7 Results of Analysis of Fuel Handling Events

The analysis of FHA is described in Subsection 15.5.8. The resultant dose consequences from an FHA is provided in Table 15.7.6-1.

15.7.8 Results of Analysis of Radioactive Releases from a Subsystem or a Component

The analysis description of the liquid and gaseous radioactive waste system process scenario releases from a postulated liquid tank failure or off-gas system failure are described in Chapter 11, Subsections 11.2.9, and 11.3.14, respectively. The off-gas system failure results are provided in Table 11.3-4 and meet the CNSC allowable limits.

15.7.9 Results of Analysis of LOCA Breaks Outside Containment

LOCA from pipe breaks outside containment described in Subsection 15.5.9.2 result in normal coolant concentration releases that are within allowable limits.

15.7.9.1 Results of Main Steam Line Break Outside Containment

The analysis of a MSL break outside containment is described in Subsection 15.5.9.2.1 and the results are provided in Table 15.7.7-1.

15.7.9.2 Results of Large Feedwater Pipe Break Outside Containment

The analysis of a large feedwater pipe break outside containment is described in Subsection 15.5.9.2.2, and the results are provided in Table 15.7.7-2.

15.7.9.3 Results of Large Isolation Condenser Pipe Break Outside Containment

The analysis of a large ICS pipe breaks outside containment is described in Subsection 15.5.9.2.3 and the results are provided in Table 15.7.7-3.

15.7.9.4 Results of Small Breaks Outside Containment

The analysis of a small pipe break outside containment is described in Subsection 15.5.9.2.4 and the results are provided in Table 15.7.7-4.

15.7.10 Results of Analysis of Internal and External Hazards

The description of internal and external hazards and associated design basis SSCs mitigating these hazards are described in Section 15.5.10 for the DSA.

15.7.11 Results of Probabilistic Safety Analysis

The Level 1 and Level 2 PSA are described in Subsection 15.6.2. The general PSA approach and the insights and applications are described in Subsections 15.6.1 and 15.6.3. The comparison of PSA results with CNSC REGDOC-2.5.2 acceptance criteria is shown in Table 15.7.9-1.

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Table 15.7.2-1: Results Summary of AOO Events

Description	Exposure	Max. Neutron Flux, % RTP	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % RTP	Δ CPR/ICPR
Decrease in Core Coolant Temperature AOOs						
LFWH AOO (Section 15.5.3.1.1)	MOC	101.1	7.17 (1040.1)	7.33 (1062.6)	100.4	0.0341
Pressure Increase and Inventory Reduction AOOs						
LR-TT AOO (Section 15.5.3.2.1)	BOC	100.0	7.55 (1094.6)	7.70 (1116.3)	100.0	0.0583
1MSRIVC AOO (Section 15.5.3.2.2)	EOR	110.7	7.47 (1083.2)	7.61 (1103.1)	100.3	0.0631
LOCV AOO (Section 15.5.3.2.3)	BOC	100.0	7.55 (1094.6)	7.70 (1116.3)	100.0	0.0583
LOPP AOO (Section 15.5.3.2.4)	BOC	100.0	7.55 (1094.6)	7.70 (1116.2)	100.0	0.0495
FWPT AOO (Section 15.5.3.3.1)	EOR	100.0	7.17 (1039.7)	7.32 (1062.1)	100.0	0.0086
Increase in Reactor Coolant Inventory AOOs						
Inadvertent Isolation Condenser Initiation - One Train (Section 15.5.3.4.1)	MOC	116.8	7.17 (1039.7)	7.32 (1062.1)	100.6	0.0464

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Table 15.7.3-1: Results Summary of the DBA and DEC Events – Non- LOCA

Description	Exposure	Max. Neutron Flux, % RTP	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % RTP	PCT °C (°F)
Decrease in Reactor Coolant Temperature Event DBA						
LFWH DBA (Section 15.5.4.1.1)	MOC	119.1	7.17 (1039.7) Note 1	7.32 (1062.1) Note 1	115.2	308.0 (586.3)
Increase in Reactor Pressure Events DBAs						
LR DBA Note 2 (Section 15.5.4.2.1)	MOC	544.3	8.69 (1259.8)	8.86 (1285.1)	111.8	511.8 (953.3)
LOPP DBA (Section 15.5.4.2.2)	EOR	151.0	8.61 (1249.2)	8.73 (1266.1)	103.5	312.4 (594.3)
RPCD DBA (Section 15.5.4.2.3)	EOR	151.4	8.70 (1262.3) ¹	8.89 (1288.9) ¹	103.5	312.6 (594.7)
MSRIVC-FWIV DBA (Section 15.5.4.2.4)	EOR	158.9	8.61 (1249.4)	8.73 (1266.3)	103.7	312.7 (594.9)
Increase in Reactor Coolant Inventory DBAs						
FWFI – All Pumps (Section 15.5.4.4.1)	MOC	123.3	7.93 (1150.0) (Note 1)	8.09 (1173.6) (Note 1)	115.5	315.5 (599.9)
Inadvertent Isolation Condenser Initiation - All Trains (Section 15.5.4.4.2)	MOC	114.3	7.17 (1039.7)	7.32 (1062.1)	100.0	308.0 (586.3)
Decrease in Reactor Coolant Inventory DBAs						
LOFW DBA (Section 15.5.4.5.1)	MOC	100.0	7.17 (1039.7)	7.32 (1062.1)	100.0	308.0 (586.3)

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Description	Exposure	Max. Neutron Flux, % RTP	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % RTP	PCT °C (°F)
RPCO DBA (Section 15.5.4.5.2)	MOC	100.0	7.17 (1039.7) Note 1	7.32 (1062.1) Note 1	100.0	302.8 (577.0)
Analysis of Design Extension Conditions Without Core Damage						
Pressure Increase DEC's						
1MSRIVC DEC (Section 15.5.5.2)	EOR	208.9	11.19 (1623.2)	11.33 (1643.3)	130.1	727.1 (1340.8)
Complex Sequence LR DEC (Section 15.5.5.3)	EOR	118.4	8.03 (1164.3)	8.17 (1184.5)	100.3	309.2 (588.6)
LOCV DEC (Section 15.5.5.4)	EOR	245.0	9.98 (1447.3)	10.12 (1467.5)	124.5	332.9 (631.2)
LOPP DEC (Section 15.5.5.5)	EOR	250.1	11.14 (1615.6)	11.28 (1635.5)	130.4	328.9 (624.0)
Reactivity and Power Distribution Anomalies - DEC's						
CCF ACRW (Section 15.5.6.1)	MOC	123.8	7.52 (1091.0)	7.68 (1113.3)	115.4	314.0 (597.2)
ICRW (Section 15.5.6.2)	BOC	111.8	7.23 (1048.5)	7.38 (1070.9)	111.8	307.5 (585.5)
Decrease in Reactor Coolant Inventory - DEC						
FW Isolation DEC (Section 15.5.5.4)	MOC	100.0	8.40 (1218.5)	8.50 (1232.1)	100.0	308.0 (586.3)

(1) The simulation of this event is ended before the RPV pressure increase resulted in an ICS train initiation. Assuming a single failure, the second ICS train initiates at 8.605 MPa (1248 psia) and decreases pressure. The pressure remains well below the acceptance criteria for the event. Results are from a bounding case with combined conservatisms.

(2) Results are from a bounding case with combined conservatisms.

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Table 15.7.3-2: Results Summary of Fuel Loading Error Event

Dispersion Factor 0-2 hour (s/m³)	Whole-Body Dose
3.00E-04	0.095-mSv (9.5 mrem)

Table 15.7.3-3: Results Summary of DBA and DEC Events For LOCA

Parameter	Value
Results for Main Steam Pipe Break Inside Containment, Conservative Case (Section 15.5.4.6.1)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	423 kPa (0.322MPaG)
Peak containment shell temperature	134 °C
Results for Feedwater Pipe Break Inside Containment, Conservative Case (Section 15.5.4.6.2)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	407 kPa (0.306 MPaG)
Peak containment shell temperature	134 °C
Results for Small Steam Pipe Break Inside Containment, Conservative Case (Section 15.5.4.6.4)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	191 kPa (0.090 MPaG)
Peak containment shell temperature	125 °C
Results for Small Liquid Pipe Break Inside Containment, Conservative Case (Section 15.5.4.6.4)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	191 kPa (0.090 MPaG)
Peak containment shell temperature	110 °C

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Table 15.7.6-1: Fuel Handling Accident (DEC) Dose Consequences

Dose Location	30-Day Whole-Body Dose ^{Note 1}	Acceptance Criteria
350 m EZ	2.91 mSv (0.291 rem)	20 mSv (whole-body dose for a 30-day period)

(1) Maximum dose for the most critical group.

**Table 15.7.7-1: BWRX-300 DNNP Main Steam Line Break Outside Containment
Accident Dose Consequences**

Coolant Condition	Dose Location	30-Day Whole-Body Dose	Acceptance Criteria
Equilibrium Iodine	350 m Exclusion Zone	8.16E-01 mSv (8.16E-02 rem)	20 mSv (whole-body dose for a 30-day period)
Iodine Spike	350 m Exclusion Zone	2.7 mSv (2.7E-01 rem)	

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**Table 15.7.7-2: BWRX-300 DNNP Feedwater Line Break Outside Containment
Accident Dose Consequences**

Coolant Condition	Dose Location	30-Day Whole-Body Dose	Acceptance Criteria
Equilibrium Iodine	350 m Exclusion Zone	9.29E-01 mSv (9.29E-02 rem)	20 mSv (whole-body dose for a 30-day period)
Iodine Spike	350 m Exclusion Zone	3.03 mSv (3.03E-01 rem)	

**Table 15.7.7-3: BWRX-300 DNNP Isolation Condenser System Line Break Outside
Containment Accident Dose Consequences**

Coolant Condition	Dose Location	30-Day Whole-Body Dose	Acceptance Criteria
Equilibrium Iodine	350 m Exclusion Zone	2.63E-04 mSv (2.63E-05 rem)	20 mSv (whole-body dose for a 30-day period)
Iodine Spike	350 m Exclusion Zone	1.18E-03 mSv (1.18E-04 rem)	

- (1) The dose consequences of an ICS line break at the BWRX-300 Darlington exclusion zone are very low. This is expected because the ICS break flow releases only steam that is assumed to entrain 2 to 3 orders of magnitude less activity than reactor water, and the ICS line break is isolated in a very short 10 second period.

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**Table 15.7.7-4: BWRX-300 DNNP Instrument Line Break Outside Containment
Accident Dose Consequences**

Coolant Condition	Dose Location	30-Day Whole-Body Dose	Acceptance Criteria
Equilibrium Iodine	350 m Exclusion Zone	9.99E-01 mSv (9.99E-02 rem)	20 mSv (whole-body dose for a 30-day period)
Iodine Spike	350 m Exclusion Zone	3.50E+0 mSv (3.50E-01 rem)	

Table 15.7.9-1: Core Damage Frequency Results

PSA Events	Core Damage Frequency (CDF) (/yr) <small>Note 1</small>
At Power	1.1 E-08
Internal Events and Low Power Shutdown (LPSD)	7.0E-10
Internal Fire	1.3E-08
Internal Flood	1.5.E-09
Seismic	5.1E-08
High Wind	4.3E-09 (Straight Wind) 1.3E-10 (Tornado)
Fuel Pool Events	1.3E-08
Fuel and Heavy Load Movements (heavy load drop can cause fuel damage or core damage)	FDF at Power 2.3E-09 FDF at LPSD 1.8E-09 CDF at LPSD 1.6E-09

(1) CNSC REGDOC-2.5.2 CDF limit is 1E-05.

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Table 15.7.9-2: Large Release Frequency Results

Level 2 PSA Scope	Large Release Frequency (LRF) Events per Year ^{Note} 1	Comment
Internal Events At Power	1.8E-09	LRF consists of large early and large late release frequency. SRF is not reported because a small release is conservatively assumed to progress into a large release.
Internal Events Low Power & Shutdown	7.0E-10	LRF is same as CDF because containment is conservatively assumed open during low power and shutdown.
Seismic Events	4.8E-08	With seismic hazard truncation.
Fire Events	1.3E-8	LRF is conservatively assumed the same as CDF. CDF is 1.3E-08/yr.
High Wind	4.3E-09/yr. for straight wind 1.3E-10/yr. for tornado	LRF is conservatively assumed the same as CDF.
Internal Flood	5.5E-10	
Fuel and Heavy Load Movements	5.7E-09	
Fuel Pool Events	1.3E-8	

(1) CNSC REGDOC-2.5.2 LRF is 1E-06.

Appendix 15A. Reference Source Term for Conditions That Are Practically Eliminated

A radiological and combustible gas accident reference source term based on a set of representative core damage conditions is established for use in specifying design attributes and/or complementary design features for DECAs as required by Section 7.3.4 of CNSC REGDOC-2.5.2.

15A.1 Safety Objectives

The safety objectives are consistent with CNSC REGDOC-2.5.2, Section 7.3.4 (Chapter 3, Reference 3.1-1) and IAEA SSR-2/1 (Chapter 3, Reference 3.1-5). The BWRX-300 design fault sequences that could lead to early or large radioactive releases are practically eliminated.

15A.2 Acceptance Criteria

Fault sequences with early or large releases are practically eliminated if either of the following criteria are met:

1. It is physically impossible for the relevant fault sequences to occur
2. There is a high level of confidence that these fault sequences are extremely unlikely

15A.3 Approach

Consistent with Chapter 3, Section 3.1.8, the identification of DECAs is commenced early in the design process, with iterations between design activities, DSA, and PSA with an increase in scope and level of detail as the design progresses.

As part of the overall strategy for addressing challenges, plant conditions, and for restoring and maintaining FSFs, the roles of each complementary design feature and other design attributes are defined. The objective is demonstrating that these complementary design features design attributes are reasonably expected to practically eliminate significant radioactive releases for DECAs.

Each complementary design feature or plant attribute is defined as part of the overall plant strategy for addressing challenges, plant conditions, and for restoring and maintaining the FSFs. The objective is demonstrating that the planned operation of these complementary design features or design attributes is reasonably expected to practically eliminate significant radioactive releases for DECAs:

1. The physical conditions, processes, and phenomena associated with the accident progression.
2. Risk significant severe accident scenarios involving hazards selected for analysis.
3. Bounding fault sequences and DSA considered in the plant design envelope and associated attributes.

The outcome is a set of conditions that are considered in the design of corresponding complementary design safety features or design attributes for DECAs and represent a set of bounding conditions that envelope other accidents.

DECAs selected are justified based on engineering judgement, DSA results and insights from the PSA. In addition, the practical elimination concept reinforces D-in-D by providing a focused evaluation of those conditions having the potential for early radioactive release or a large radioactive release.

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The initial DEC selection of is based on predecessor BWR reactor research and OPEX for the purpose of framing the conceptual design. A more rigorous selection is achieved with iterations between design activities, DSA, and PSA with an increase in scope and level of detail as the design progresses. The reference source terms for the selected conditions are used to develop and confirm the complementary design features or design attributes

If the accident sequence is practically eliminated as a result of a single initiating event such as the failure of the RPV in all operational states, the demonstration relies on achieving a high level of quality at all stages of the component lifetime.

A condition is practically eliminated by considering:

- DL features consisting of equipment and supporting operational organizational provisions (e.g., SAMGs). Certain equipment types may receive increased DL1 design provisions such as providing highly reliable components that are shown not to fail as part of the DSA.
- DL strength – adequate margins, reliability, and qualification against all operating conditions
- DL independence – redundancy physical separation, diversity, and functionality

15A.4 Practically Eliminated Events

For the conceptual design, practical elimination considerations are essential in framing conditions where complementary design features or design attributes are used with the accompanying rationale for selecting DEC's that comprise the reference source term.

The following practically eliminated conditions and provisions achieving practical elimination in Table 15A-1 are based on deterministic and probabilistic methods, operational experience, engineering judgement, and the results of research and analysis.

Table 15A-1: Practically Eliminated Conditions

Practically Eliminated Conditions	Provisions to Achieve Practical Elimination Complimentary Design Features / Design and Supporting Operating Provisions
A sudden mechanical failure of the RPV that eliminates the capability to hold and cool the core.	<p>DL1 provisions for a strong reactor coolant pressure boundary for all states within the plant design envelope without relying on subsequent DLs, including DEC. The following selection of RPV materials and operational uses throughout the plant lifetime provides greater confidence that the likelihood of sudden RPV failure is extremely unlikely:</p> <ul style="list-style-type: none"> • Material fracture toughness • Tensile properties • Defect free fabrication • Defect tolerance • Structural integrity • Operational limits for all operational conditions (normal, AOOs, DBAs) • Loading conditions • Highest reliability and quality according to ASME BPVC
A sudden mechanical failure of the RIVs that results in catastrophic failure creating significant early or late containment challenges.	<p>DL1 requirements for the highest reliability and quality according to ASME BPVC Code Class 1 components provides greater confidence that the likelihood of sudden mechanical failure of the RIVs (part of the RCPB) throughout the plant lifetime is extremely unlikely.</p> <p>These requirements provide assurance that the quality, materials of construction, fabrication, installation, testing, examination, repair/replacement, and ongoing inspection, testing, and monitoring are commensurate with the RIV FSF.</p> <p>DL1 design provisions for nozzle elevations significantly above the core, and passive isolation condenser cooling minimize the likelihood of substantial core damage in the event of a coolant loss from RIV failure.</p>
A severe and unlikely condition where more than 50% of the control rods fail to insert leading to an accident with increasing severity that challenges the reactor and containment system is practically eliminated.	<p>The design offers D-in-D with diverse and redundant shutdown systems making multiple sequential failures extremely unlikely. However, conservatively, the design includes complementary design features that provides a diverse means limiting increases in the reactor vessel pressure in the extremely unlikely condition where more than 50% of the control rods fail to insert.</p> <p>DL4a complementary design feature diverse protection system actuation logic that is diverse and independent from the DL3 SC1 I&C System actuation logic that provides greater confidence that the condition is extremely unlikely.</p> <p>DL4a complementary design feature of the FMCRD run-in function provides greater confidence that the condition is extremely unlikely.</p>

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Practically Eliminated Conditions	Provisions to Achieve Practical Elimination Complimentary Design Features / Design and Supporting Operating Provisions
	<p>DL4a complementary design feature of alternate control rod insertion ensures the hydraulic control unit scram pressure is released to the FMCRDs even in the unlikely event of the solenoid pilot valves failing to scram. This provides greater confidence that the condition is extremely unlikely.</p> <p>DL4b ultimate pressure regulation complementary design feature provides a diverse means to limit increases in reactor vessel pressure for this extremely unlikely condition, thus making failure propagation from the reactor pressure vessel to the containment extremely unlikely.</p> <p>DL4b BIS (see Appendix 15B.2) complementary design feature provides a means to place the plant in a stable controlled configuration if control rods fail to insert.</p>
<p>A single control rod falling out of the core and inserting enough positive reactivity to cause a calculated fuel enthalpy rise greater than the lower bound clad failure limits.</p>	<p>DL1 design provision that the control blades include a bayonet style coupling design provision that requires a 45-degree rotation to uncouple, thus making it physically impossible for the control rod blade to become uncoupled from the drive during reactor operation.</p> <p>DL2 dual separation detection devices that sense if the hollow piston is no longer on the ball nut block control rod withdrawal and do not allow a separation distance to occur between the FMCRD and the control rod, or from the ball nut to an unlatched hollow piston. This essentially limits possible separation such that it is not physically possible for a control rod drop accident involving a single control rod falling completely out of the core to occur.</p> <p>Chapter 4, Section 4.5 discusses these design provisions for the FMCRD and the control rods.</p>

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Practically Eliminated Conditions	Provisions to Achieve Practical Elimination Complimentary Design Features / Design and Supporting Operating Provisions
<p>Containment heating following a postulated RPV failure does not occur.</p> <p>This chain of events is called a high pressure melt ejection and these four mechanisms may cause a rapid increase in containment pressure and temperature:</p> <ol style="list-style-type: none"> 1. Blowdown of the reactor pressure vessel 2. Efficient debris-to-gas heat transfer 3. Exothermic metal/steam and metal/oxygen reactions 4. Hydrogen combustion <p>These mechanisms lead to increased loads on the containment building that are collectively referred to as direct containment heating.</p>	<p>DL1 design reduces the likelihood of being at a high pressure at the time of vessel failure is eliminated by the large inventory of coolant in the RPV. Core cooling challenges are only possible when corresponding RPV depressurization occurs. With adequate control rod insertion, RPV depressurization is eliminated by the DL3 passive ICS and the DL3 RIV isolation or diverse DL4a RIV isolation.</p> <p>DL4a ICS actuation provides greater confidence that RPV failure occurring at high pressure is extremely unlikely.</p> <p>DL4b RPV ultimate pressure regulation complementary design feature provides a diverse means to limit RPV pressure increases ensuring that there are no credible core damage accident sequences without cooling when the reactor is pressurized, making direct containment heating following a postulated vessel failure highly unlikely with a high degree of confidence.</p> <p>DL4b complementary design feature is the containment ultimate overpressure vent that ensures containment does not overpressurize when RPV ultimate pressure regulation is actuated.</p>
<p>Containment challenges from large steam explosions do not occur.</p> <p>Molten core debris interacting with water can produce explosive interactions as well as non-explosive, dynamic interactions from mutual fragmentation and mixing of the core material and water.</p>	<p>RPV pressure is sufficiently high to suppress explosive conditions where anticipated thermal interactions are dynamic but not explosive.</p> <p>DL1 design provides a strong containment structure and components and sufficiently large containment volume making containment load challenges from large steam explosions physically impossible.</p> <p>DL1 design and operating physical limitations (e.g., no subcooled water pool) associated with explosive interactions preventing containment challenges.</p>
<p>Containment integrity challenges from combustible gas detonation does not occur.</p>	<p>DL1 design provision for an inert containment atmosphere and combustible gas management operating provisions limit concentrations well below detonation levels. This makes containment challenges due to combustible gas detonation physically impossible.</p> <p>DL4b combustible gas mitigation devices (e.g., passive mixing and venting, passive autocatalytic recombiners) are strategically located above the containment head in the vicinity of the pools. This provides greater confidence that any combustible gas leakage from containment does not result in damage to SSCs required for safety.</p>

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Practically Eliminated Conditions	Provisions to Achieve Practical Elimination Complimentary Design Features / Design and Supporting Operating Provisions
Containment failure because of molten core concrete interaction.	If the RPV fails, a DL1 design provision for a corium shield/liner confines and prevents contact between the molten core and the containment liner. The corium shield/liner spreading area is cooled by the over-lying water. Steam is released through the hardened vent preventing containment failure.
Containment failure by quasi-static overpressurization from a long-term loss of containment heat removal (SA with unsuccessful containment isolation and passive containment heat removal).	DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static overpressurization from a long-term loss of containment heat removal physically impossible.
An interfacing system LOCA outside containment.	DL1 design provisions ensures subsystems connected to the RCPB are designed to an ultimate rupture strength at least equal to the reactor coolant system design pressure make an interfacing system LOCA physically impossible.
Containment bypass consequential to SA progression. SA fault sequences leading to or involving RIV leakage and core damage.	DL1 RIV design includes ball valves with hydraulic actuators. Design, qualification, and valve procurement ensures the important FSF of RPV isolation occurs. DL1 design and operating provisions for a strong RCPB for all states within the plant design envelope are the highest reliability and quality complying with ASME BPVC. The mechanical failure likelihood of the RIV internals throughout the plant lifetime due to SA conditions is extremely unlikely.
A SA with an open containment (plant in shutdown state). SA events during shutdown involve loss of fuel pool cooling sequences or LOCAs leading to drain-down.	DL1 design provisions of a large inventory of available water and operating provisions with slow developing fault sequences provide ample time for several diverse means of providing makeup water and preventing fuel being uncovered in the fuel pool or the reactor core. These design features provide greater confidence that the likelihood of SA with an open containment is extremely unlikely.
Fuel coolant interaction (in-vessel steam explosion).	DL1 design provisions of a passive ICS (ECCS system) with a large water inventory with inherent margins eliminates system challenges leading to loss of adequate core cooling. This provides confidence that the likelihood of an in-vessel steam explosion from loss of core cooling is extremely unlikely.

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Practically Eliminated Conditions	Provisions to Achieve Practical Elimination Complimentary Design Features / Design and Supporting Operating Provisions
A high pressure melt ejection with potential for debris dispersion such that debris is not retained in lower containment.	<p>DL4b ultimate pressure regulation complementary design feature provides a diverse means to limit the increase in reactor vessel pressure ensuring that there are no credible core damaging accident sequences without cooling while the reactor is also at high pressure, making containment failure from high pressure melt ejection physically impossible</p> <p>DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static overpressurization from a high pressure melt ejection physically impossible.</p> <p>DL1 design provision for a corium shield/liner supported by operating provisions confine and prevent the spread of a molten core, making containment failure from molten core concrete interaction physically impossible.</p>
An out of core criticality event during fuel handling.	<p>DL1 design provisions and supporting operating provisions provide design and administrative controls that effectively manage factors that influence system reactivity and the likelihood of criticality. These factors are enrichment, mass, moderation, geometry, reflection, interaction, and spacing.</p> <p>Plant procedures prohibit the handling and storage of more than one fuel assembly. The amount of water moderator available from internal flooding is limited by the design of the storage location structures thus making an out of core criticality physically impossible. Refer to Section 15.5.9.3 for the BWRX-300 Out of Core Criticality Analysis.</p>

Appendix 15B. Complementary Design Features for Mitigating Design Extension Conditions

Consistent with CNSC REGDOC-2.5.2, Section 6.1, "Application of a D-in-D," and Section 6.3 "Accident prevention and plant safety characteristics," complementary design features are used to prevent accident progression or mitigate the consequences of DEC. In accordance with the BWRX-300 Safety Strategy, complementary design features are identified that address plant challenges, conditions that require restoration, and maintaining the FSFs. These complementary design features can be reasonably expected to practically eliminate or mitigate significant radioactive releases for DEC. Practical elimination of DEC is discussed in the previous Appendix 15A. There are several complementary design features provided ensuring that DEC are either practically eliminated or extremely unlikely to occur. Complementary design features mitigating functions are provided in Table 15B-2.

Most complementary design features are component features such as vents and catalytic recombiners. However, the boron injection system described in the following section is a complete system used in inserting negative reactivity in the core in the highly improbable event where shutdown using CRDs via hydraulic or motor run-in cannot be accomplished.

15B.1 Boron Injection System

The Boron Injection System (BIS) provides a separate, diverse means, D-in-D backup system to for manually inserting negative reactivity into the reactor core. The BIS assures reactor shutdown by mixing a neutron absorber with the primary coolant.

The BIS is only required in the highly improbable event when shutdown using CRDs via hydraulic or motor run-in cannot be accomplished. The system introduces sufficient negative reactivity into the reactor primary system assuring reactor shutdown to a subcritical state with no control rod motion.

Injection of a neutron absorber solution for reactivity control is a DL4b, Safety Category 3 function.

The system achieves cold shutdown conditions with a predetermined mass of water in the reactor vessel using an enriched Boron-10 solution injected into the reactor vessel. The CRD system and the BIS do not share any components.

15B.1.1 Safety Design Bases

The BIS provides a means of achieving cold subcriticality by mixing a neutron absorber with the primary coolant. This condition creates a design requirement for reactivity control from full power to cold 20°C (68°F) subcritical state using an enriched Boron-10 solution.

The ICS for overpressure control and CRD/HCU/FMCRD for reactivity control address failure-to-scrum impacts or other reactivity events where boron injection provides an additional layer of defence.

The neutron absorber injection is manually initiated by the operator from either the MCR or SCR.

The minimum amount of neutron absorber required to shut down the reactor is calculated based upon the minimum natural boron concentration required for shutdown and the weight of the water in the RPV at normal level including the shutdown cooling loops at cold shutdown conditions. This ensures that total quantity of stored neutron absorber is sufficient to achieve the minimum sodium pentaborate concentration required for shutdown.

The BIS is a complimentary design feature that provides D-in-D for long-term reactivity control DEC.

The BIS system is designated as Safety Class 3.

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The BIS can be operated in the event of a LOPP but does not perform any non-safety or safety class functions during off-normal conditions.

The BIS injection line penetrates containment and is directly connected to the reactor vessel through the ICS. This portion of the BIS through the ICS system that requires containment isolation is part of the RCPB and is classified as DL3, Quality Group A, Safety Category 1, and Seismic Category 1-A and 1-B. The ICS equipment is Safety Class 1.

15B.1.2 System and Equipment Functions

The BIS simplified flow diagram of the boron injection mode is shown on Figure 15B-2. The BIS consists of a storage tank, test tank, injection pump, piping, valves, and instrumentation and controls necessary to prepare and inject the neutron absorbing solution into the reactor and for system testing. The air-operated injection valve and the Class 1 injection piping are part of the BIS. The BIS consists of a single, 100% equipment train located in the RB outside containment, tying into an ICS return line between the redundant actuation valves and the RIVs.

The neutron absorber is pumped from the storage tank through the injection valves into the reactor vessel by remote manual operation of the pump, storage tank outlet valve, and injection valve.

The BIS injects the neutron absorber solution into the reactor core compensating for various reactivity effects that could occur during plant operation.

The neutron absorber is an aqueous solution of decahydrate ($\text{Na}_2\text{B}_{10}\text{O}_{16} \cdot 10\text{H}_2\text{O}$). Enriched sodium pentaborate solution is made by mixing granular enriched sodium pentaborate with water. The neutron absorber design concentration has a 4.4°C solution precipitation temperature. This low precipitation temperature eliminates the need for storage tank heating during normal standby conditions or heat tracing the instrument lines and pump suction piping.

The volume versus concentration limits is calculated accounting for normal reactor vessel water volume and water volume in the shutdown cooling piping. This neutron absorber solution quantity is the amount above the pump suction shutoff level in the storage tank. No credit is taken for the portion of the storage tank volume that cannot be injected.

15B.1.3 Component Description

The BIS is a single 100% capacity system that includes:

- A single triplex injection pump capable of 2.27 m³/hr (10 gpm) pumping against the maximum reactor pressure of 12.41 MPaG (1800 psig)
- A test tank and associated piping and valves
- An injection line with containment isolation valves
- A demineralized water connection
- A service air connection
- System piping drains and collection subsystem

All equipment is located outside primary containment allowing access for testing and inspection activities during all plant operating conditions.

Operation in the injection mode requires BIS pump start and opening the storage tank outlet valve and injection valves. The storage tank outlet valve and injection valve are air-operated. The air-operated valves open on a system initiation signal, and close [fail as-is] on loss or removal of the pressurized air source or close on a manual valve closure signal.

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The injection location is the condensate return line of the ICS "C" loop downstream of the two ICS condensate return valves, providing a direct flow path into the reactor. The injection location requires a separate containment penetration for the BIS injection line. Overpressure protection of the BIS is provided by a relief valve located on the pump discharge piping. To avoid water hammer at the onset of BIS injection, the piping is fully-filled by a connection with the demineralized water system.

The testing subsystem consists of a test tank containing demineralized water and the associated valves and piping in parallel with the normal suction flow path from the storage tank. During the non-testing phase of system normal operation, the testing network is isolated from the balance of the system by shutoff valves. Periodic operational tests of the BIS are conducted at any time regardless of the state of reactor operation. The testing subsystem demonstrates:

- Pumping demineralized water at rated pressure and flow to and from the test tank
- Pumping demineralized water from the test tank into the reactor vessel against existing reactor pressure (this test may only be performed during reactor shutdown) to ensure a functioning flow path

A tank zero level shutoff prevents damage to pump from insufficient suction. The storage tank, test tank, pump and interconnecting piping and valves are in the Reactor Building on the -8.5 m Level in Room 1551 that is nearest to the ICS connection line. There is a single containment penetration for the BIS injection line. The injection line connects with the ICS "C" return line downstream of the two condensate return valves providing a flow path into the reactor vessel.

15B.1.4 Materials

System piping meets ASME BPVC Section III, Subsection NC from the storage tank outlet connection up to the air-operated injection valve. The system piping from and including the air-operated injection valve (AOV) to the connection with the ICS is ASME B&PVC Section III, Subsection NB. Containment isolation is provided by a single check valve inside containment for immediate isolation and the check valve and AOV for long-term containment isolation. All system piping is either 304 or 316 stainless steel.

All equipment not required for injection, such as system drains, test equipment, and service air and demineralized water are classified as Quality Group D and designed to ASME B31.1 standard and are 304 or 316 stainless steel.

The BIS meets Seismic Class 1-A for the pump suction piping, pump discharge piping and injection piping up to the connection with the ICS. The AOVs and pump are Seismic Category 1-B. The test tank and test loop piping, demineralized water piping, service air piping, and system drains after the isolation valves are non-seismic.

The storage tank meets ASME B&PVC Section III, Subsection NC standards with 304 or 316 stainless steel and contains a hatch, heater system, sparger, and connections for an outlet, overflow, vent, service air and demineralized water inlet, and associated instrumentation.

The injection pump meets ASME B&PVC Section III, Subsection NC standards and is either 304 or 316 stainless steel for all wetted parts.

All equipment in the BIS in contact with neutron absorbing solution is stainless steel for corrosion protection.

15B.1.5 Interfaces with Other Equipment and Systems

System interfaces for the boron injection system are shown in Table 15B-2 and Figure 15B-1.

15B.1.6 System and Equipment Operation

The BIS has four operating modes:

- Standby
- System Injection
- Pump Test
- System Injection Test Mode

BIS functions are required during power operation or reactor startup.

15B.1.7 Instrumentation and Control

The BIS has the following instrument and control features:

1. High and low storage tank solution temperature annunciates in the control room at temperatures outside the allowed range
2. High and low storage tank solution level is annunciated in the control room when the level is outside its normal allowed limit
3. Low storage tank level automatically shuts off the injection pump when the solution level in the storage tank is below the zero level

Either the storage tank or test tank discharge valve must be fully open for the pump to run.

The following describes key instrumentation features of the BIS:

1. Storage tank solution is measured by a level transmitter. MCR level indication and high/low level alarms provided in the MCR HSI display.
2. Pump discharge pressure is sensed by a single pressure transmitter that provides MCR pressure indication through the HSI display
3. Storage tank solution temperature is sensed by a temperature element providing MCR storage tank solution temperature through the HSI display
4. Pump flow is sensed by a single flow element and flow transmitter with flow indication in the MCR
5. Pump status indication (energized status) provided in the MCR HSI display
6. System status indication and annunciators through the MCR HSI display for system bypass or inoperable condition
7. Valve position indication through the MCR HSI display for the storage tank outlet valve and test tank outlet valve
8. Pump overload trip or power loss indication provided through the MCR HSI display

A single pressure gauge is locally mounted for pressure indication during the pump test mode and is visible from the test loop control valve aiding control valve positioning.

Main Control Room Operation

Manual initiation, operation, and control of the BIS is from the MCR through the HSI display. The means for manual actuation and for monitoring shutdown status from the MCR meets CNSC REGDOC-2.5.2, Section 8.4.3, addressing monitoring and operator action of shutdown systems.

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BIS actuation is dual-action initiation through the HSI display in the MCR that guards against accidental or inadvertent actuation of the system ensuring intentional system actuation.

Controls for the BIS are located through the HSI display in the MCR facilitating system operation. System and component operating status, including any system bypasses, system out of service, or manual overrides is provided. Manual system initiation and shutdown is provided.

The following BIS displays, alarms and controls are provided in the MCR HSI display:

System Annunciators

- High and low alarms for storage tank solution temperature
- High and low alarms for storage tank solution level
- Manual or automatic system out-of-service condition

System Indication

- Storage tank solution level
- Pump discharge pressure
- Pump operation (on/off)
- Storage tank and test tank outlet valves position (open/closed)
- Storage tank solution temperature
- System manually out of service
- Pump overload trip or power loss
- Pump flow rate

Local Panel Operation

A local panel is provided for the following:

Controls:

- Heater controls

Indications:

- Status indications for the mixing heater
- Storage tank solution temperature
- Storage tank solution level

Secondary Control Room

The BIS is initiated from the SCR if the MCR is not operational.

15B.1.8 Inspection, Testing and Maintenance

The BIS is functionally tested to ensure the pump develops rated flow and discharge pressure and the flow path from the pump suction to the RPV without contaminating the reactor with neutron absorber solution during each planned outage or reactor shutdown for refueling. Functional testing is performed by circulating demineralized water from the test tank, return to the test tank for the flow and pressure test, and with the injection valve open for the RPV injection test.

Maintenance Provisions

The BIS design is provided with adequate equipment removal paths and personnel access for repair and replacement. The following features are provided to facilitate component maintenance:

1. The relief valve is provided with flanged inlet and outlet connections facilitating removal for bench testing
2. Isolation valves are provided on the pump suction and discharge side for maintenance
3. The storage tank heater can be removed without draining the tank
4. Sufficient pull space is provided for removing/replacing the storage tank heater
5. The storage tank is fitted with a top-mounted entry hatch and an external ladder for equipment installation, maintenance, and chemical addition
6. In the event of a sodium pentaborate system injection or the sodium pentaborate leakage into the system piping, the demineralized water supply can be used to flush the piping downstream the storage tank outlet valve
7. Sufficient pump room is provided for removing plungers, pistons, rods, crankshaft, and inspecting parts
8. Ample head room is provided above the pump for a crane, hoist, or tackle
9. Sufficient space is provided around the storage tank for inspection and maintenance
10. Sufficient space is provided for access and local instrument calibration
11. System maintenance is performed by closing the storage tank outlet isolation valve
12. A maintenance valve downstream of the inboard containment isolation valve can be closed allowing system maintenance and isolation valve testing
13. Test vents and drains for testing CIVs

Dedicated piping and collection for sodium pentaborate solution is provided where drainage occurs. The collection vessel is a portable stainless steel drum. A low containment surrounds the system preventing the spread of sodium pentaborate leaks. No special temporary or permanent provisions such as ladders, scaffolding, overhead cranes, etc., are required for maintenance.

Surveillance Testing and In-Service Inspection Provisions

All active BIS components can be tested during normal plant operation. Pre-operational tests are conducted demonstrating the system flow path, adequate pump net positive suction head (NPSH), and pressure drops in accordance with system design documents.

Pre-operational and periodic hydrostatic testing of the system per OLCs are performed complying with ASME B&PVC Section III. Periodic testing of the system and components is scheduled at a frequency commensurate with the OLCs and meets the requirements of CNSC REGDOC-2.5.2, Section 8.4.2.

15B.1.9 Radiological Aspects

There are no radiological releases from a failure in the boron injection system. Breaks outside containment for the injection flow path through the injection location in the condensate return line of the ICS "C" loop downstream of the two ICS condensate return valves through a separate containment penetration is evaluated in Chapter 15, Subsection 15.5.9.2.3.

15B.1.10 Performance and Safety Evaluation

The BIS injects a neutron absorber quantity into the reactor vessel producing a minimum concentration to achieve a cold shutdown accounting for leakage and imperfect mixing.

The required shutdown concentration is achieved in a mass of water equal to the sum of the mass of water in the reactor vessel plus the mass of water in the SDC system.

The boron injection rate is based on a boron concentration rate of change between 8 to 20 ppm/minute in the reactor water that includes the weight of water in the reactor and shutdown cooling loops at normal level and at 20°C (68°F). The BIS pump can inject the sodium pentaborate solution at all reactor pressures from 12.41 MPaG (1800 psig) (vessel bottom) to zero MPaG.

The BIS can be actuated and operated in the event of LOOP. BIS contains safety features such as containment isolation and reactor coolant pressure boundary piping (Class 1 piping) for the injection line. The BIS injection line is part of the ICS described in Chapter 6, Subsection 6.2.1.

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Table 15B-1: Complementary Design Features Mitigating Functions

DEC Function	Complementary Design Feature
Reactivity Control	<p>DL4a complementary design feature diverse protection system actuation logic that is diverse and independent from the SC1 actuation logic.</p> <p>DL4a complementary design feature of the FMCRD run-in function.</p> <p>DL4a complementary design feature of alternate control rod insertion ensures the hydraulic control unit scram pressure is released to the FMCRDs even in the unlikely event of the solenoid pilot valves failing to scram.</p> <p>DL4b ultimate pressure regulation complementary design feature provides a diverse means to limit increases in reactor vessel pressure for this extremely unlikely condition, thus making failure propagation from the reactor pressure vessel to the containment extremely unlikely.</p> <p>DL4b BIS complementary design feature (see Section 15B.1) provides a means to place the plant in a stable controlled configuration if control rods fail to insert.</p> <p>Ex-vessel sequences do not result in a critical configuration.</p>
RPV Depressurization	See reactivity control complementary design feature for RPV ultimate pressure regulation.
In-Vessel Core Cooling	ICS includes DL4a actuations ensuring core cooling availability.
Cooling of Corium Debris in the Containment Corium Shield/Liner	<p>The corium/shield liner complementary design feature prevents core melt debris from interacting with the containment liner preventing core-concrete interaction.</p> <p>Debris bed cooling is accomplished by flooding the corium/shield liner with a pool of water following RPV failure. Heat transfer from the corium debris bed to the containment water pool prevents containment temperature rising that occurs due to heat radiated from an uncovered debris bed. In this manner, potential high temperature challenges to mitigating equipment and containment pressure boundary components such as penetrations seals are limited.</p>
Cooling High Pressure Melt Ejection Debris	<p>DL4b ultimate pressure regulation complementary design feature provides a diverse means to limit the increase in reactor vessel pressure ensuring that there are no credible core damaging accident sequences without cooling while the reactor is also at high pressure, making containment failure from high pressure melt ejection physically impossible</p> <p>DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static overpressurization from a high pressure melt ejection physically impossible.</p> <p>DL1 design provision for a corium shield/liner supported by operating provisions confine and prevent the spread of a molten core, making containment failure from molten core concrete interaction physically impossible.</p>

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DEC Function	Complementary Design Feature
Containment Isolation	<p>DL3 containment isolation is maintained providing a barrier to radionuclide releases to the environment.</p> <p>Diverse independent DL4a isolation functions are provided in the event of a CCF` of DL3 isolation functions. The CIV function includes the containment structure, penetration, valves, and isolation barriers.</p>
Containment Pressure Control:	<p>Containment pressure control via heat removal is required to prevent containment over-pressurization due to steam generation and/or exothermic chemical reactions (non-condensable gas generation).</p> <p>Passive containment cooling is normally in service including an SA event.</p> <p>A DL4b containment ultimate overpressure vent is included to relieve pressure to the reactor cavity and equipment pool to prevent potential containment failure.</p>
Combustible Gas Control	<p>Combustible gas control is present at the start of a potential accident sequence due to the inert containment atmosphere. Hydrogen generation could develop if a SA occurs. Oxygen during a SA is insufficient to create a combustible containment atmosphere.</p> <p>The ICS includes a non-condensable gas removal design feature (autocatalytic recombiner discussed in Section 6.2) that ensures combustible gas concentrations do not develop.</p>
Post-Accident Monitoring	<p>Post-accident monitoring provides information to facilitate post-accident response and evaluation of RPV and containment conditions.</p>

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Table 15B-2: Boron Injection System Interfaces

Interfacing System	Interface Description	Interface Boundary
Water, Gas, and Chemical Pads	Provide demineralized water for initial storage tank fill and chemical mixing, system flushing, test tank fill for system testing, and maintaining the system full during standby.	The interface is at the upstream side of BIS pressure control valve.
Plant Pneumatics System	Provide service air for periodic storage tank solution mixing through the air sparger in the tank.	The interface is at the upstream side of BIS pressure control valve.
Safety Class 2 and 3 Electrical Distribution System	Electrical power for the BIS pump.	The interface boundary is at BIS equipment terminals.
Non-Safety Electrical Distribution System	Electrical power for the storage tank electric heater.	The interface boundary is at BIS equipment terminals.
Electrical Power	Provide electrical power for operation of the storage tank heater, and injection pump.	The interface boundary is at BIS equipment terminals.
Reactor Building Structure	Provide structural support and protection for BIS equipment.	The interface is at the equipment foundation and support structures for the BIS equipment.
Safety Class 2 and Safety Class 3 Instrumentation and Control	Provide for input and output for BIS instrumentation and controlled components.	The interface is at the DCIS.
Isolation Condenser System	Provide an injection path to the reactor for the neutron absorber solution and system isolation from the reactor system.	At the ICS "C" loop return piping.

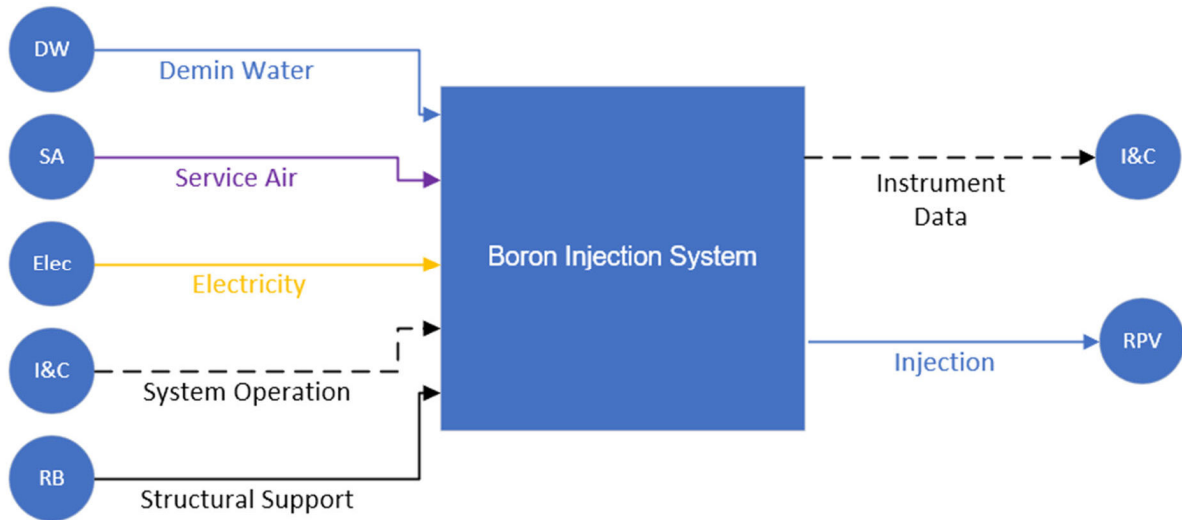


Figure 15B-1: BWRX-300 Sodium Pentaborate Injection System Interfaces

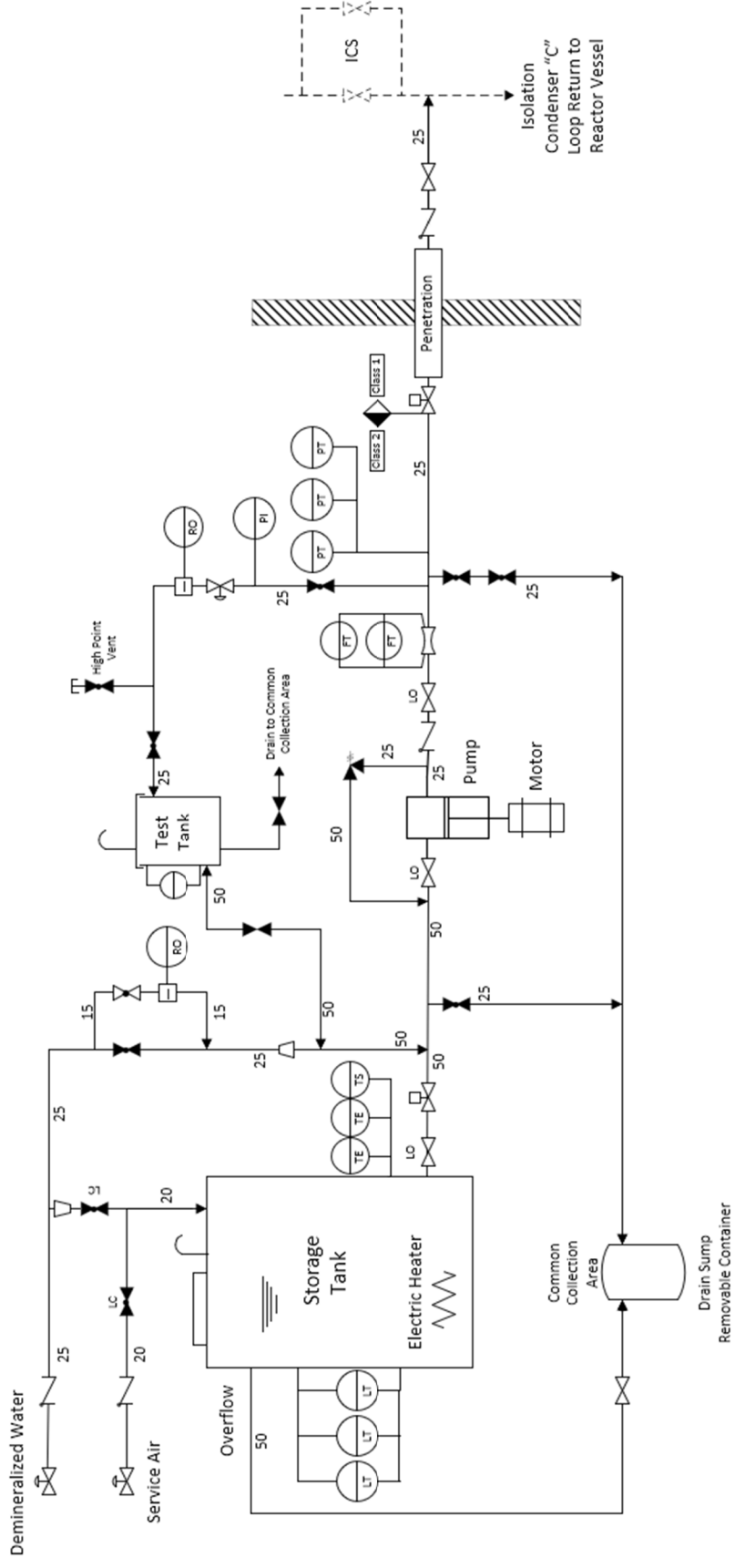


Figure 15B-2: Boron Injection System Simplified Flow Diagram

Table 15.5-1: Key Initial Conservative LOCA Evaluations

Parameter	Value	Notes
Thermal Power	887.4 MW	102% of rated power. Hot shutdown initial power is also included in steam pipe break cases
Dome pressure	7308.4 kPa	Upper end of normal operating range
Initial feedwater temperature	241.9 °C for MSL Break	Initial temperature of 191.9 °C is also included in FW pipe break cases
Initial temperature of the ICS and reactor cavity pools	43.3 °C	
Initial containment pressure	119.7 kPa	Upper end of normal operating range
Initial containment temperature	43.3 °C	Lower end of normal operating range
Initial water level in downcomer	21.1 m (large breaks)	Lower initial level (0.152 m below normal level) is also evaluated for small breaks

Table 15.5-2: Summary of Core-Wide Decay Ratio Results

Cycle Exposure (GWd/ST)	Core-Wide Decay Ratio	
	Nominal (241.9 °C)	LFWH to 191.9 °C with SCRR mitigation (AOO Conditions)
0 (BOC)	0.56	0.53
4 (MOC)	0.71	0.57
6 (EOR)	0.64	0.56

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Table 15.5-3: Input Parameters and Initial Conditions and Assumptions Used in Non-LOCA Analyses

Parameter	Value
Heat Balance Related Parameters	
Rated thermal power level, MWt	870
Core flow, kg/s (Mlbm/hr) – analysis value	Calculated
Reference rated core flow, kg/s (Mlbm/hr)	1890.0 (15.0)
Steam flow, kg/s (Mlbm/hr) – analysis value	Calculated
Reference rated steam flow, kg/s (Mlbm/hr)	503.2 (3.993)
Feedwater (FW) flow, kg/s (Mlbm/hr) – analysis value	Calculated
Reference rated FW flow, kg/s (Mlbm/hr)	507.0 (4.024)
Nominal dome pressure, MPa (psia)	7.17 (1040)
Nominal FW temperature, °C (°F)	241.8 (467.3)
Normal Reactor Water Level (NWL, L5), m (in) Above Vessel Zero (AVZ)	21.097 (830.591)
Control Rod Drive / High Pressure Injection System Related Parameters	
Control rod position versus time	Table 15.5-4
Control Rod Drive Mechanism (CRDM) run-in fast speed (minimum requirement), mm/s (in/s)	70.0 (2.76)
CRDM withdrawal (maximum speed), mm/s (in/s)	28.0 (1.10)
Main Steam Line (MSL) Related Parameters	
Number of MSLs	2
Minimum MSL length (average of all lines): flow path from vessel to Turbine Stop Valve (TSV), m (ft)	59.0 (194)
Minimum MSL volume (total of all lines including header): vessel to TSV, cubic m (cubic ft)	25.07 (885.3)
Minimum MSL pressure difference between the vessel dome pressure and the turbine throttle pressure at rated conditions, kPa (psi)	207 (30.0)
MSRIV minimum closure time, s	3.0
MSRIV closure profile for minimum closure time, s	
100% open area	0.0
100% open area	0.6
1% open area	1.7
0% open area	3.0
Feedwater Related Parameters	
Number of FW pumps	2
Number of motor-driven FW pumps	2
One FW pump is operating, and one FW pump is in standby	

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Parameter	Value
Maximum flow demanded is between, % Rated	105 - 115
Maximum flow of both FW pumps assuming a CCF, % Rated	220
FW temperature reduction for Loss of Feedwater Heating (LFWH) AOO, °C (°F)	50 (90)
Minimum time constant of FW temperature response, s This value is assumed based on operating experience and allows simplified modeling of the FW temperature response	60
FW pump trip coast down time constant	3
Isolation Condenser System (ICS) Related Parameters	
ICS heat removal capacity per train, MW	33.75
ICS maximum initial temperature, °C (°F)	43.3 (110)
ICS nominal initial temperature, °C (°F) (used in most transient events as the value is not key)	20.85 (69.53)
ICS minimum initial temperature, °C (°F)	10 (50)
ICS condensate return valve maximum opening time, s	10
ICS train condensate return line minimum internal diameter, mm (in)	178 (7.00)
The elevation difference from centerline of the horizontal connections between the ICS steam line distribution header to the centerline of the ICS return injection to the chimney, minimum, m (in)	15.808 (622.362)
Balance of Plant Related Parameters	
TSV minimum fast closure steam flow shutoff rate from rated power, % rated steam flow / s	667
TCV minimum fast closure steam flow shutoff rate from rated power, % rated steam flow / s	667
TCV minimum slow (servo) closure steam flow shutoff from rated power % rated steam flow / s	40
TBV capacity at rated conditions % rated steam flow	25
Miscellaneous Parameters	
Biased initial CPR, (reduction versus nominal initial CPR) Note: the value is approximate because it is slightly different depending on the initial condition. This is used in CN-PA events and some DEC events (as noted in event description)	~0.1
Biased initial hot fuel rod power, (multiplier relative to average rod power). Note: This is used in CN-PA events and some DEC events (as noted in event description). This input is only important if there is boiling transition and a subsequent fuel cladding temperature increase during the event.	1.6

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Table 15.5-4: CRD Scram Time

Reactor Vessel Bottom Gauge Pressure MPaG (psig)	Rod Insertion Position	Required Maximum Time (s) Note 1
≤ 8.75 MPaG (1269 psig)	10%	≤ 0.46
	40%	≤ 1.20
	60%	≤ 1.71
	100%	≤ 3.70
≤ 9.48 MPaG (1375 psig) Note 2	10%	≤ 0.56
	40%	≤ 1.40
	60%	≤ 2.03
	100%	≤ 4.20

- (1) The times include 0.2 s delay from de-energizing the scram pilot valves to control rods movement.
- (2) These times are used in pressurization increase events in the DBA category even if the pressure is not greater than 8.75 MPaG (1269 psig) during the rod insertion.

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Table 15.5-5: Defense Lines Inputs Used in Non-LOCA Analyses

DL ID	Function Name	Inputs	Setpoint / Delay Analytical Limits
DL2-01	Maintain Target Pressure (Performed by RPC)	Steam flow demand is a function of reactor dome pressure	Target dome pressure is normal dome pressure in Table 15.5-3 Design description of RPC is provided in Chapter 7, Section 7.3.3.2, Item 5
DL2-02	Maintain Target Level (Performed by RLC)	RPV level error (function of core power, reactor water level, steam flow, steam flow enthalpy, FW enthalpy)	Target reactor level is NWL (L5) in Table 15.5-3 Design description of RLC is provided in Chapter 7, Section 7.3.3.2, Item 3
DL2-04	Control Rod Block on ATLM	Determination of the thermal limits or soft duty guidelines violation	Design description of ATLM is provided in Chapter 7, Section 7.3.3.2, Item 21
DL2-05	Control Rod Block on MRBM	Indication of potential fuel damage thermal limits being exceeded	Design description of MRBM is provided in Chapter 7, Section 7.3.3.2, Item 22
DL2-08	Anticipatory Hydraulic Scram on Turbine Trip or Generator Load Rejection Demand	Turbine Trip or Generator Load Rejection Demand	Total time delay = 0.05 s after generator load rejection or turbine trip signal
DL2-09	TBV Fast Open on Turbine Trip or Generator Load Rejection Demand	Turbine Trip or Generator Load Rejection Demand	Before or at the same time as TCV or TSV closure
DL2-13	Turbine Trip on High Main Condenser Pressure Setpoint 2	High Main Condenser Pressure Setpoint 2	No analytical limit
DL2-14	TBV Closure on High Main Condenser Pressure Setpoint 3	High Main Condenser Pressure Setpoint 3	Set to allow for time for bypass to be open after turbine trip signal for expected rates of condenser pressure increase. Total time delay = 1 s
DL2-19	FW and Condensate Pump Trip on High RPV Level	High RPV Water Level (L8)	L8: 22.277 m (877.047 in.) AVZ Total time delay = 1.0 s
DL2-21	Anticipatory Hydraulic Scram on MSRIV/MSCIV Position	MSRIV/MSCIV Position	<90% open Total time delay (scram) = 0.05 s

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DL ID	Function Name	Inputs	Setpoint / Delay Analytical Limits
DL2-25	Start Standby FW Pump on Loss of Operating FW Pump	Loss of Operating FW Pump	Analysis allows 10 seconds identify and confirm operating pump is lost and to initiate standby pump. Analysis allows 15 seconds to ramp up the standby pump to full flow
DL2-27	Select Control Rod Run-In on FW Temperature Decrease	FW Temperature Decrease	FW temperature reduction of 16.6°C (30°F) or more. Total time delay = 5 s
DL2-31	ICS Pressure Control on High Reactor Pressure	High Reactor Pressure	Settings are not important for Transient DSA results Design description is provided in Chapter 7, Section 7.3.3.2, Item 6
DL2-39	Anticipatory Hydraulic Scram on High RPV Level	High RPV Level (L8)	See DL2-19
DL3-01	Hydraulic Scram on High RPV Pressure	High RPV Pressure (HP1)	7.787 MPaG (1129.4 psig) Total time delay = 0.7 s
DL3-02	Hydraulic Scram on Low RPV Pressure	Low RPV Pressure (LP1)	5.516 MPaG (800.0 psig) Total time delay = 0.7 s The transient analysis used a lower value this value is required consistent with LOCA analyses.
DL3-03	Hydraulic Scram on Low RPV Level	Low RPV Level (L3)	L3: 19.645 m (773.425 in) AVZ Total time delay = 1.0 s
DL3-04	Hydraulic Scram on High Neutron Flux	High Neutron Flux	125% of rated Total time delay = 0.09 s
DL3-05	Hydraulic Scram on High Simulated Thermal Power	High Simulated Thermal Power	115% of rated Signal Time Constant = 7 s Total time delay = 0.09 s
DL3-11	ICS Train 1 Initiation on High RPV Pressure	High RPV Pressure (HP2)	8.289 MPaG (1202.2 psig) Total time delay = 0.7 s
DL3-12	ICS Train 2 Initiation on High RPV Pressure	High RPV Pressure (HP3)	8.504 MPaG (1233.4 psig) Total time delay = 0.7 s
DL3-13	ICS Train 3 Initiation on High RPV Pressure	High RPV Pressure (HP4)	8.719 MPaG (1264.6 psig) Total time delay = 0.7 s
DL3-14	ICS Initiation on Low RPV Water Level	Low RPV Water Level (L2)	L2: 14.224 m (560.0 in) AVZ Total time delay = 1.0 s

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DL ID	Function Name	Inputs	Setpoint / Delay Analytical Limits
DL3-17	MSRIV/MS CIV Isolation on Low RPV Pressure	Low RPV Pressure (LP1)	5.516 MPaG (800.0 psig) Total time delay = 0.7 s In non-LOCA analysis, this is assumed to occur at a lower pressure that is conservative.
DL3-18	MSRIV/MS CIV Isolation on Low RPV Water Level	Low RPV Water Level (L2)	L2: see DL3-14
DL3-23	FW Isolation on High RPV Water Level	High RPV Water level (L9)	L9: 22.377 m (880.984 in) AVZ Total time delay = 1.0 s
DL4a-12	MSRIV/MS CIV Isolation on Sustained Low FW Flow	Sustained Low FW flow	Analysis allows 70 seconds to confirm sustained low FW flow
DL4a-40	CRD Fast Motor Run-In on High Flux After Scram Signal	High Flux After Scram Signal	Analysis allows 5 s after a scram signal to check for high flux and initiate function
DL4a-41	FW Pump/Condensate Pump Trip on High Flux After Scram Signal	High Flux After Scram Signal	Analysis allows 5 s after a scram signal to check for high flux and initiate function
DL4a	ICS Tran Initiation on High Flux After Scram Signal	High Flux After Scram Signal	Analysis allows 5 s after a scram signal to check for high flux and initiate function

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Table 15.5-6: Sequence of Events for Loss of Feedwater Heating (AOO)

Time (s)	Event
0	Initiate a 50°C (90°F) temperature reduction in the FW system
5.5	SCRRI inserts control rods on indication of FW temperature reduction
~60	Control rods stop moving
~400	New steady state achieved

Table 15.5-7: Sequence of Events for Turbine Trip (AOO)

Time (s)	Events
0.0	Reactor scrams on initiation on turbine trip signal
0.2	Scram rods begin to move
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
>6.0	New steady state

Table 15.5-8: Sequence of Events for Closure of One Main Steam Reactor Isolation Valve (AOO)

Time (s)	Event
0.0	Initiate closure of one MSRIV
0.8	Anticipatory scram on MSRIV position
3.0	MSRIV in first steam line is closed
3.0	Closure of MSRIV in second steam line initiated on leak detection indication
6.0	MSRIV in second steam line is closed
>25.0	High RPV pressure reached, ICS train initiated
>25.0	New Steady state

Table 15.5-9: Sequence of Events for Loss of Condenser Vacuum (AOO)

Time (s)	Event
0.0	Anticipatory scram initiated on turbine trip signal caused by loss of condenser vacuum
0.2	Scram rods begin to move
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
24.0	TBVs close on high main condenser pressure
≥24.0	ICS initiation on high RPV pressure
>24.0	New steady state

Table 15.5-10: Sequence of Events for Loss-of-Preferred Power (AOO)

Time (s)	Event
0.0	FW pumps lose power
0.0	Anticipatory scram initiated on generator load rejection caused by generator output breakers opening on loss of power
0.2	Scram rods begin to move
0.25	TCVs begin to close due to generator load rejection signal
0.25	TBVs begin to open
6.25	TBVs close following loss-of-preferred power
>20.0	ICS initiation on high RPV pressure
>20.0	New steady state

Table 15.5-11: Sequence of Events for Feedwater Pump Trip (AOO)

Time (s)	Event
0.0	Initiate FW pump trip
10.0	Standby FW pump starts
25.0	Standby FW pump at 100% rated FW flow
~200	New steady state achieved near 100% power and 100% FW flow

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Table 15.5-12: Sequence of Events for Inadvertent Isolation Condenser Initiation One Train (AOO)

Time (s)	Event
0	Initiate opening of IC condensate return valve on one train
>150	New steady state achieved near initial conditions

Table 15.5-13: Sequence of Events for Loss of Feedwater Heating (DBA)

Time (s)	Event
0	Initiate FW temperature reduction
~90	High STP reached, scram initiated
~110	Low RPV pressure reached, Main Steam isolation initiated
~300	High RPV level L9 reached, FW isolation initiated
> 400	High RPV pressure reached; ICS train initiated (not simulated)

Table 15.5-14: Sequence of Events for Generator Load Rejection (DBA)

Time (s)	Event
0.0	TCVs begin to close due to a generator load rejection signal
0.42	High neutron flux reached, Scram initiated
0.62	Control rods begin to move
152	High RPV pressure reached, ICS train initiated
166	High RPV level L9 reached, FW isolation initiated
>300	A controlled state is achieved

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Table 15.5-15: Sequence of Loss-of-Preferred Power (DBA)

Time (s)	Event
0.0	TCV begin to close, and FW pump trips
0.80	High neutron flux reached, Scram initiated
1.0	Control rods begin to move
152	High RPV pressure reached, ICS train initiated
>300	A controlled state is achieved

Table 15.5-16: RPV Pressure Control Downscale (DBA)

Time (s)	Event
0.0	TCVs begin to close, and TBVs remain closed
0.80	High neutron flux reached, Scram initiated
1.0	Control rods begin to move
155	High RPV pressure is reached, ICS train initiated
167	High RPV level L9 reached, FW isolation initiated
>300	A controlled state is achieved

Table 15.5-17: Closure of All MSRIVs and FW Isolation Valves (DBA)

Time (s)	Event
0.0	MSRIVs begin to close, and FW isolation valves close
1.09	High Neutron Flux reached, Scram initiated
1.29	Control rods begin to move
122	High RPV pressure reached, ICS initiated
>300	A controlled state is achieved

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Table 15.5-18: Sequence of Events for Feedwater Flow Increase – All Pumps

Time (s)	Event
0	Initiate instant increase in speed of both FW pumps
~20	High RPV level L9 reached, FW isolation initiated
~30	High STP reached, Scram initiated
~90	Low RPV pressure reached, Main Steam Reactor Pressure Isolation Valve isolation initiated
>200	High RPV pressure reached, ICS train initiated (not simulated)

Table 15.5-19: Sequence of Events for Condenser Initiation – All Trains

Time (s)	Events
0	Initiate opening of all IC condensate return valves
~35	High RPV level L9 reached, FW isolation initiated
~55	Low RPV pressure reached, scram, and MSRIV isolation initiated

Table 15.5-20: Sequence of Events for Loss of Feedwater (DBA)

Time (s)	Event
0.0	Initiate loss of FW flow
~30	Low RPV level L3 reached, scram initiated
~85	Low RPV pressure reached, MSRIV isolation initiated
~130	Low RPV level L2 reached, all IC Trains initiate
>300	New steady state achieved

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Table 15.5-21: Sequence of Events for RPV Pressure Control Open (DBA)

Time (s)	Event
0.0	TBVs and TCVs open
~70	Low RPV pressure reached, reactor scram and MSRIV isolation initiated
~225	High RPV level L9 reached, FW isolation initiated
>300	High RPV pressure reached, ICS train initiated (not simulated)

Table 15.5-22: Not Used

Table 15.5-23: Timing of Events for Main Steam Pipe Break Inside Containment, CN-DSA

Tim (seconds)	Event	Notes
0.0	Double-ended guillotine rupture of main steam pipe break inside the containment	
0.0	FW pump trip and coast down	This is a consequence of LOPP
0.0	TSV or TCV starts closing	Conservative assumption
1.0	Control rods start to insert on scram initiation	High drywell pressure setpoint for scram, reactor isolation and isolation condenser initiation is reached in less than 1 second.
1.0	ICS condensate return valve starts opening	
1.0	CUW stops	
3.0	Control rods are inserted sufficiently to diminish fission from prompt neutrons	This is a conservatively long duration for fission from prompt neutrons to diminish. Fission power starts decreasing when the control rods are partially inserted. In addition, voiding in the core due to rapid depressurization also causes reactor power to decrease rapidly.
5.0	RIVs start to close	The delay time is significantly conservative given that containment pressure reaches the isolation setpoint in less than 1 second.
10.0	RIVs are fully closed	
11.0	ICS condensate return valve is fully open	
12.2	Peak containment pressure is reached	
>12.2	Containment pressure starts decreasing	

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**Table 15.5-24: Timing of Events for Feedwater Pipe Break
Inside Containment, CN-DSA**

Time (seconds)	Event	Notes
0.0	Double-ended guillotine rupture of FW pipe break inside containment concurrent with LOPP	
0.0	FW pump trip and coast down	This is a consequence of LOPP
0.0	TSV or TCV starts closing	Conservative assumption
1.0	Control rods start being inserted on scram initiation	High drywell pressure setpoint for scram, reactor isolation and isolation condenser initiation is reached in less than 1 second
1.0	ICS-A and B condensate return valve starts opening	
3.0	Control rods are inserted sufficiently to diminish fission from prompt neutrons	This is a conservatively long duration for fission from prompt neutrons
5.0	FWRIVs and CIVs start to close	
10.0	FWRIVs and CIVs are fully closed	
10.1	Peak containment pressure is reached	
11.0	Isolation condenser valves are fully open	

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**Table 15.5-25: Timing of Events for Small Steam Pipe Break
Inside Containment CN-DSA**

Time (seconds)	Event	Notes
0.0	Small steam pipe break concurrent with LOPP	
0.0	Pressure controller freezes. TCV remains in initial position, turbine chest pressure decreases rapidly	Assuming continued steam discharge to the turbine is conservative. TCV closure as a consequence of LOPP is not credited.
0.0	FW pump trip and coast down	
10.6	Steam line low pressure setpoint is reached	
12.3	Reactor scram, 0.7 s delay after low pressure setpoint is reached and another 1 s for scram delay	
15.6	MSRIVs start closing 5 seconds after steam line low pressure	
20.6	MSRIVs are fully closed	
63.6	Level decreases to Level 2	
64.6	ICS-A and ICS-B condensate return valves start opening	
74.6	ICS-A and ICS-B condensate return valves are fully open	
234000	Peak containment pressure reached	

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Table 15.5-26: Timing of Events for Small Liquid Pipe Break Inside Containment, CN-DSA

Time (seconds)	Event	Notes
0.0	Small liquid pipe break concurrent with LOPP	
0.0	Pressure controller freezes. TCV remains at initial position, turbine chest pressure decreases rapidly	Assuming continued steam discharge to the turbine is conservative. TCV closure as a consequence of LOPP is not credited.
0.0	FW pump trip and coast down	
10.6	Steam line low pressure setpoint is reached	
12.3	Reactor scram, 0.7 s delay after low pressure setpoint is reached and another 1 s for scram delay	
15.6	MSRIVs start closing 5 seconds after steam pipe low pressure	
20.6	MSRIVs are fully closed	
58.9	Level decreases to Level 2	
59.9	ICS-A and ICS-B condensate return valves start opening	
69.9	ICS-A and ICS-B condensate return valves are fully open	
232800	Peak containment pressure occurs	

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Table 15.5-27: Sequence of Events for Closure of One Main Steam Reactor Pressure Isolation Valve (DEC)

Time (s)	Event
0.0	Initiate closure of one MSIV
0.8	Anticipatory scram signal on MSRIV position. Scram fails
3.0	MSRIV in first steam line is closed
3.0	Closure of MSRIV in second steam line initiated on leak detection indication
5.8	Feedwater pump trips on high flux after scram signal
5.8	All ICS trains initiate on high flux after scram signal
5.8	CRDM run-in initiation on high flux after scram signal
6.0	MSRIV in second steam line is closed
>300	A controlled state is achieved

Table 15.5-28: Sequence of Events for Complex Sequence Generator Load Rejection (DEC)

Time (s)	Event
0.0	Scram initiation on load rejection signal
0.2	Control rods begin to move. Half of the control rods fail to insert
0.25	TCVs begin to close
0.25	TBVs begin to open
>200	A controlled state is achieved

Table 15.5-29: Sequence of Events for Loss of Condenser Vacuum (DEC)

Time (s)	Event
0.0	Scram fails on initiation on turbine trip signal caused by loss of condenser vacuum
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
5.0	CRDM run-in initiation on high flux after scram signal
5.0	ICS initiation on high flux after scram signal
24.0	TBVs close on high main condenser pressure
>300	A controlled state is achieved

Table 15.5-30: Sequence of Events for Loss of Preferred Power (DEC)

Time (s)	Event
0.0	FW pumps lose power
0.0	Scram fails on generator load rejection caused by generator output breakers opening on loss of power
0.25	TCVs begin to close due to generator load rejection signal
0.25	TBVs begin to open
5.0	CRDM run-in initiation on high flux after scram signal
5.0	ICS initiation on high flux after scram signal
6.25	TBVs close following loss-of-preferred power
>300	A controlled state is achieved

Table 15.5-31: Sequence of Events for All Control Rod Withdrawal at Power (ACRW)

Time (s)	Event
0.1	Initiate withdrawal of all rods
19	High STP reached, scram initiated
63	High RPV Level L9 reached, FW isolation initiated
84	Low RPV pressure reached, Main Steam isolation initiated
> 200	High RPV pressure reached; ICS train initiates on high pressure (not simulated)

Table 15.5-32: Sequence of Events for Inadvertent Control Rod Withdrawal at Power Single Rod (ICRW)

Time (s)	Event
0.1	Initiate withdrawal of one rod
>70	Power reaches a stable level
long term	Operators act to control reactor power

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Table 15.5-33: Sequence of Events for Feedwater Isolation (DEC)

Time (s)	Event
0.0	Initiate FW Isolation
5	FW flow at zero
~30	Low RPV level L3 reached, no scram (DL3 failure)
65	Reactor scram initiated on sustained low FW flow
70	MSRIV isolation initiated on sustained low FW flow
250	ICS Trains initiate on high RPV pressure
>400	Controlled state achieved

Table 15.5-34: Fuel Handling Accident Sequence of Events

Sequence of Events	Elapsed Time
The BWRX-300 reactor is shut down for refueling operations that begins 24 hours after shutdown. Over this period the core isotopic inventory decays.	24 hours
During a refueling operation a fuel assembly is moved over the top of the core or fuel pool, and the fuel bundle, grapple, mast, and head fall on top of the core or spent fuel racks.	0
Rods in the dropped bundle and struck bundles fail, releasing the fission gases in the plenum and gap of the damaged rods to the pool water.	0
Fission gases rise through the pool water to refueling operation floor common airspace surrounding the top of the reactor cavity and spent fuel pool.	0
The building ventilation system high radiation alarm alerts all plant personnel to evacuate, and the Reactor Building (RB) air handling system dampers close to isolate the area.	0
Plant workers evacuate the RB and adjacent areas.	15 minutes

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Table 15.5-35: BWRX-300 Core Parameters

Parameter/Description	Value
Thermal Power	870(MWth)
Core Size (Number of Bundles)	240
Fuel Type	GNF2
Bundle Average Enrichment	3.84-4.68 ^w % U235
Core Average Exposure	38,000 MWd/Mt

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Table 15.5-36: BWRX-300 Core Inventory 24 Hours After Shutdown

Nuclide	Activity (Ci/MWth)	Activity (MBq/MWth)	Nuclide	Activity (Ci/MWth)	Activity (MBq/MWth)
Co-58	3.08E+02	1.14E+07	Te-131m	2.31E+03	8.55E+07
Co-60	6.51E+02	2.41E+07	Te-132	3.13E+04	1.16E+09
Kr-85	5.40E+02	2.00E+07	I-131	2.54E+04	9.40E+08
Kr-85m	1.78E+02	6.59E+06	I-132	3.23E+04	1.20E+09
Kr-87	2.91E-02	1.08E+03	I-133	2.54E+04	9.40E+08
Kr-88	5.56E+01	2.06E+06	I-134	1.36E-03	5.03E+01
Rb-86	7.40E+01	2.74E+06	I-135	4.16E+03	1.54E+08
Sr-89	2.56E+04	9.47E+08	Xe-133	5.16E+04	1.91E+09
Sr-90	4.48E+03	1.66E+08	Xe-135	1.49E+04	5.51E+08
Sr-91	5.70E+03	2.11E+08	Cs-134	9.46E+03	3.50E+08
Sr-92	7.64E+01	2.83E+06	Cs-136	2.60E+03	9.62E+07
Y-90	4.62E+03	1.71E+08	Cs-137	5.97E+03	2.21E+08
Y-91	3.32E+04	1.23E+09	Ba-139	3.20E-01	1.18E+04
Y-92	1.13E+03	4.18E+07	Ba-140	4.51E+04	1.67E+09
Y-93	8.00E+03	2.96E+08	La-140	4.88E+04	1.81E+09
Zr-95	4.73E+04	1.75E+09	La-141	7.08E+02	2.62E+07
Zr-97	1.84E+04	6.81E+08	La-142	1.03E+00	3.81E+04
Nb-95	4.80E+04	1.78E+09	Ce-141	4.44E+04	1.64E+09
Mo-99	4.00E+04	1.48E+09	Ce-143	2.55E+04	9.44E+08
Tc-99m	3.81E+04	1.41E+09	Ce-144	3.82E+04	1.41E+09
Ru-103	4.33E+04	1.60E+09	Pr-143	4.02E+04	1.49E+09
Ru-105	7.65E+02	2.83E+07	Nd-147	1.70E+04	6.29E+08
Ru-106	2.05E+04	7.59E+08	Np-239	4.53E+05	1.68E+10
Rh-105	2.15E+04	7.96E+08	Pu-238	2.28E+02	8.44E+06
Sb-127	2.63E+03	9.73E+07	Pu-239	1.87E+01	6.92E+05
Sb-129	1.95E+02	7.22E+06	Pu-240	2.70E+01	9.99E+05
Te-127	2.87E+03	1.06E+08	Pu-241	7.83E+03	2.90E+08
Te-127m	4.19E+02	1.55E+07	Am-241	1.47E+01	5.44E+05
Te-129	1.08E+03	4.00E+07	Cm-242	3.03E+03	1.12E+08
Te-129m	1.31E+03	4.85E+07	Cm-244	2.15E+02	7.96E+06

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Table 15.5-37: Non-LOCA Fraction of Fission Product Inventory in Gap

Group	Gap Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

Table 15.5-38: BWRX-300 FHA Activity Released from Fuel

Nuclide	Core Inventory (Ci/MWth)	Core Inventory (MBq/MWth)	Activity Released from Fuel (Ci)	Activity Released from Fuel (MBq)
Kr-85m	1.78E+02	6.59E+06	8.20E+01	3.03E+06
Kr-85	5.40E+02	2.00E+07	4.98E+02	1.84E+07
Kr-87	2.91E-02	1.08E+03	1.34E-02	4.96E+02
Kr-88	5.56E+01	2.06E+06	2.56E+01	9.48E+05
I-131	2.54E+04	9.40E+08	1.87E+04	6.93E+08
I-132	3.23E+04	1.20E+09	1.49E+04	5.51E+08
I-133	2.54E+04	9.40E+08	1.17E+04	4.33E+08
I-134	1.36E-03	5.03E+01	6.27E-04	2.32E+01
I-135	4.16E+03	1.54E+08	1.92E+03	7.09E+07
Xe-133	5.16E+04	1.91E+09	2.38E+04	8.80E+08
Xe-135	1.49E+04	5.51E+08	6.87E+03	2.54E+08

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**Table 15.5-39: BWRX-300 Fuel Handling Accident (FHA) Activity Released
from the Reactor Cavity Pool**

Nuclide	Activity Released to the Environment (TBq)
Kr-85m	2.80E+00
Kr-85	1.80E+01
Kr-87	3.70E-04
Kr-88	8.30E-01
I-131	3.40E+00
I-132	2.30E+00
I-133	2.10E+00
I-134	7.90E-08
I-135	3.30E-01
Xe-133	8.80E+02
Xe-135	2.40E+02

Table 15.5-40: Not Used

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Table 15.5-41: FWLB Line Break Accident Airborne Release Source Term

Nuclide	Equilibrium Release (TBq)	Spike Release (TBq)
Kr-83m	1.76E-03	1.76E-03
Kr-85m	4.84E-05	4.84E-05
Kr-85	1.17E-05	1.17E-05
Kr-87	3.30E-04	3.30E-04
Kr-88	1.83E-04	1.83E-04
Kr-89	6.30E-02	6.30E-02
Xe-131m	1.03E-05	1.03E-05
Xe-133m	6.01E-06	6.01E-06
Xe-133	8.80E-05	8.80E-05
Xe-135m	1.61E-03	1.61E-03
Xe-135	8.80E-04	8.80E-04
Xe-137	3.01E-03	3.01E-03
Xe-138	8.06E-03	8.06E-03
I-131	4.89E-02	9.60E-01
I-132	5.25E-01	1.03E+01
I-133	3.62E-01	7.24E+00
I-134	1.58E+00	3.08E+01
I-135	7.24E-01	1.43E+01
Rb-89	8.50E-01	8.50E-01
Cs-134	1.03E-02	1.03E-02
Cs-136	8.14E-03	8.14E-03
Cs-137	1.57E-02	1.57E-02
Cs-138	9.05E-01	9.05E-01
Ba-137m	1.57E-02	1.57E-02
HTO	9.79E-03	9.79E-03
Cr-51	5.43E-04	5.43E-04
Mn-54	2.71E-04	2.71E-04
Fe-59	1.45E-04	1.45E-04
Co-60	2.53E-04	2.53E-04
Cu-64	2.71E-03	2.71E-03
Sr-89	1.14E-03	1.14E-03
Sr-90	5.79E-05	5.79E-05

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Nuclide	Equilibrium Release (TBq)	Spike Release (TBq)
Y-90	5.79E-05	5.79E-05
Sr-91	7.42E-01	7.42E-01
Sr-92	1.70E+00	1.70E+00
Y-91	1.52E-02	1.52E-02
Y-92	5.07E-01	5.07E-01
Y-93	5.25E-02	5.25E-02
Zr-95	3.08E-02	3.08E-02
Nb-95	3.08E-02	3.08E-02
Mo-99	1.50E-01	1.50E-01
Tc-99m	1.50E-01	1.50E-01
Ru-103	7.60E-03	7.60E-03
Rh-103m	7.60E-03	7.60E-03
Ru-106	1.14E-03	1.14E-03
Rh-106	1.14E-03	1.14E-03
Te-129m	1.52E-02	1.52E-02
Te-131m	1.99E-02	1.99E-02
Te-132	3.62E-03	3.62E-03
Ba-140	1.59E-01	1.59E-01
La-140	1.59E-01	1.59E-01
Ce-141	7.60E-03	7.60E-03
Ce-144	1.14E-03	1.14E-03
Pr-144	1.14E-03	1.14E-03
Np-239	1.23E-01	1.23E-01

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Table 15.5-42: ICS Line Break Accident Airborne Release Source Term

Nuclide	Equilibrium Release (TBq)	Spike Release (TBq)
Kr-83m	9.62E-04	9.62E-04
Kr-85m	2.65E-05	2.65E-05
Kr-85	6.41E-06	6.41E-06
Kr-87	1.80E-04	1.80E-04
Kr-88	1.00E-04	1.00E-04
Kr-89	3.45E-02	3.45E-02
Xe-131m	5.61E-06	5.61E-06
Xe-133m	3.29E-06	3.29E-06
Xe-133	4.81E-05	4.81E-05
Xe-135m	8.82E-04	8.82E-04
Xe-135	4.81E-04	4.81E-04
Xe-137	1.64E-03	1.64E-03
Xe-138	4.41E-03	4.41E-03
I-131	2.20E-05	4.41E-04
I-132	2.20E-04	4.41E-03
I-133	1.60E-04	3.21E-03
I-134	6.81E-04	1.40E-02
I-135	3.05E-04	6.01E-03
Rb-89	1.88E-05	1.88E-05
Cs-134	2.28E-07	2.28E-07
Cs-136	1.80E-07	1.80E-07
Cs-137	3.49E-07	3.49E-07
Cs-138	2.00E-05	2.00E-05
Ba-137m	3.49E-07	3.49E-07
HTO	2.08E-04	2.08E-04
Cr-51	1.20E-08	1.20E-08
Mn-54	6.01E-09	6.01E-09
Fe-59	3.21E-09	3.21E-09
Co-60	5.61E-09	5.61E-09
Cu-64	6.01E-08	6.01E-08
Sr-89	2.53E-08	2.53E-08
Sr-90	1.28E-09	1.28E-09

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Nuclide	Equilibrium Release (TBq)	Spike Release (TBq)
Y-90	1.28E-09	1.28E-09
Sr-91	1.64E-05	1.64E-05
Sr-92	3.77E-05	3.77E-05
Y-91	3.37E-07	3.37E-07
Y-92	1.12E-05	1.12E-05
Y-93	1.16E-06	1.16E-06
Zr-95	6.81E-07	6.81E-07
Nb-95	6.81E-07	6.81E-07
Mo-99	3.33E-06	3.33E-06
Tc-99m	3.33E-06	3.33E-06
Ru-103	1.68E-07	1.68E-07
Rh-103m	1.68E-07	1.68E-07
Ru-106	2.53E-08	2.53E-08
Rh-106	2.53E-08	2.53E-08
Te-129m	3.37E-07	3.37E-07
Te-131m	4.41E-07	4.41E-07
Te-132	8.02E-08	8.02E-08
Ba-140	3.53E-06	3.53E-06
La-140	3.53E-06	3.53E-06
Ce-141	1.68E-07	1.68E-07
Ce-144	2.53E-08	2.53E-08
Pr-144	2.53E-08	2.53E-08
Np-239	2.73E-06	2.73E-06

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Table 15.5-43A: Instrument Line Break (ILB) Accident Airborne Equilibrium Release Source Term

Radionuclides	Activity Released to the Environment in 0-1800 seconds(TBq)	Activity Released to the Environment in 1800 - 9700 seconds(TBq)	Activity Released to the Environment in 9700 - 15500 seconds(TBq)	Activity Released to the Environment in 15500 - 20500 seconds(TBq)
Kr-83m	2.12E-03	5.56E-03	1.35E-03	5.82E-04
Kr-85m	5.82E-05	1.53E-04	3.72E-05	1.60E-05
Kr-85	1.41E-05	3.71E-05	9.01E-06	3.88E-06
Kr-87	3.97E-04	1.04E-03	2.53E-04	1.09E-04
Kr-88	2.20E-04	5.80E-04	1.41E-04	6.07E-05
Kr-89	7.58E-02	1.99E-01	4.84E-02	2.09E-02
Xe-131m	1.23E-05	3.25E-05	7.88E-06	3.40E-06
Xe-133m	7.23E-06	1.90E-05	4.62E-06	1.99E-06
Xe-133	1.06E-04	2.78E-04	6.76E-05	2.91E-05
Xe-135m	1.94E-03	5.10E-03	1.24E-03	5.34E-04
Xe-135	1.06E-03	2.78E-03	6.76E-04	2.91E-04
Xe-137	3.62E-03	9.51E-03	2.31E-03	9.95E-04
Xe-138	9.70E-03	2.55E-02	6.19E-03	2.67E-03
I-131	6.29E-03	2.03E-02	9.49E-03	4.09E-03
I-132	6.76E-02	2.18E-01	1.02E-01	4.39E-02
I-133	4.66E-02	1.50E-01	7.03E-02	3.03E-02
I-134	2.03E-01	6.53E-01	3.06E-01	1.32E-01
I-135	9.32E-02	3.00E-01	1.41E-01	6.06E-02
Rb-89	1.09E-01	3.51E-01	1.65E-01	7.10E-02
Cs-134	1.32E-03	4.25E-03	2.00E-03	8.61E-04

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Radionuclides	Activity Released to the Environment in 0-1800 seconds(TBq)	Activity Released to the Environment in 1800 - 9700 seconds(TBq)	Activity Released to the Environment in 9700 - 15500 seconds(TBq)	Activity Released to the Environment in 15500 - 20500 seconds(TBq)
Cs-136	1.04E-03	3.36E-03	1.58E-03	6.79E-04
Cs-137	2.01E-03	6.49E-03	3.05E-03	1.31E-03
Cs-138	1.16E-01	3.73E-01	1.75E-01	7.55E-02
Ba-137m	2.01E-03	6.49E-03	3.05E-03	1.31E-03
HTO	1.66E-03	5.08E-03	2.11E-03	9.11E-04
Cr-51	6.94E-05	2.24E-04	1.05E-04	4.53E-05
Mn-54	3.47E-05	1.12E-04	5.25E-05	2.26E-05
Fe-59	1.85E-05	5.97E-05	2.80E-05	1.21E-05
Co-60	3.24E-05	1.04E-04	4.90E-05	2.11E-05
Cu-64	3.47E-04	1.12E-03	5.25E-04	2.26E-04
Sr-89	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Sr-90	7.40E-06	2.39E-05	1.12E-05	4.83E-06
Y-90	7.40E-06	2.39E-05	1.12E-05	4.83E-06
Sr-91	9.49E-02	3.06E-01	1.44E-01	6.19E-02
Sr-92	2.17E-01	7.01E-01	3.29E-01	1.42E-01
Y-91	1.94E-03	6.27E-03	2.94E-03	1.27E-03
Y-92	6.48E-02	2.09E-01	9.81E-02	4.23E-02
Y-93	6.71E-03	2.16E-02	1.02E-02	4.38E-03
Zr-95	3.93E-03	1.27E-02	5.95E-03	2.57E-03
Nb-95	3.93E-03	1.27E-02	5.95E-03	2.57E-03
Mo-99	1.92E-02	6.19E-02	2.91E-02	1.25E-02

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Radionuclides	Activity Released to the Environment in 0-1800 seconds(TBq)	Activity Released to the Environment in 1800 - 9700 seconds(TBq)	Activity Released to the Environment in 9700 - 15500 seconds(TBq)	Activity Released to the Environment in 15500 - 20500 seconds(TBq)
Tc-99m	1.92E-02	6.19E-02	2.91E-02	1.25E-02
Ru-103	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Rh-103m	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Ru-106	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Rh-106	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Te-129m	1.94E-03	6.27E-03	2.94E-03	1.27E-03
Te-131m	2.54E-03	8.21E-03	3.85E-03	1.66E-03
Te-132	4.63E-04	1.49E-03	7.01E-04	3.02E-04
Ba-140	2.04E-02	6.57E-02	3.08E-02	1.33E-02
La-140	2.04E-02	6.57E-02	3.08E-02	1.33E-02
Ce-141	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Ce-144	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Pr-144	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Np-239	1.57E-02	5.07E-02	2.38E-02	1.03E-02

Table 15.5-43B: Instrument Line Break (ILB) Accident Airborne Iodine Spike Release Source Term

Radionuclides	Activity Released to the Environment in 0-1800 seconds (TBq)	Activity Released to the Environment in 1800 - 9700 seconds (TBq)	Activity Released to the Environment in 9700 - 15500 seconds (TBq)	Activity Released to the Environment in 15500 - 20500 seconds (TBq)
Kr-83m	2.12E-03	5.56E-03	1.35E-03	5.82E-04
Kr-85m	5.82E-05	1.53E-04	3.72E-05	1.60E-05
Kr-85	1.41E-05	3.71E-05	9.01E-06	3.88E-06
Kr-87	3.97E-04	1.04E-03	2.53E-04	1.09E-04
Kr-88	2.20E-04	5.80E-04	1.41E-04	6.07E-05
Kr-89	7.58E-02	1.99E-01	4.84E-02	2.09E-02
Xe-131m	1.23E-05	3.25E-05	7.88E-06	3.40E-06
Xe-133m	7.23E-06	1.90E-05	4.62E-06	1.99E-06
Xe-133	1.06E-04	2.78E-04	6.76E-05	2.91E-05
Xe-135m	1.94E-03	5.10E-03	1.24E-03	5.34E-04
Xe-135	1.06E-03	2.78E-03	6.76E-04	2.91E-04
Xe-137	3.62E-03	9.51E-03	2.31E-03	9.95E-04
Xe-138	9.70E-03	2.55E-02	6.19E-03	2.67E-03
I-131	1.24E-01	3.98E-01	1.86E-01	8.03E-02
I-132	1.33E+00	4.28E+00	2.00E+00	8.63E-01
I-133	9.32E-01	3.00E+00	1.41E+00	6.06E-01
I-134	3.96E+00	1.28E+01	5.97E+00	2.57E+00
I-135	1.84E+00	5.93E+00	2.78E+00	1.20E+00
Rb-89	1.09E-01	3.51E-01	1.65E-01	7.10E-02
Cs-134	1.32E-03	4.25E-03	2.00E-03	8.61E-04

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Radionuclides	Activity Released to the Environment in 0-1800 seconds (TBq)	Activity Released to the Environment in 1800 - 9700 seconds (TBq)	Activity Released to the Environment in 9700 - 15500 seconds (TBq)	Activity Released to the Environment in 15500 - 20500 seconds (TBq)
Cs-136	1.04E-03	3.36E-03	1.58E-03	6.79E-04
Cs-137	2.01E-03	6.49E-03	3.05E-03	1.31E-03
Cs-138	1.16E-01	3.73E-01	1.75E-01	7.55E-02
Ba-137m	2.01E-03	6.49E-03	3.05E-03	1.31E-03
HTO	1.66E-03	5.08E-03	2.11E-03	9.11E-04
Cr-51	6.94E-05	2.24E-04	1.05E-04	4.53E-05
Mn-54	3.47E-05	1.12E-04	5.25E-05	2.26E-05
Fe-59	1.85E-05	5.97E-05	2.80E-05	1.21E-05
Co-60	3.24E-05	1.04E-04	4.90E-05	2.11E-05
Cu-64	3.47E-04	1.12E-03	5.25E-04	2.26E-04
Sr-89	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Sr-90	7.40E-06	2.39E-05	1.12E-05	4.83E-06
Y-90	7.40E-06	2.39E-05	1.12E-05	4.83E-06
Sr-91	9.49E-02	3.06E-01	1.44E-01	6.19E-02
Sr-92	2.17E-01	7.01E-01	3.29E-01	1.42E-01
Y-91	1.94E-03	6.27E-03	2.94E-03	1.27E-03
Y-92	6.48E-02	2.09E-01	9.81E-02	4.23E-02
Y-93	6.71E-03	2.16E-02	1.02E-02	4.38E-03
Zr-95	3.93E-03	1.27E-02	5.95E-03	2.57E-03
Nb-95	3.93E-03	1.27E-02	5.95E-03	2.57E-03
Mo-99	1.92E-02	6.19E-02	2.91E-02	1.25E-02

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Radionuclides	Activity Released to the Environment in 0-1800 seconds (TBq)	Activity Released to the Environment in 1800 - 9700 seconds (TBq)	Activity Released to the Environment in 9700 - 15500 seconds (TBq)	Activity Released to the Environment in 15500 - 20500 seconds (TBq)
Tc-99m	1.92E-02	6.19E-02	2.91E-02	1.25E-02
Ru-103	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Rh-103m	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Ru-106	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Rh-106	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Te-129m	1.94E-03	6.27E-03	2.94E-03	1.27E-03
Te-131m	2.54E-03	8.21E-03	3.85E-03	1.66E-03
Te-132	4.63E-04	1.49E-03	7.01E-04	3.02E-04
Ba-140	2.04E-02	6.57E-02	3.08E-02	1.33E-02
La-140	2.04E-02	6.57E-02	3.08E-02	1.33E-02
Ce-141	9.72E-04	3.13E-03	1.47E-03	6.34E-04
Ce-144	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Pr-144	1.46E-04	4.70E-04	2.21E-04	9.51E-05
Np-239	1.57E-02	5.07E-02	2.38E-02	1.03E-02

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Table 15.5-44: Main Steam Line Break Accident Airborne Release Source Term

Isotope	Equilibrium Iodine	Pre-Incident Iodine Spike
	TBq	TBq
Kr-83m	5.76E-03	5.76E-03
Kr-85m	1.58E-04	1.58E-04
Kr-85	3.84E-05	3.84E-05
Kr-87	1.08E-03	1.08E-03
Kr-88	6.00E-04	6.00E-04
Kr-89	2.06E-01	2.06E-01
Xe-131m	3.36E-05	3.36E-05
Xe-133m	1.97E-05	1.97E-05
Xe-133	2.88E-04	2.88E-04
Xe-135m	5.28E-03	5.28E-03
Xe-135	2.88E-03	2.88E-03
Xe-137	9.84E-03	9.84E-03
Xe-138	2.64E-02	2.64E-02
I-131	4.30E-02	8.44E-01
I-132	4.62E-01	9.08E+00
I-133	3.19E-01	6.37E+00
I-134	1.39E+00	2.71E+01
I-135	6.37E-01	1.26E+01
Rb-89	7.46E-01	7.46E-01
Cs-134	9.05E-03	9.05E-03
Cs-136	7.15E-03	7.15E-03
Cs-137	1.38E-02	1.38E-02
Cs-138	7.94E-01	7.94E-01
Ba-137m	1.38E-02	1.38E-02
HTO	9.50E-03	9.50E-03
Cr-51	4.76E-04	4.76E-04
Mn-54	2.38E-04	2.38E-04
Fe-59	1.27E-04	1.27E-04
Co-60	2.22E-04	2.22E-04
Cu-64	2.38E-03	2.38E-03
Sr-89	1.00E-03	1.00E-03

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Isotope	Equilibrium Iodine	Pre-Incident Iodine Spike
	TBq	TBq
Sr-90	5.08E-05	5.08E-05
Y-90	5.08E-05	5.08E-05
Sr-91	6.51E-01	6.51E-01
Sr-92	1.49E+00	1.49E+00
Y-91	1.33E-02	1.33E-02
Y-92	4.45E-01	4.45E-01
Y-93	4.61E-02	4.61E-02
Zr-95	2.70E-02	2.70E-02
Nb-95	2.70E-02	2.70E-02
Mo-99	1.32E-01	1.32E-01
Tc-99m	1.32E-01	1.32E-01
Ru-103	6.67E-03	6.67E-03
Rh-103m	6.67E-03	6.67E-03
Ru-106	1.00E-03	1.00E-03
Rh-106	1.00E-03	1.00E-03
Te-129m	1.33E-02	1.33E-02
Te-131m	1.75E-02	1.75E-02
Te-132	3.18E-03	3.18E-03
Ba-140	1.40E-01	1.40E-01
La-140	1.40E-01	1.40E-01
Ce-141	6.67E-03	6.67E-03
Ce-144	1.00E-03	1.00E-03
Pr-144	1.00E-03	1.00E-03
Np-239	1.08E-01	1.08E-01

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Table 15.5-45: Description of Criticality Accident Parameters

Details	Parameters
Site/Date	SACLAY (1960/03/15)
Critical Mass	2200 kg
Critical Geometry	Oxide Fuel in Water
Total # Fissions	3×10^{18}
Fission Excursion	< 180 sec.

Table 15.5-46: Out of Core Criticality Scenario Results

Total Number of Fissions	4×10^{19} fissions
Excursion Duration	< 180 s
Total Radiation Dose (350 m)	0.4 mSv

Table 15.5-47: Protective Actions and Generic Criteria for OCCC

Protective Actions	Generic Criteria
Evacuation	50 mSv effective dose in 7 days
Sheltering	5 mSv effective dose in 1 day
Temporary Relocation	50 mSv in the first year

Table 15.5-48: Conservatisms Used in the Non-LOCA DSA

Plant State	Conservatism	
	Code	Plant Parameters and System Performances
AOO Base Line Analyses	Best Estimate	<ul style="list-style-type: none"> Rated power initial conditions Conservative setpoints and plant performance parameters Conservatism in the plant parameters and the derived acceptance criteria is established conservatively such that there is no need to account for uncertainties in the analysis method DL2 functions are credited in the analyses, but DL3 functions are credited if needed to meet acceptance criteria
AOO Conservative	Best Estimate	Fault evaluations indicate that failures in DL2 result in DBA events
DBA Baseline	Best Estimate	<ul style="list-style-type: none"> Rated power initial conditions Conservative setpoints and plant performance parameters Conservatism in the plant parameters and the derived acceptance criteria is established conservatively such that there is no need to account for uncertainties in the analysis method DL2 and DL3 functions are credited if needed to meet acceptance criteria
DBA Conservative	Graded Approach	<ul style="list-style-type: none"> Conservative initial conditions are established (e.g., bias fuel and reactor pressure initial) for events that are limiting Conservative setpoints and plant performance parameters Analysis method uncertainties are addressed with a graded approach depending on margin to the derived acceptance criteria
DEC	Best Estimate	<ul style="list-style-type: none"> Rated power initial conditions are used Nominal setpoints and plant performance parameters are used (conservative setpoints and plant performance may be used for convenience/simplification) Sensitivity analyses performed to understand cliff edge effects

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Table 15.5-49: DL3 Functions Credited in Conservative LOCA Analyses

Credited Function	Action	Signal	Analytical Setpoint
DL3-02	Hydraulic scram	Low steam pipe pressure	5.617 MPa
DL3-07	Hydraulic scram	High containment pressure	18.5 kPaG (2.8 psig)
DL3-09	Hydraulic scram	Line Break Indication (MS, FW, ICS)	For breaks larger than 19 mm (0.75 in) in diameter (Note 1)
DL3-14	ICS initiation	Low RPV water level	14.22 m
DL3-15	ICS initiation	High Containment pressure	18.5 kPaG (2.8 psig)
DL3-16	ICS initiation	Line Break Indication (MS, FW, ICS)	For breaks larger than 19 mm (0.75 in) in diameter (Note 1)
DL3-17	MS RIV closure (Note 2)	Low steam pipe pressure	5.617 MPa
DL3-18	MS RIV closure (Note 2)	Low RPV water level	14.22 m
DL3-22	Reactor and Containment Isolation Valve closure (Note 2)	High Containment pressure	18.5 kPaG (2.8 psig) (Note 1)
DL3-20 DL3-21	MS RIV closure	Line break indication in MSL, FW or SDC	Within 1 seconds for breaks larger than 19 mm (0.75 in) in diameter
DL3-25	FW and SDC RIV closure	Line break indication in FW or SDC	Within 1 second for breaks larger than 19 mm (0.75 in) in diameter
DL3-26	CUW RIV closure	Line break indication in CUW	Within 1 second for breaks larger than 19 mm (0.75 in) in diameter
DL3-27 DL3-28 DL3-29	ICS RIV closure of the broken ICS train	Line break indication in the respective ICS trains	Within 1 second for breaks larger than 19 mm (0.75 in) in diameter

- (1) This setpoint is reached in less than 1 s in large break cases.
- (2) For large breaks, isolation valves are assumed to start closing with a 5 second delay from the time of pipe break and are fully closed in 10 seconds. For small breaks, the isolation valves start closing with a 5 second delay after the setpoint is reached and are fully closed in another 5 seconds. Containment isolation is credited only for FW pipe breaks. Isolation functions may also include CIVs. This table shows only the valves credited to close by the isolation signal.

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Table 15.5-50: DL2 and DL4a Functions Credited in DEC LOCA Analyses

DL3 Function Failure	Action	DL2 or DL4a Credited Function	Notes
DL3-02	Hydraulic scram on low steam pipe pressure	DL2-08 on turbine trip demand	DL2-08 is initiated on LOPP, and a faster scram than DL3-02 credited in CN-DSA.
DL3-07	Hydraulic scram on high containment pressure	DL4a-05	
DL3-09	Hydraulic scram on Line Break Indication (MS, FW, ICS)	DL4a-26	
DL3-14	ICS initiation on Low RPV water level	DL4a-33	
DL3-15	ICS initiation on High Containment pressure	DL4a-11	
DL3-16	ICS initiation on Line Break Indication (MS, FW, ICS)	DL4a-27 and DL4a-28	
DL3-17	MS RIV closure on Low steam pipe pressure	DL2-41	DL3-17 is credited in small break cases for limiting inventory losses to the turbine. In DL2, the pressure controller throttles the flow until DL2-41 closes the MS RIVs. This combination of functions results in less inventory losses than the CN-DSA sequences.
DL3-18	MS RIV closure on Low RPV water level	DL4a-34	
DL3-22	Reactor and Containment Isolation Valve closure on High Containment pressure	DL4a-16	
DL3-20 DL3-21	MSRIV closure on Line break indication in MSL, FW or SDC	DL4a-14 and DL4a-15	
DL3-25	FW and SDC RIV closure on line break indication in FW or SDC	DL4a-17	
DL3-27 DL3-28 DL3-29	ICRIV closure of the broken ICS train line break indication in the respective ICS train	None	See discussion in 15.2.4.8.1.

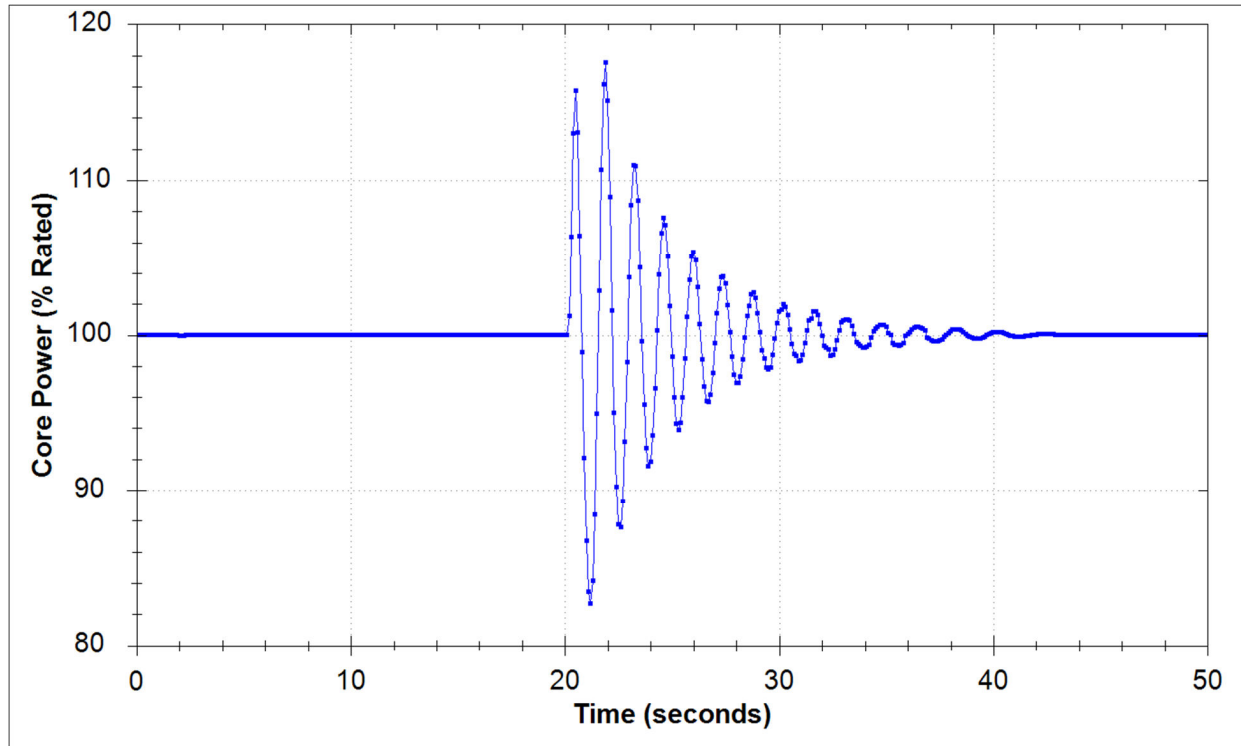


Figure 15.5-1: MOC Transient Results for Core-Wide Stability Analysis

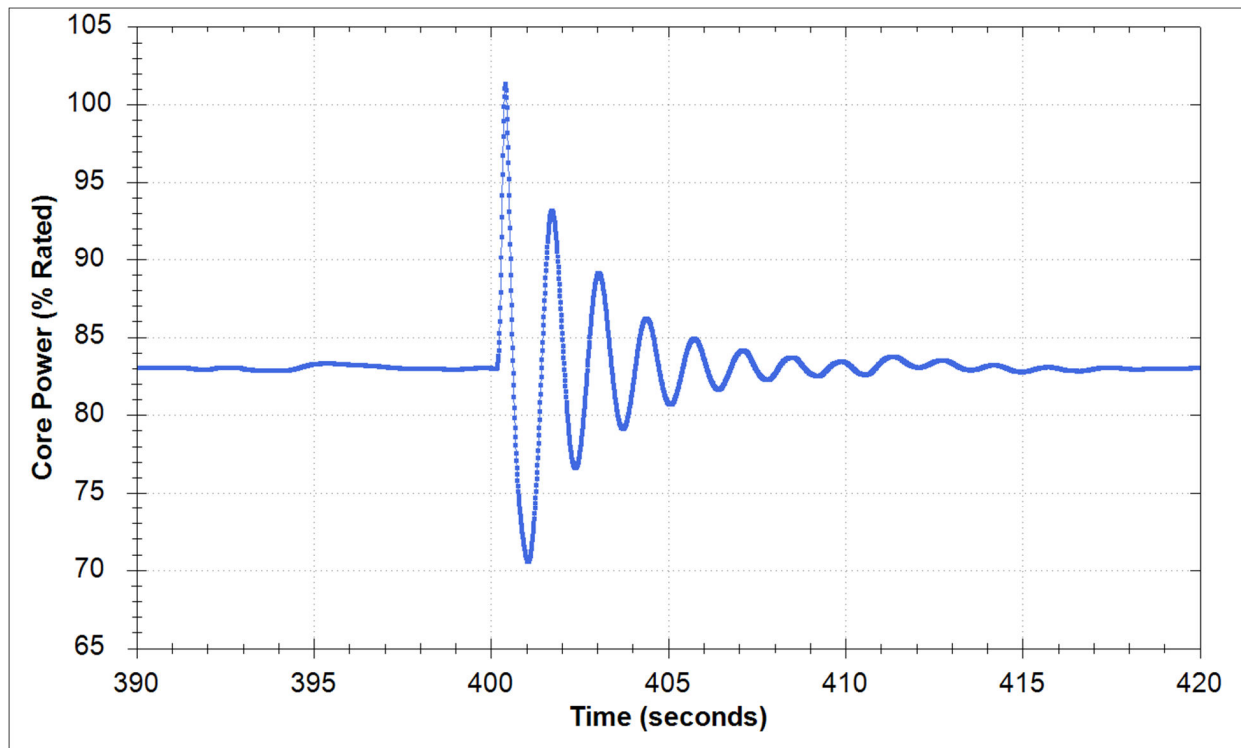


Figure 15.5-2: MOC Transient Results for LFWH with SCCRI Core-Wide Stability Analysis

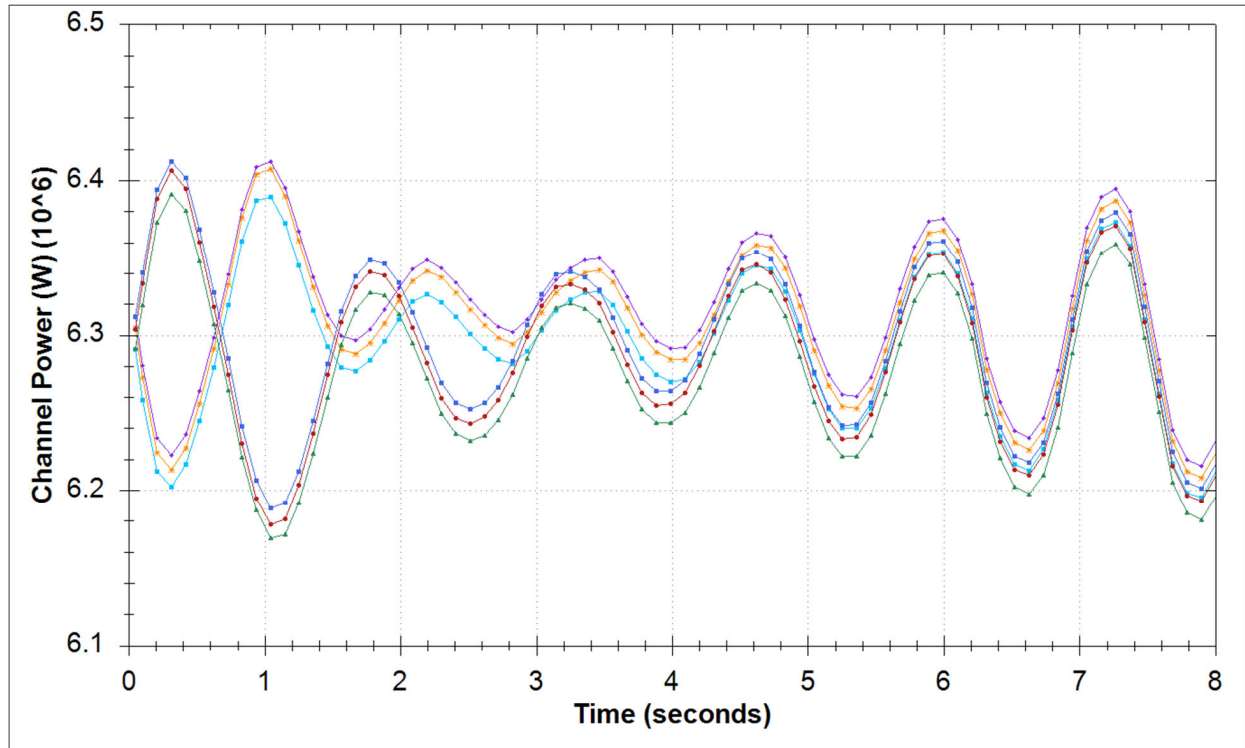


Figure 15.5-3: Regional Mode Stability Response at MOC for FW Temperature of 241.9°C

Figure 15.5-4: Not Used

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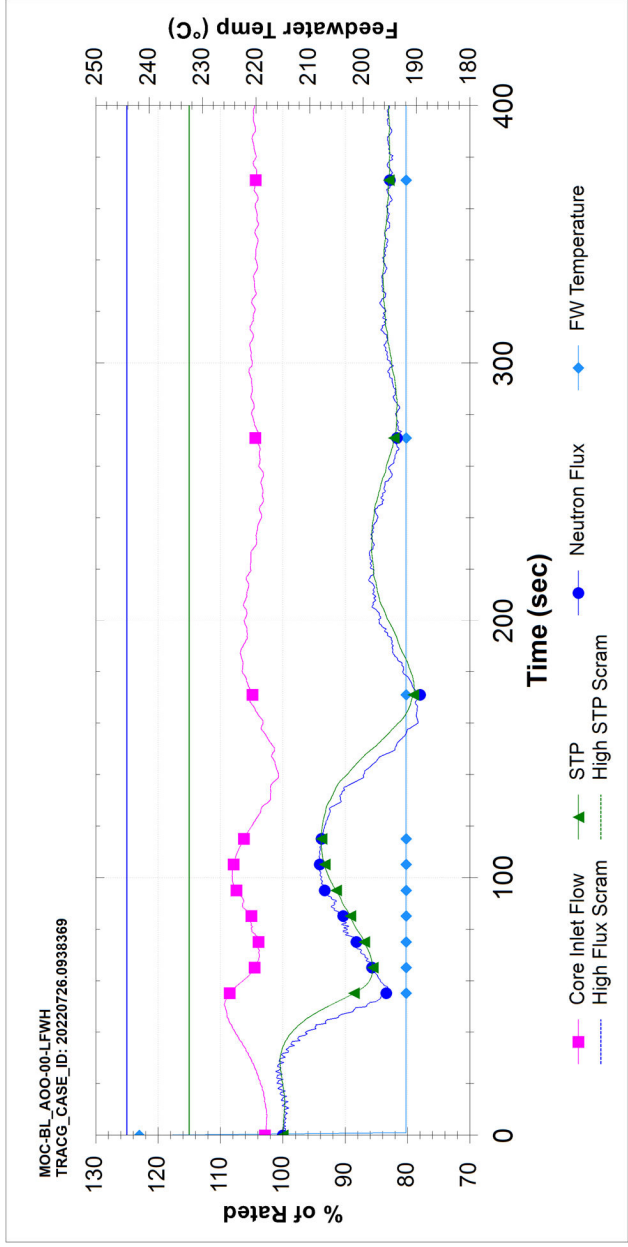


Figure 15.5-5: Loss of Feedwater Heating (AOO)

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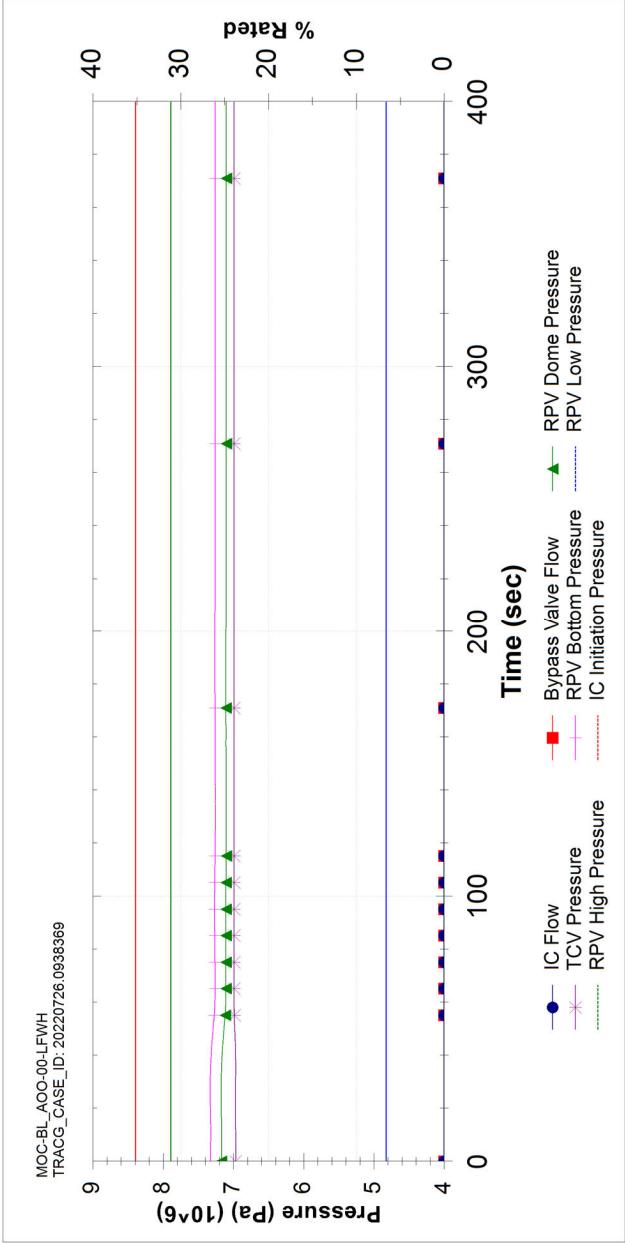


Figure 15.5-6: Loss of Feedwater Heating (AOO)

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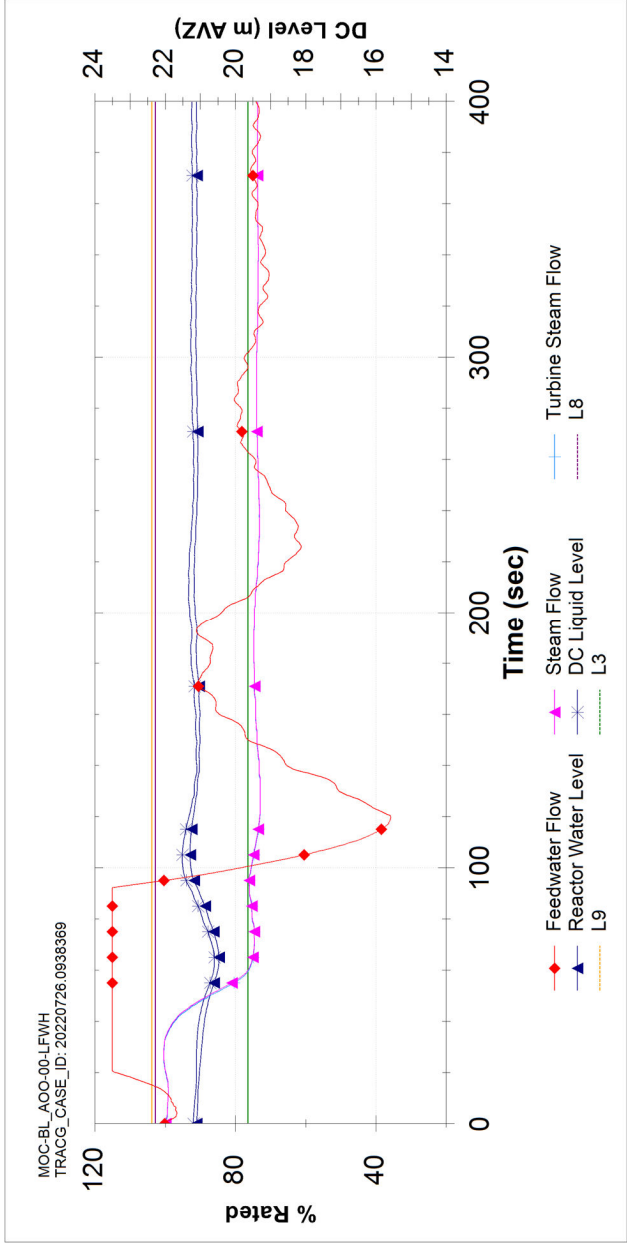


Figure 15.5-7: Loss of Feedwater Heating (AOO)

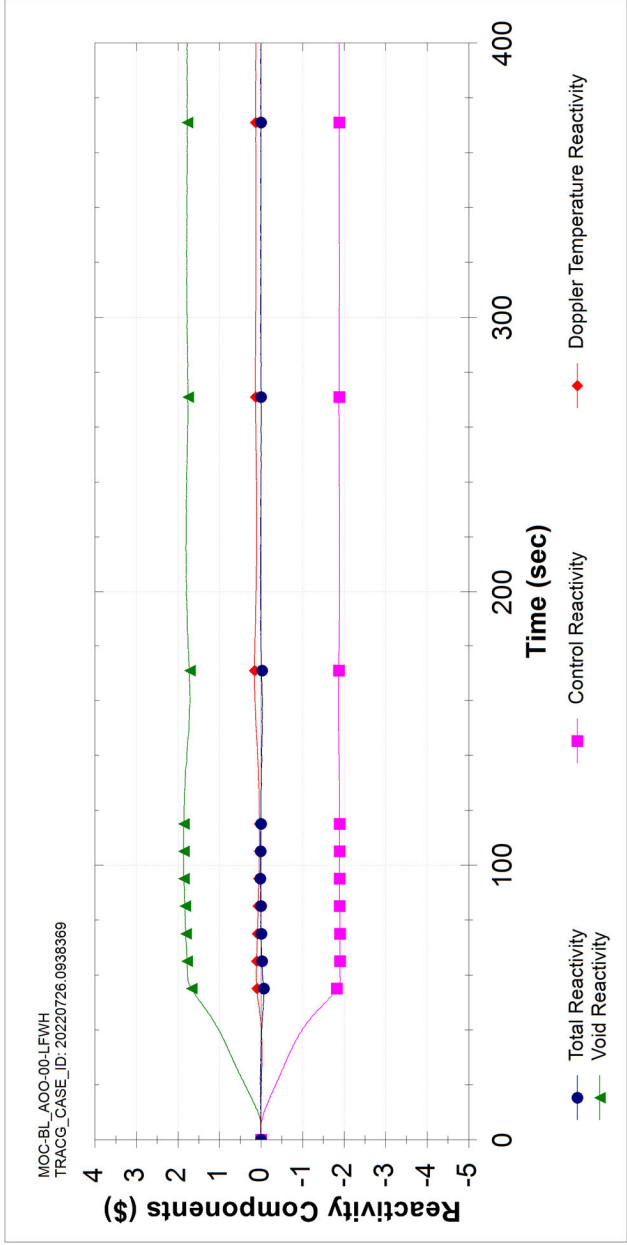


Figure 15.5-8: Loss of Feedwater Heating (AOO)

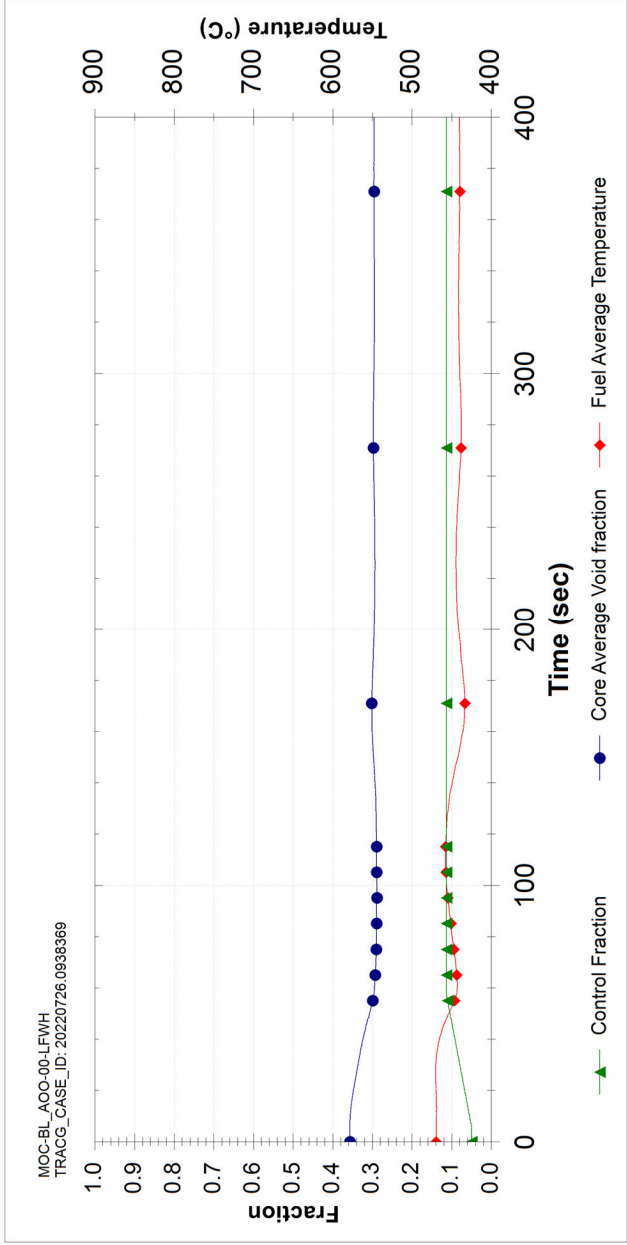


Figure 15.5-9: Loss of Feedwater Heating (AOO)

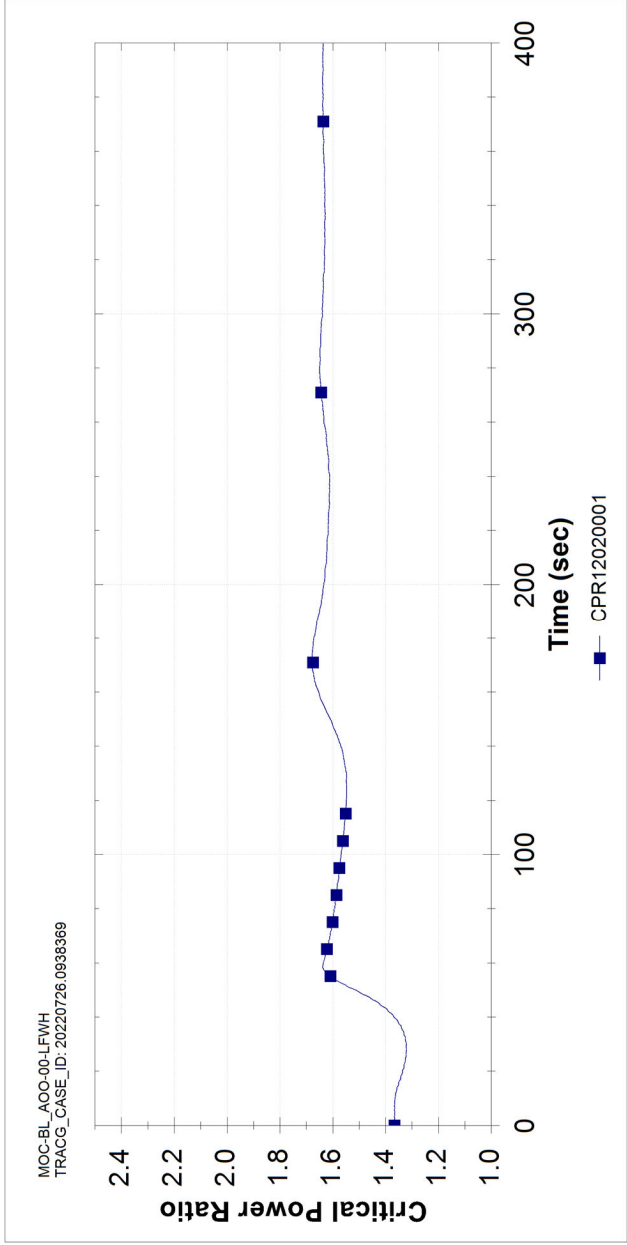


Figure 15.5-10: Loss of Feedwater Heating (AOO)

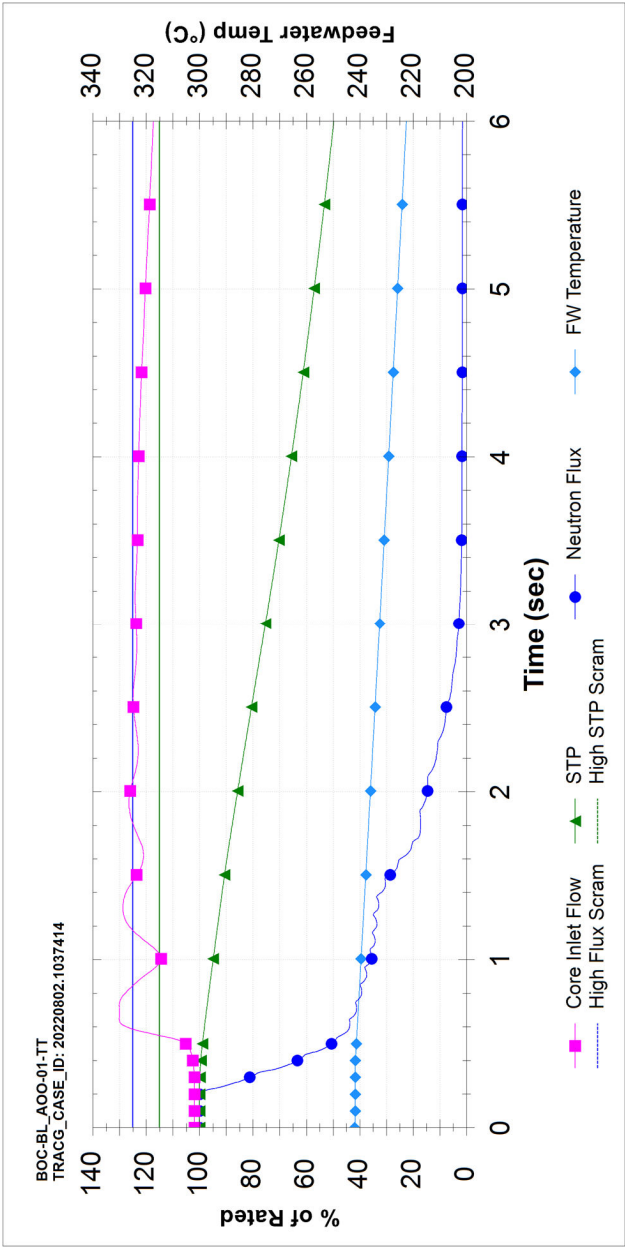


Figure 15.5-11: Turbine Trip (AOO)

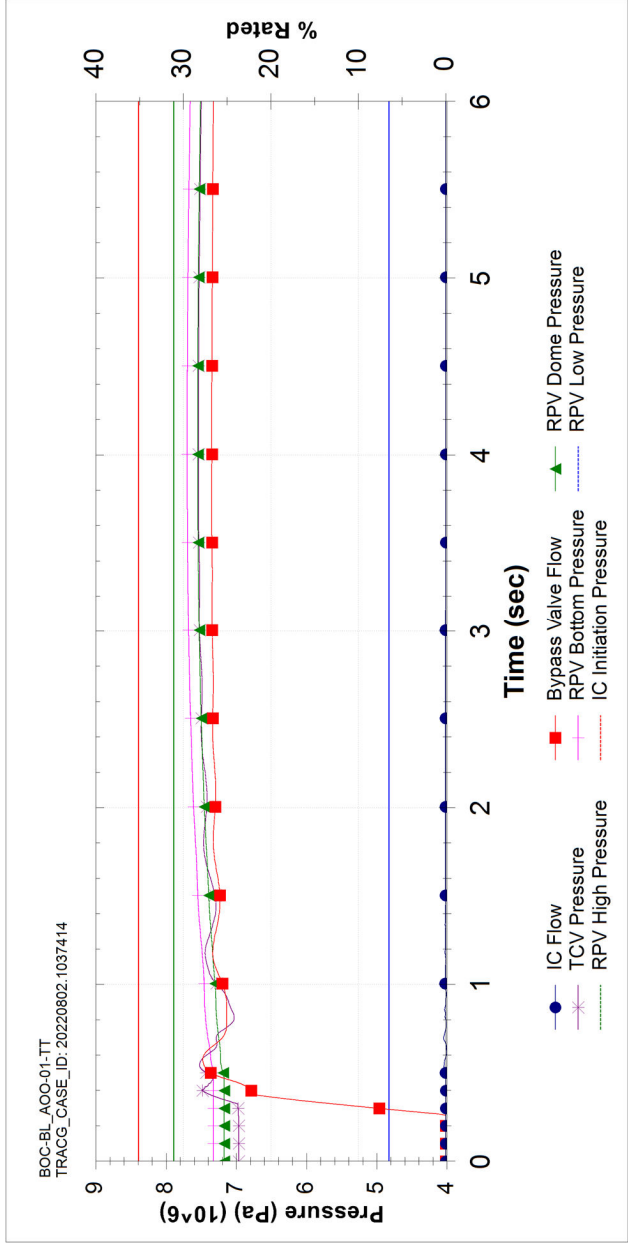


Figure 15.5-12: Turbine Trip (AOO)

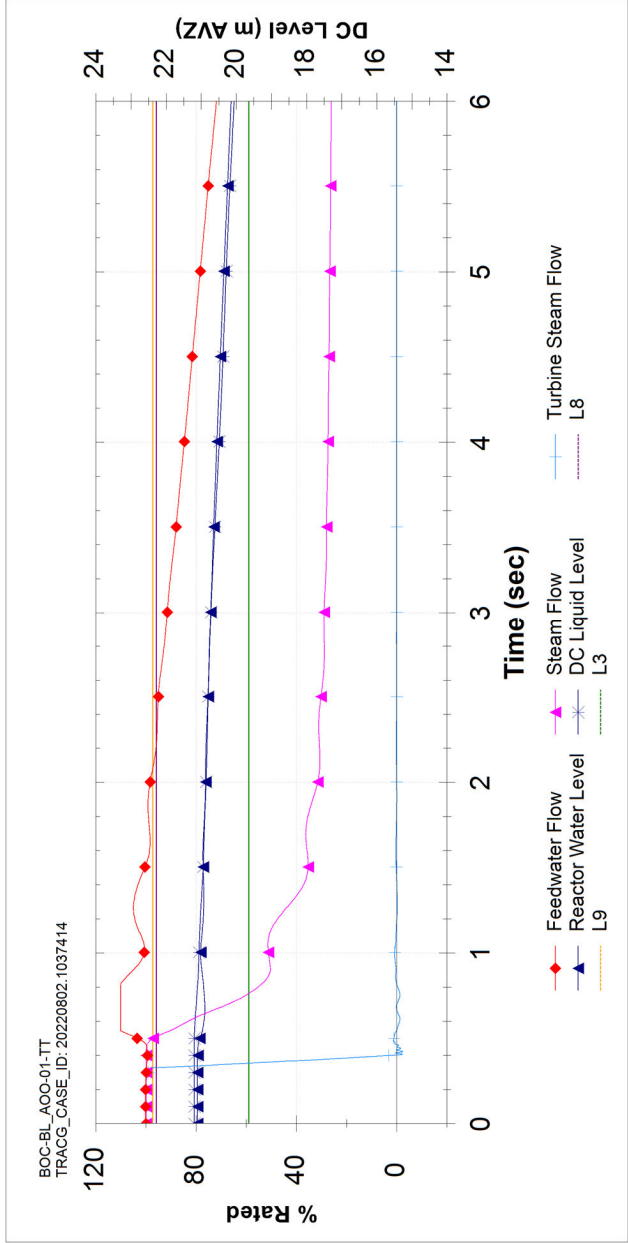


Figure 15.5-13: Turbine Trip (AOG)

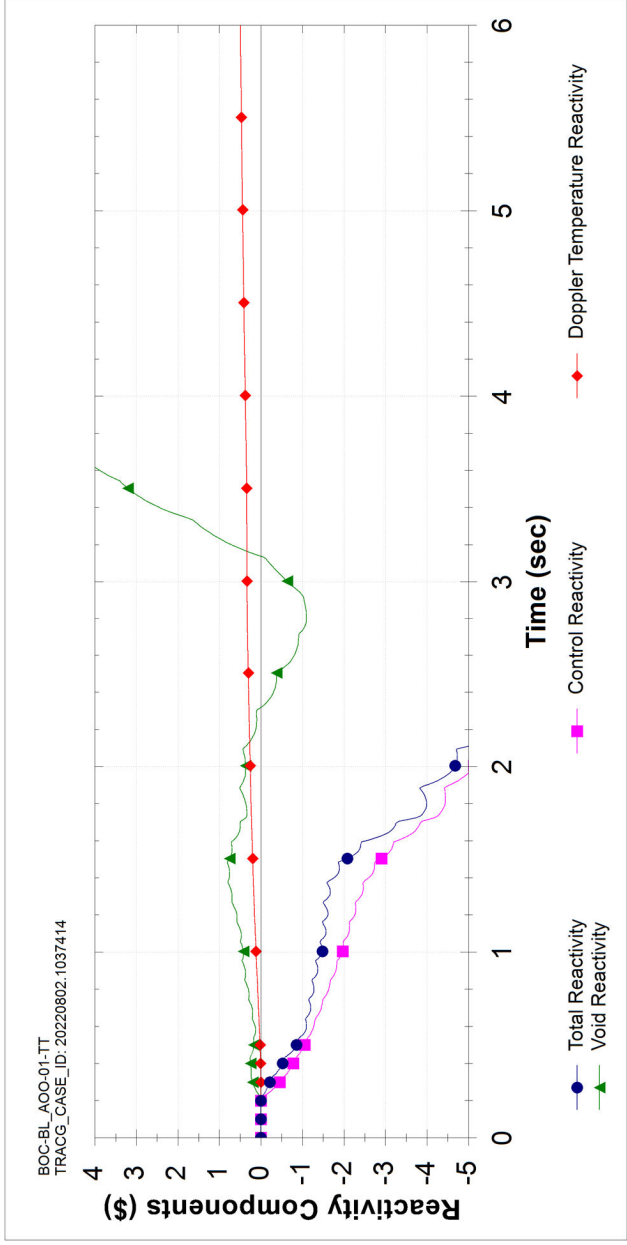


Figure 15.5-14: Turbine Trip (AOO)

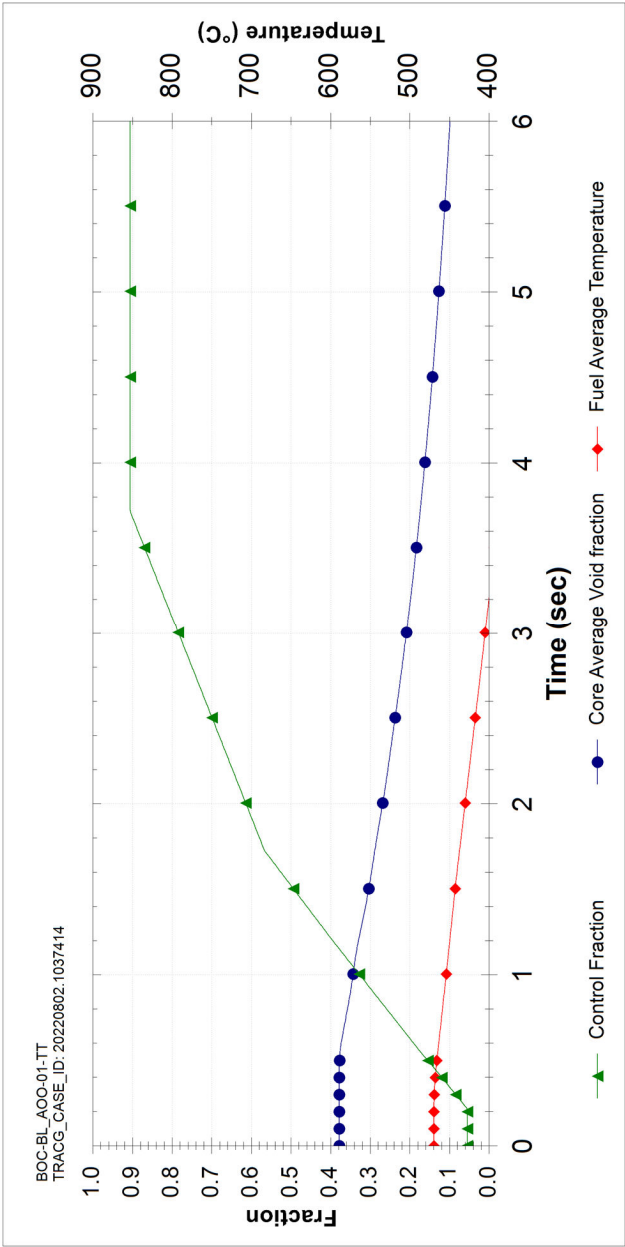


Figure 15.5-15: Turbine Trip (AOG)

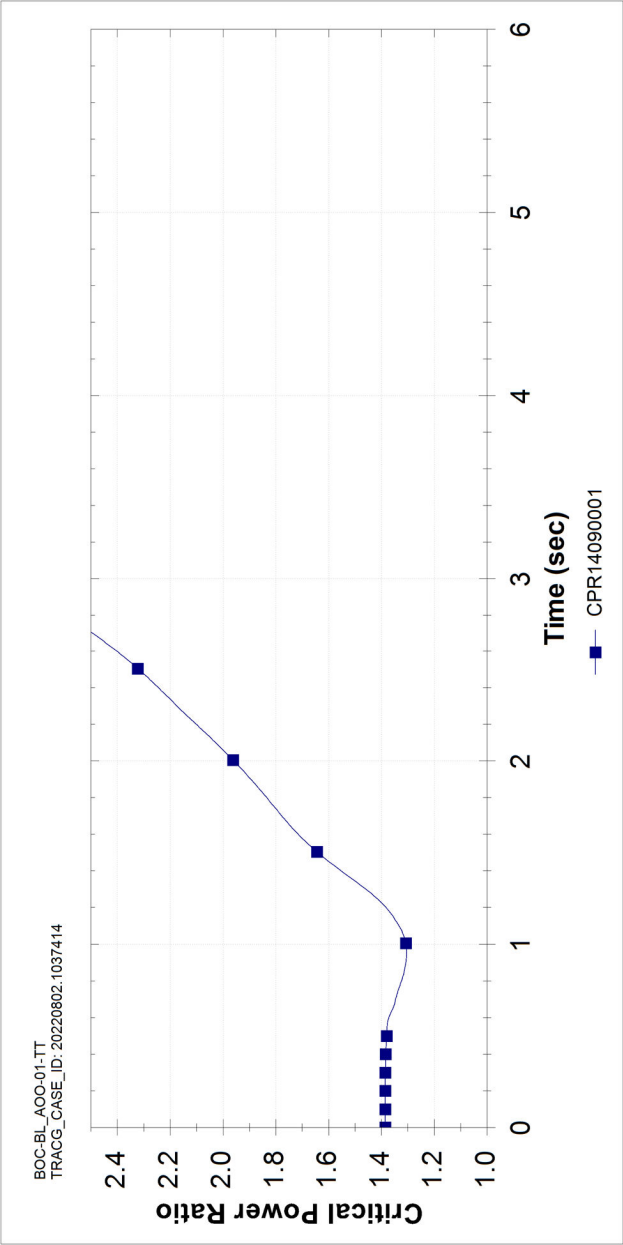


Figure 15.5-16: Turbine Trip (AOG)

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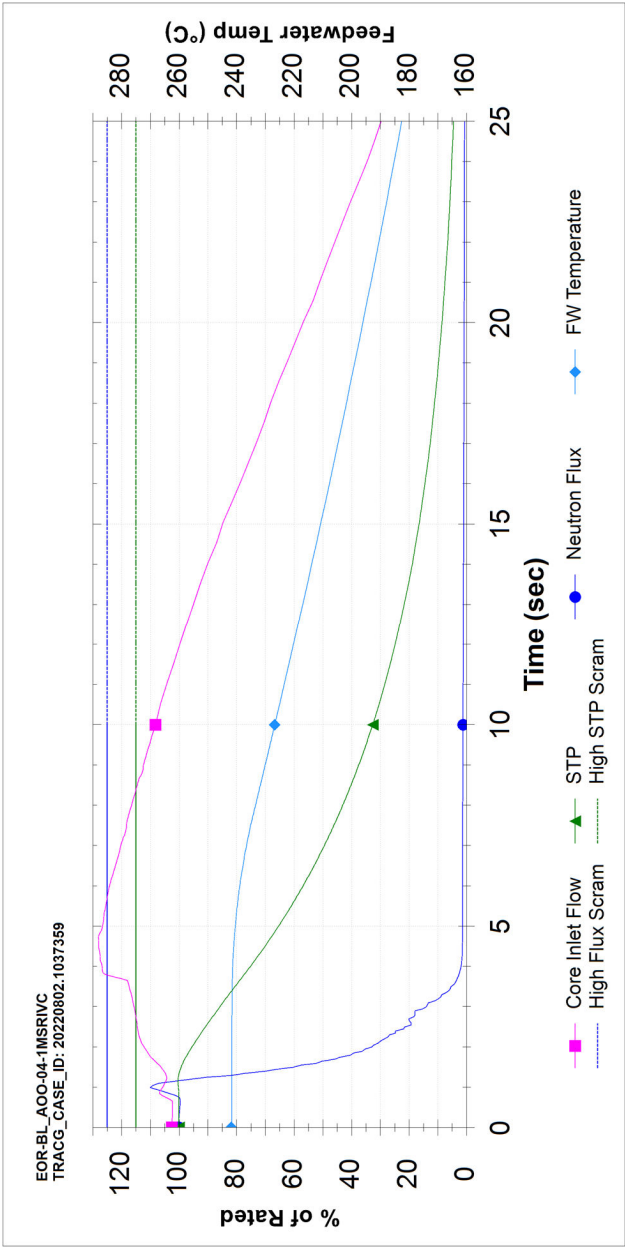


Figure 15.5-17: Closure of One Main Steam Reactor Isolation Valve (AOO)

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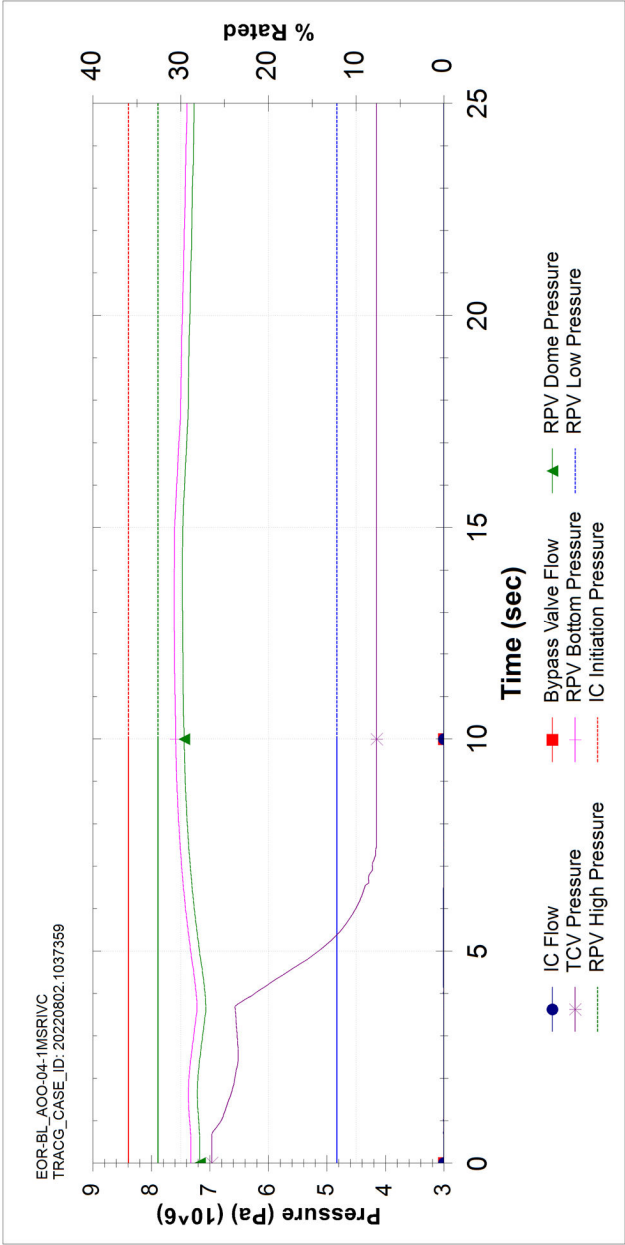


Figure 15.5-18: Closure of One Main Steam Reactor Isolation Valve (AOO)

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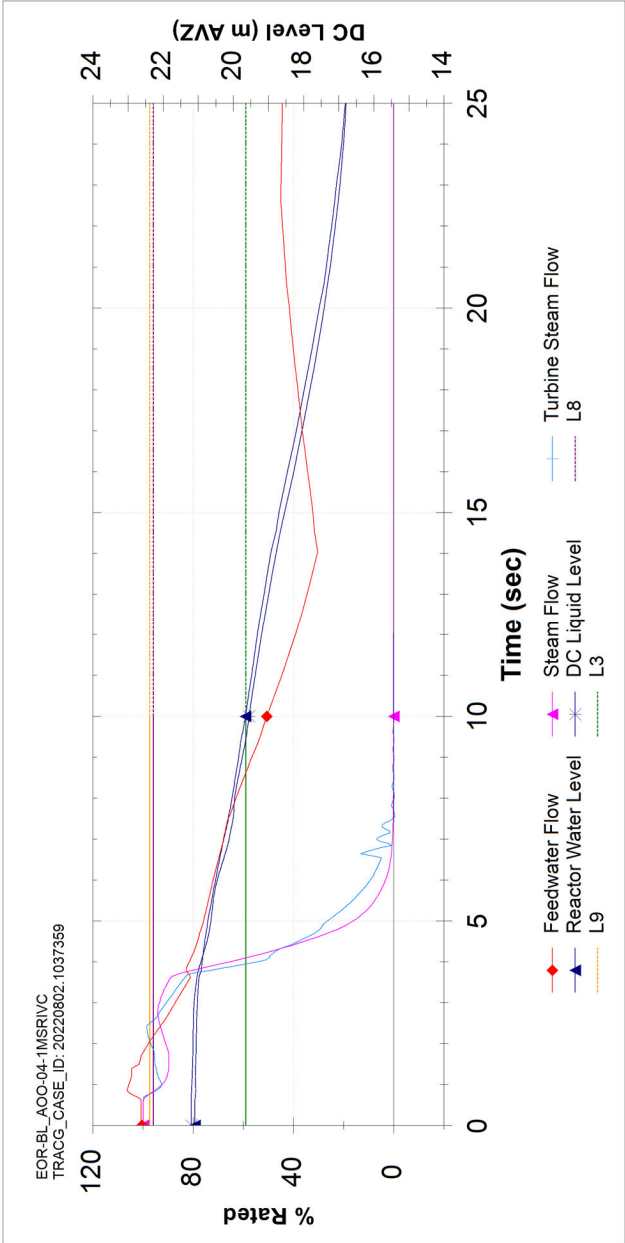


Figure 15.5-19: Closure of One Main Steam Reactor Isolation Valve (AOO)

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NON-PROPRIETARY INFORMATION

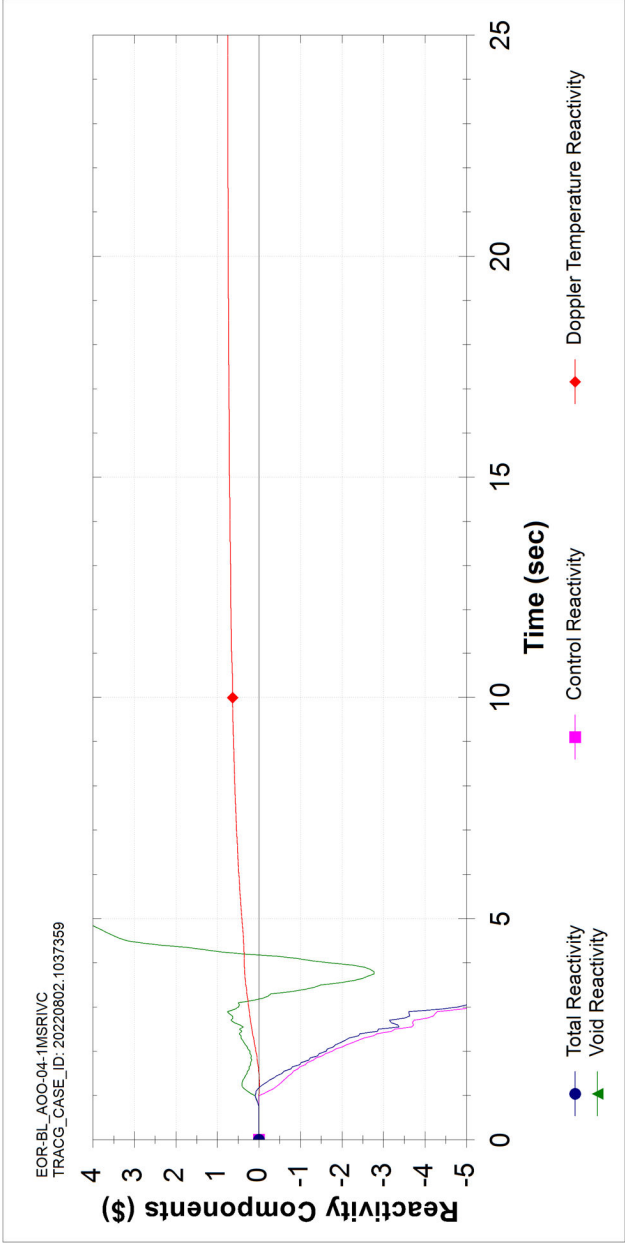


Figure 15.5-20: Closure of One Main Steam Reactor Isolation Valve (AOO)

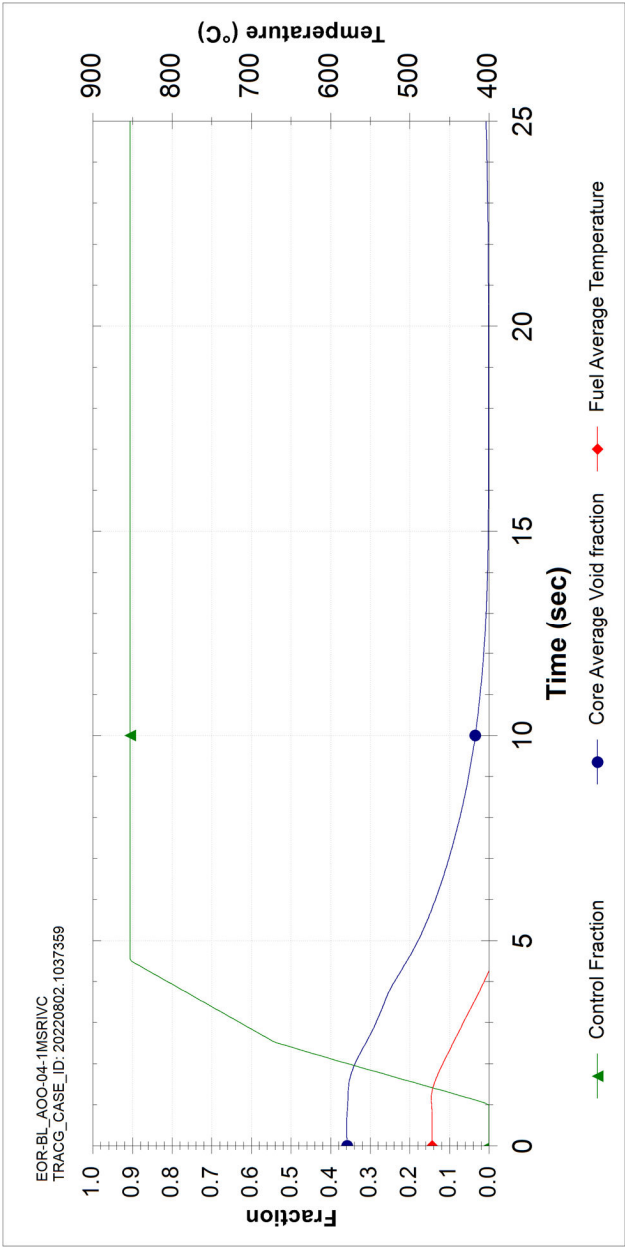


Figure 15.5-21: Closure of One Main Steam Reactor Isolation Valve (AOO)

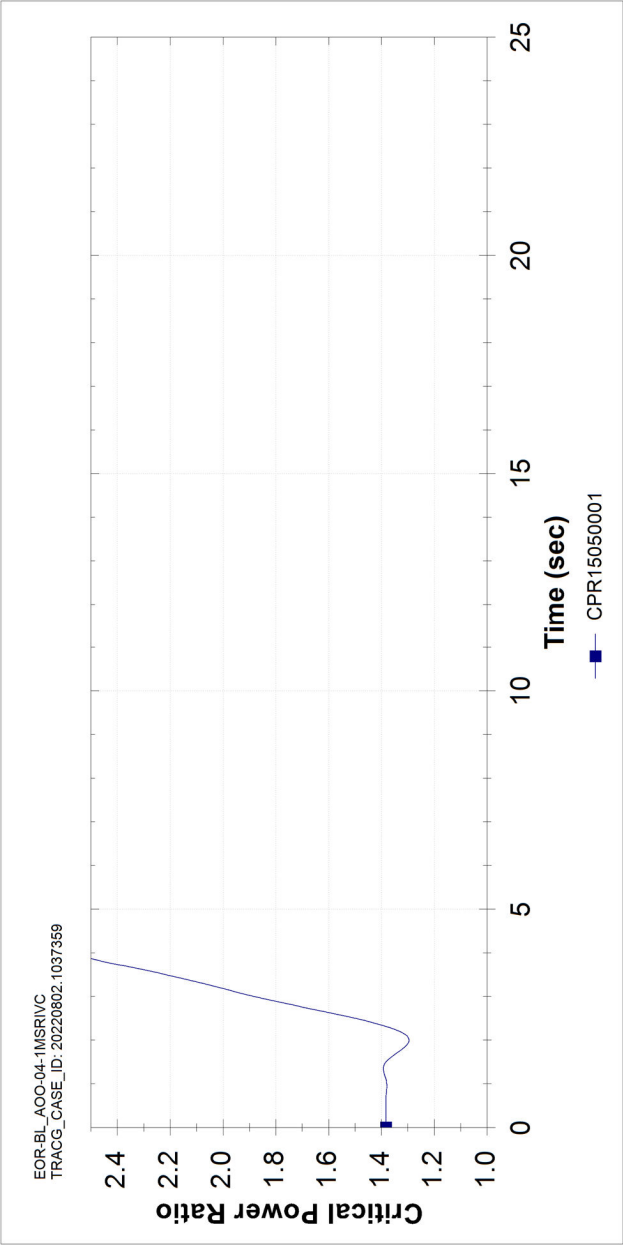


Figure 15.5-22: Closure of One Main Steam Reactor Isolation Valve (AOO)

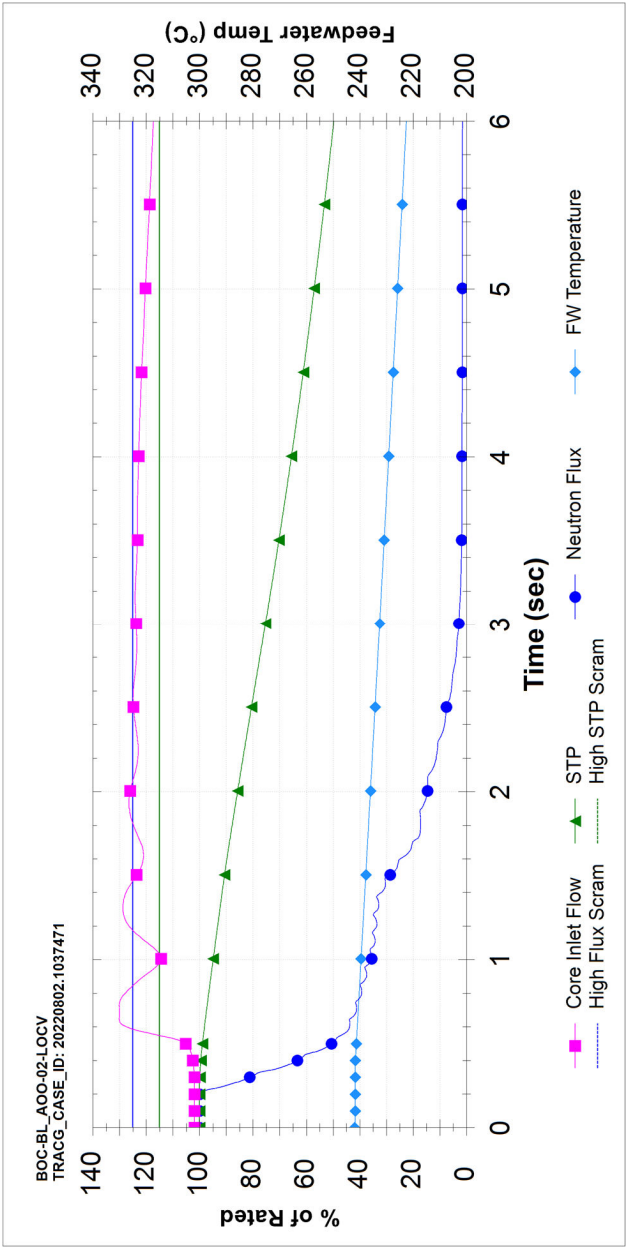


Figure 15.5-23: Loss of Condenser Vacuum (AOO)

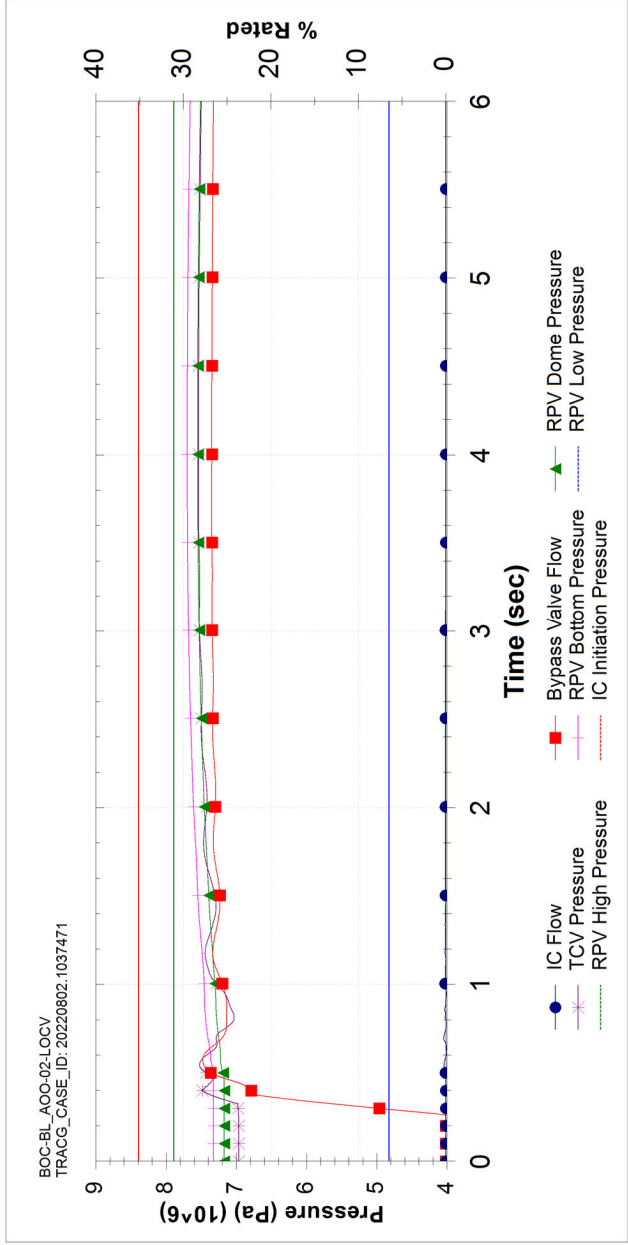


Figure 15.5-23a: Loss of Condenser Vacuum (AOO)

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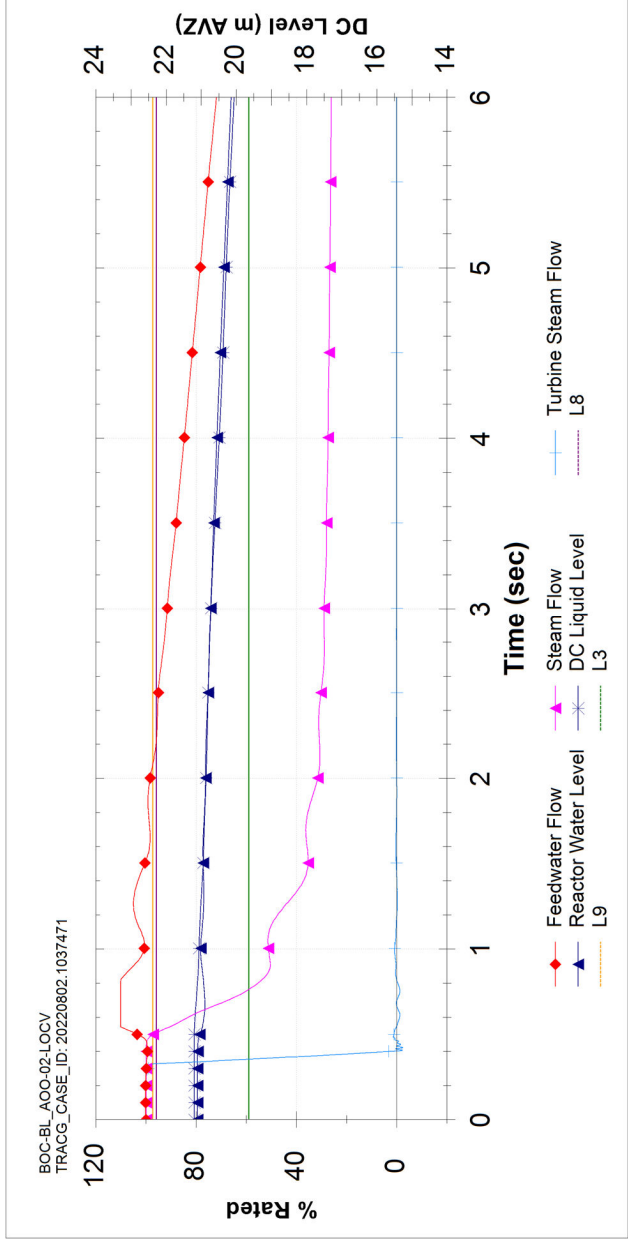


Figure 15.5-24: Loss of Condenser Vacuum (AOO)

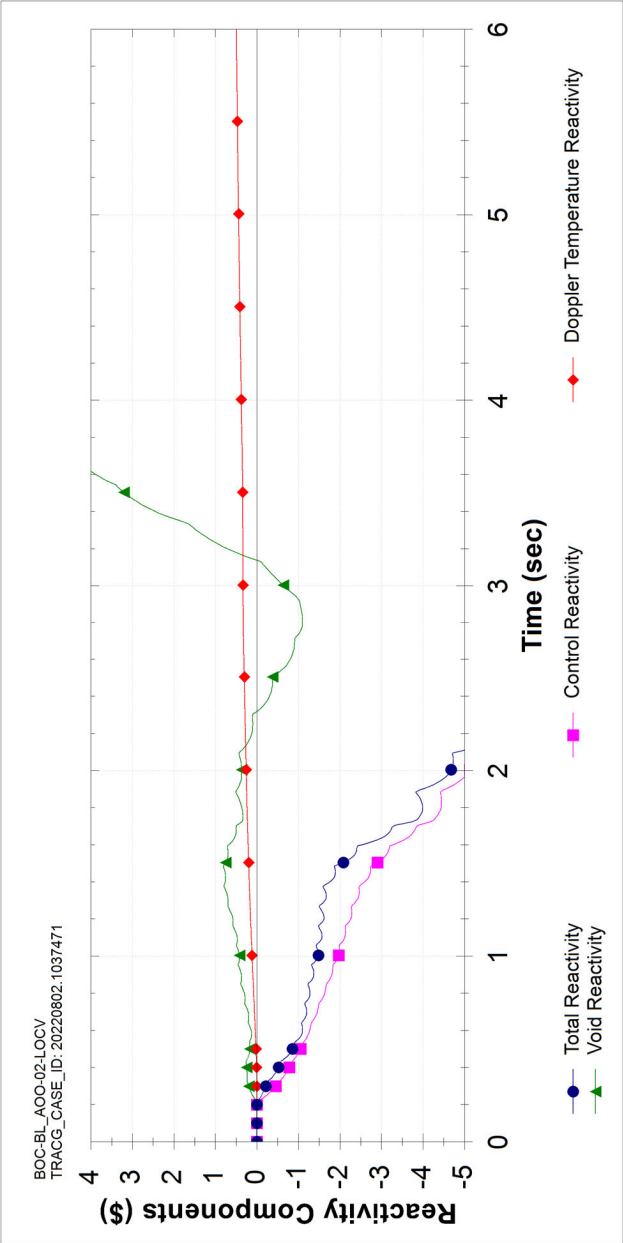


Figure 15.5-25: Loss of Condenser Vacuum (AOO)

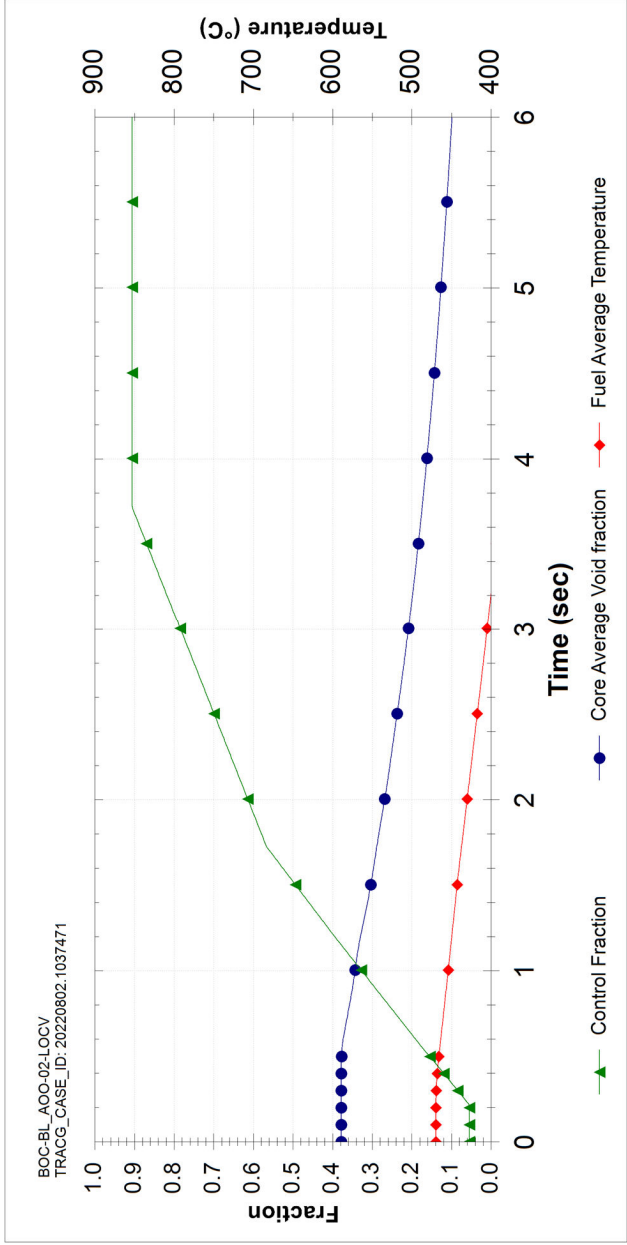


Figure 15.5-26: Loss of Condenser Vacuum (AOO)

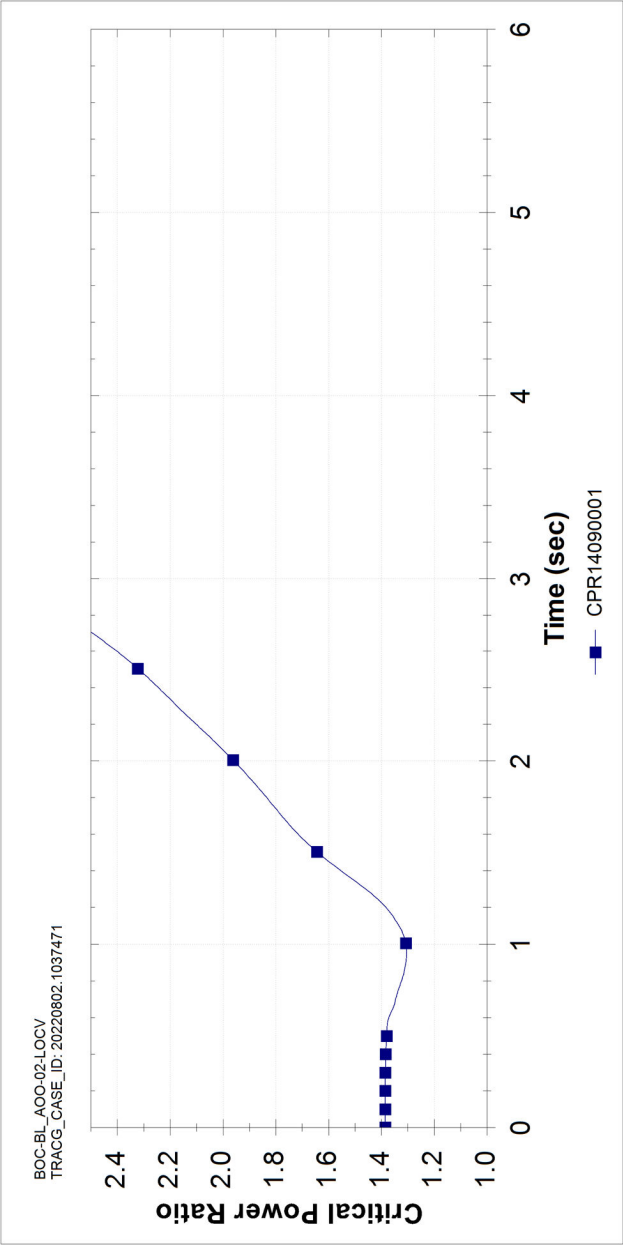


Figure 15.5-27: Loss of Condenser Vacuum (AOO)

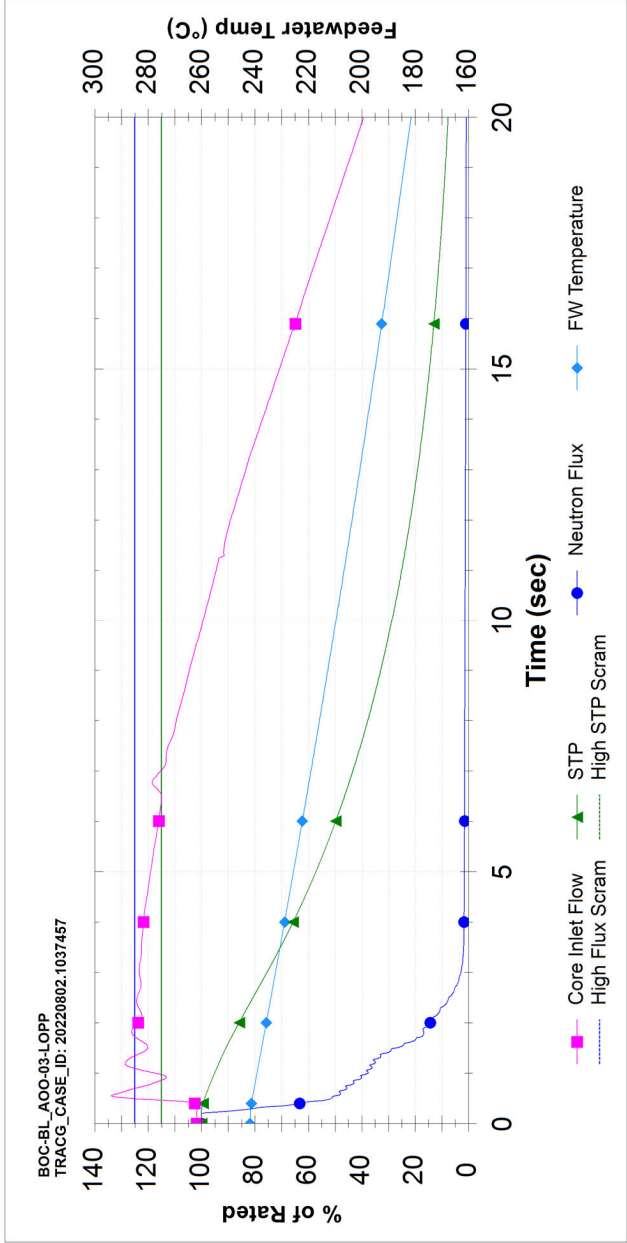


Figure 15.5-28: Loss of Preferred Power (AOO)

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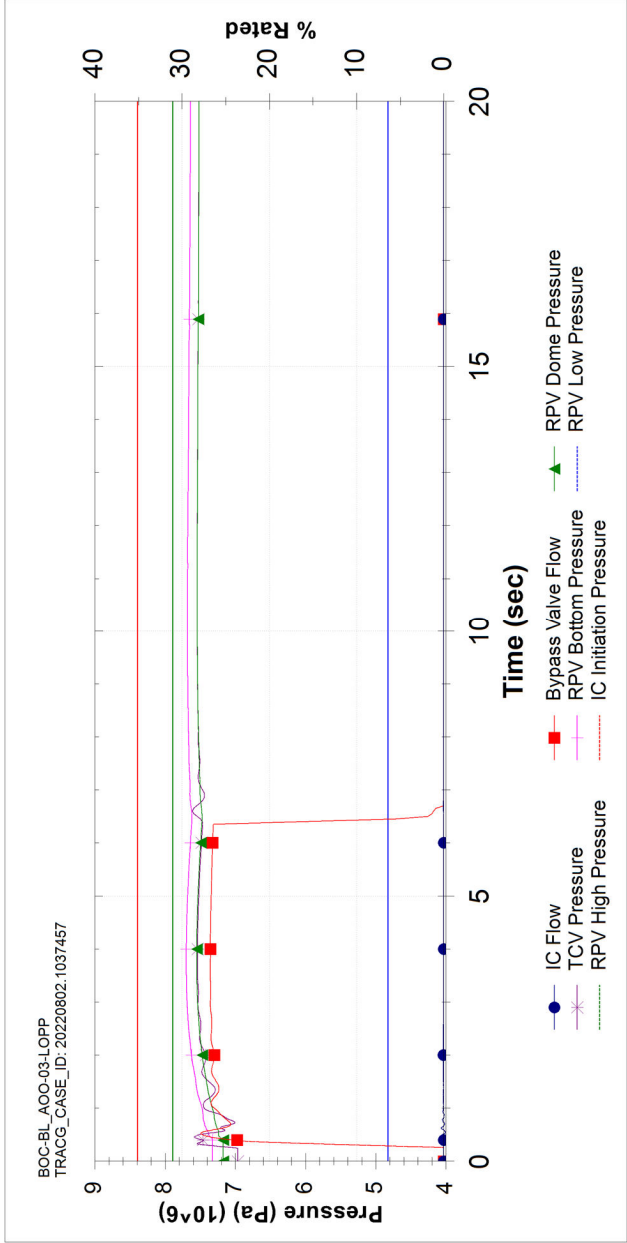


Figure 15.5-29: Loss of Preferred Power (AOO)

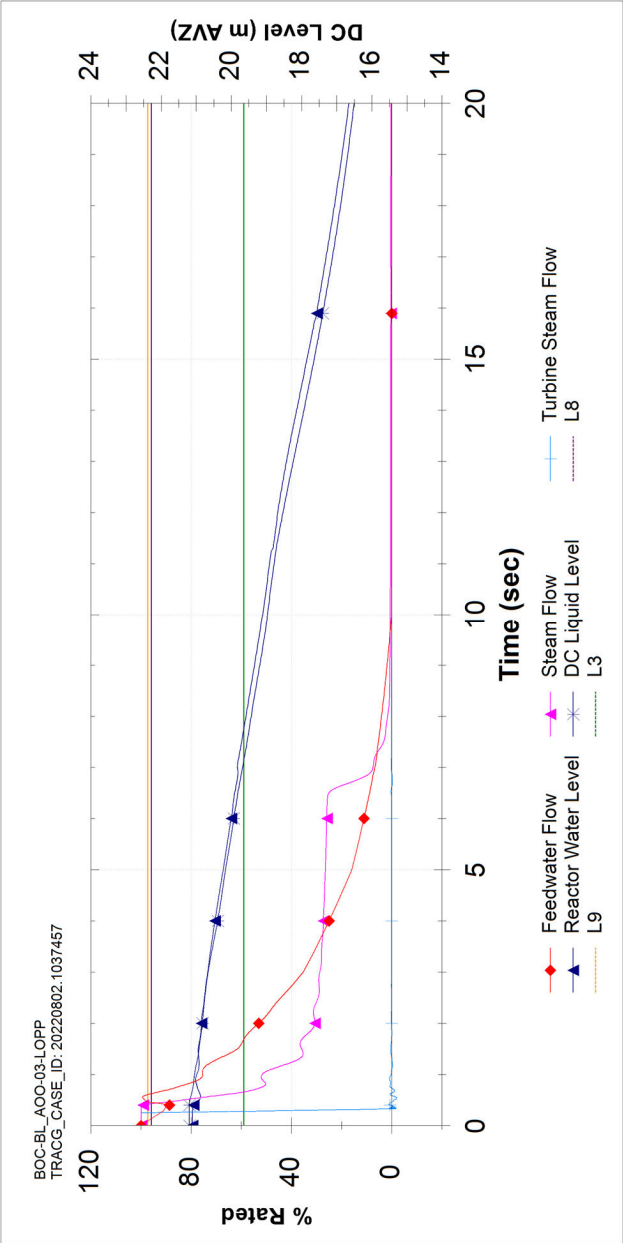


Figure 15.5-30: Loss of Preferred Power (AOO)

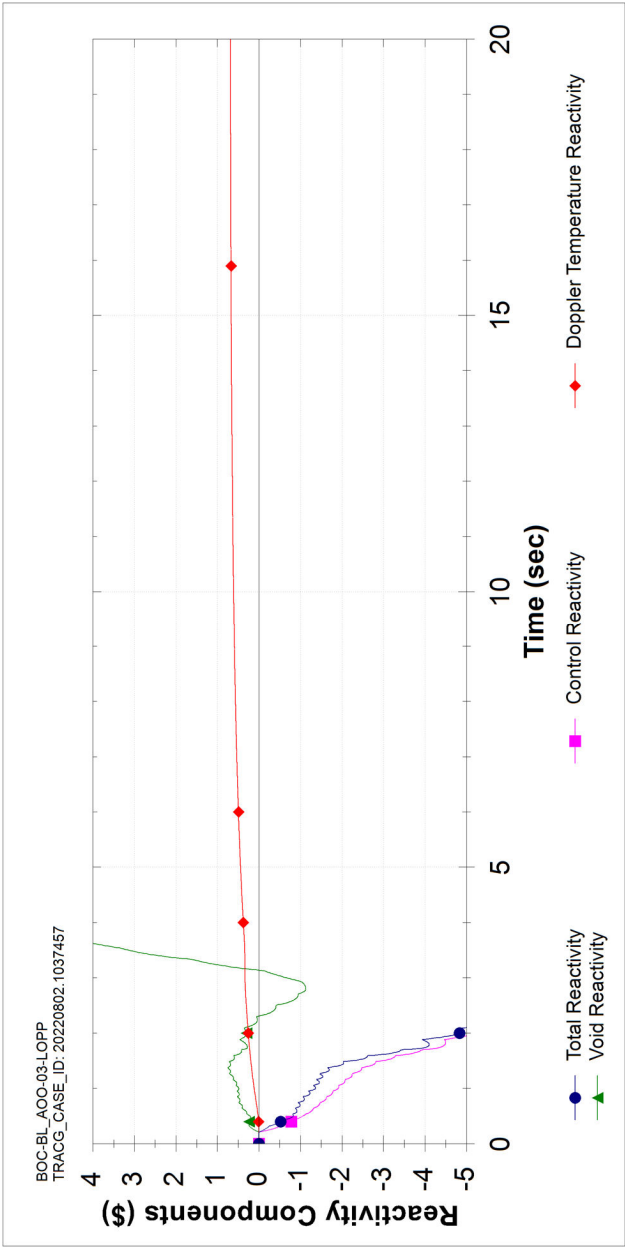


Figure 15.5-31: Loss of Preferred Power (AOO)

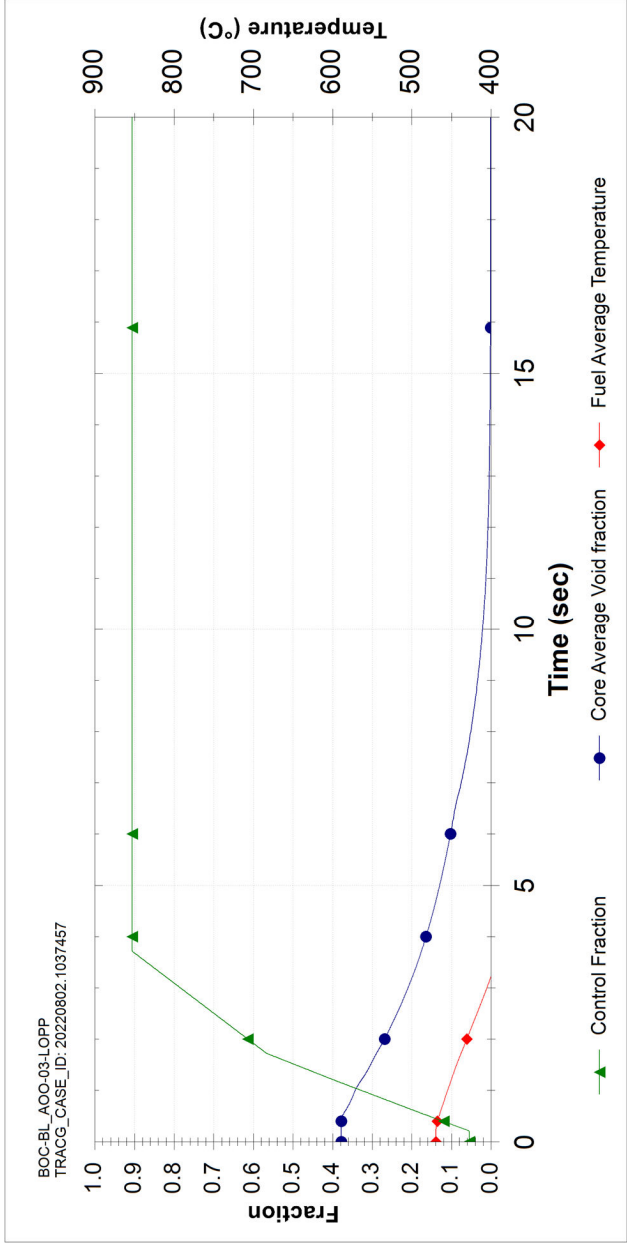


Figure 15.5-32: Loss of Preferred Power (AOG)

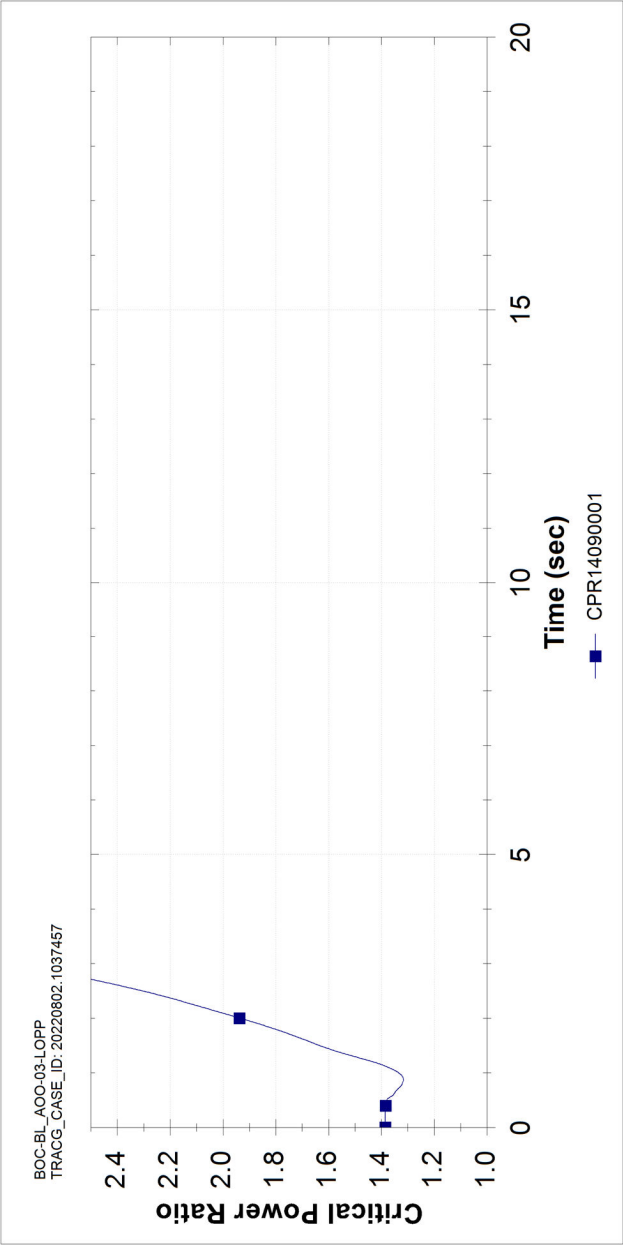


Figure 15.5-33: Loss of Preferred Power (AOG)

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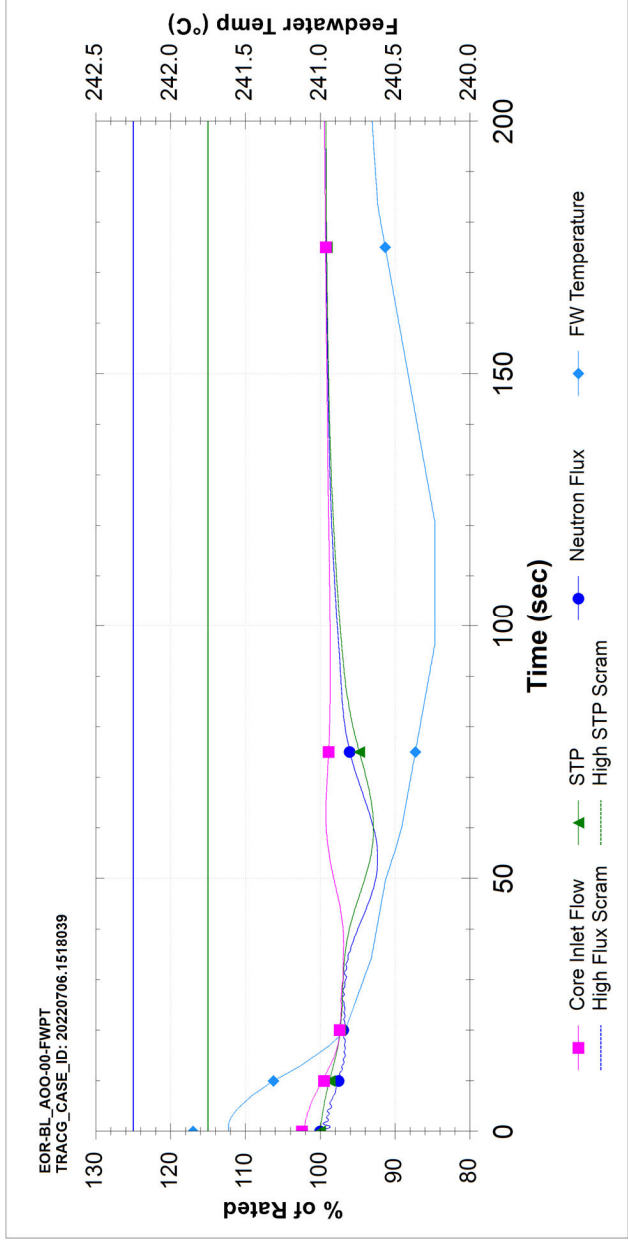


Figure 15.5-34: Feedwater Pump Trip – One Pump (AOO)

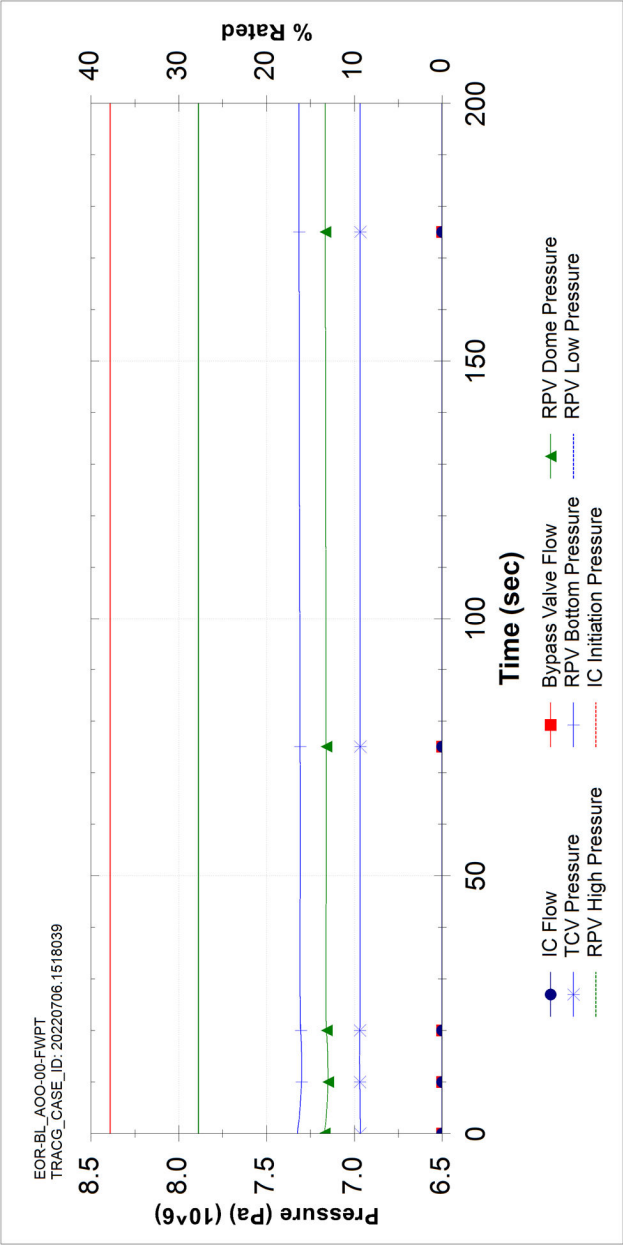


Figure 15.5-35: Feedwater Pump Trip – One Pump (AOO)

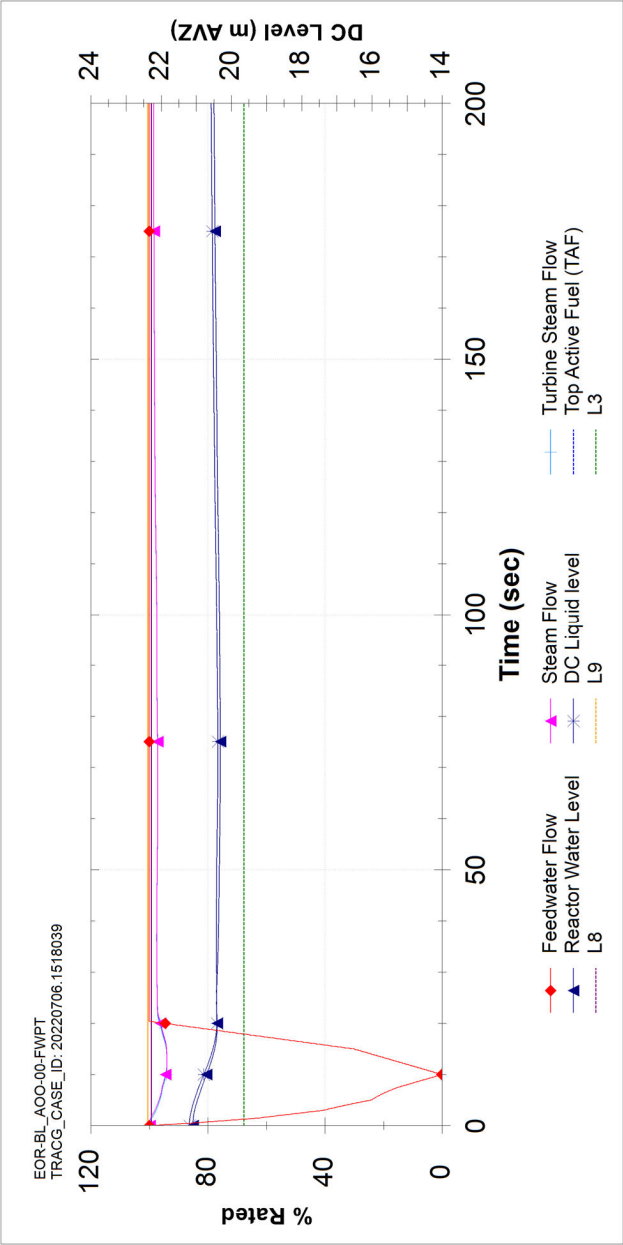


Figure 15.5-36: Feedwater Pump Trip – One Pump (AOG)

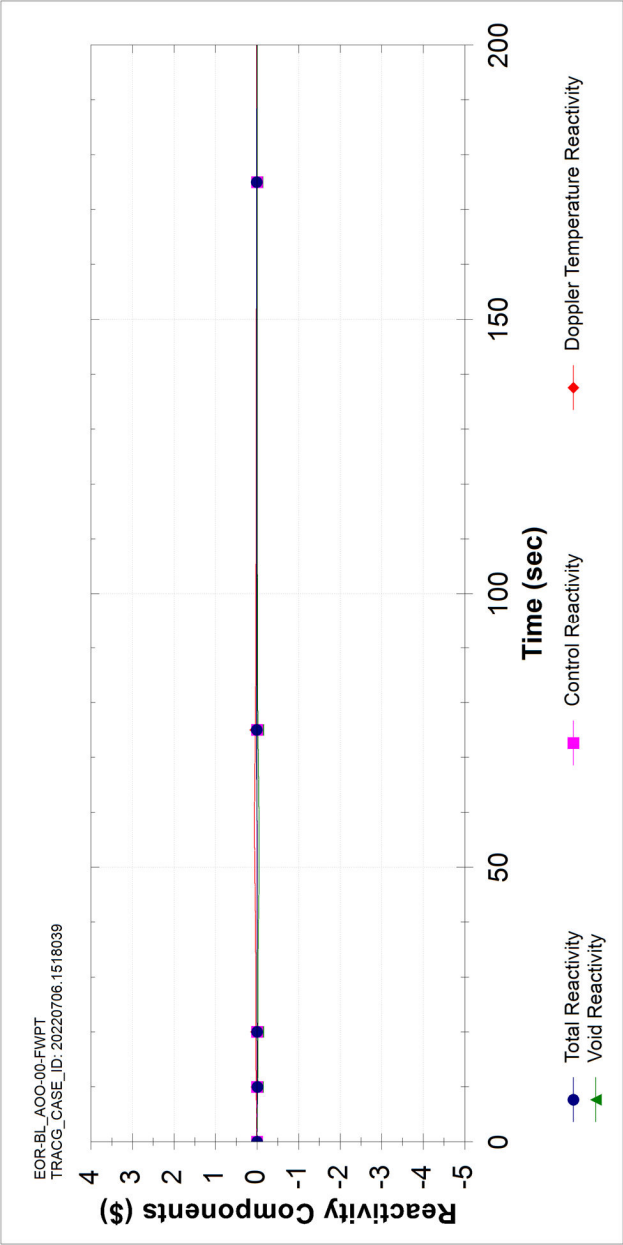


Figure 15.5-37: Feedwater Pump Trip – One Pump (AOO)

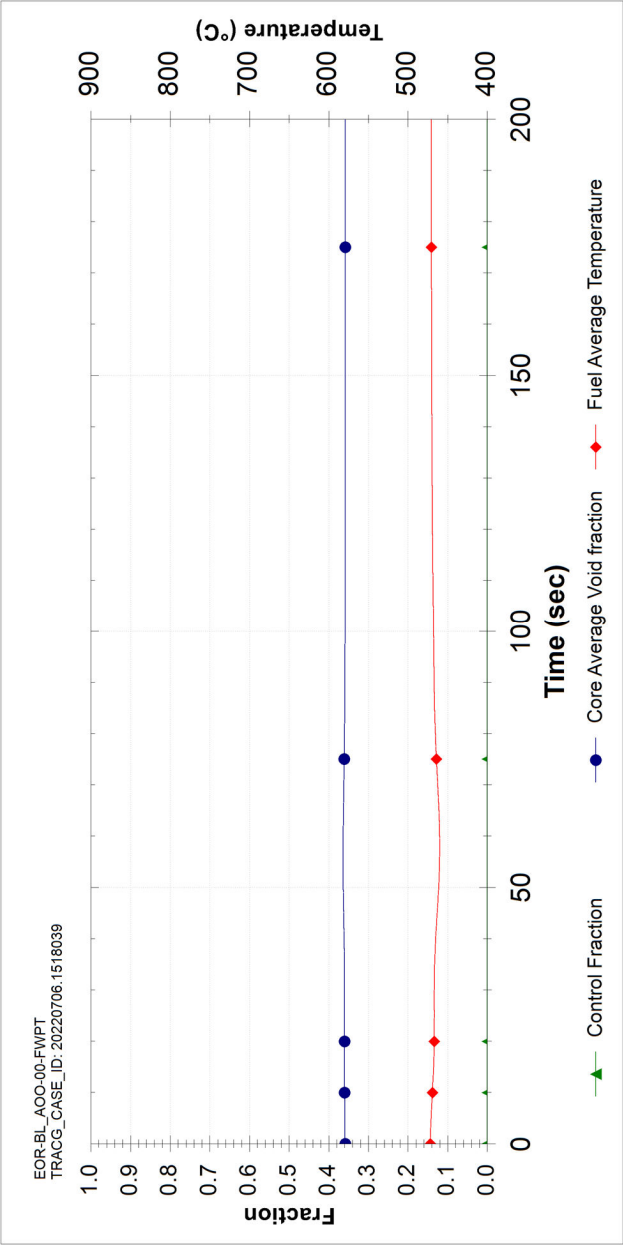


Figure 15.5-38: Feedwater Pump Trip – One Pump (AOG)

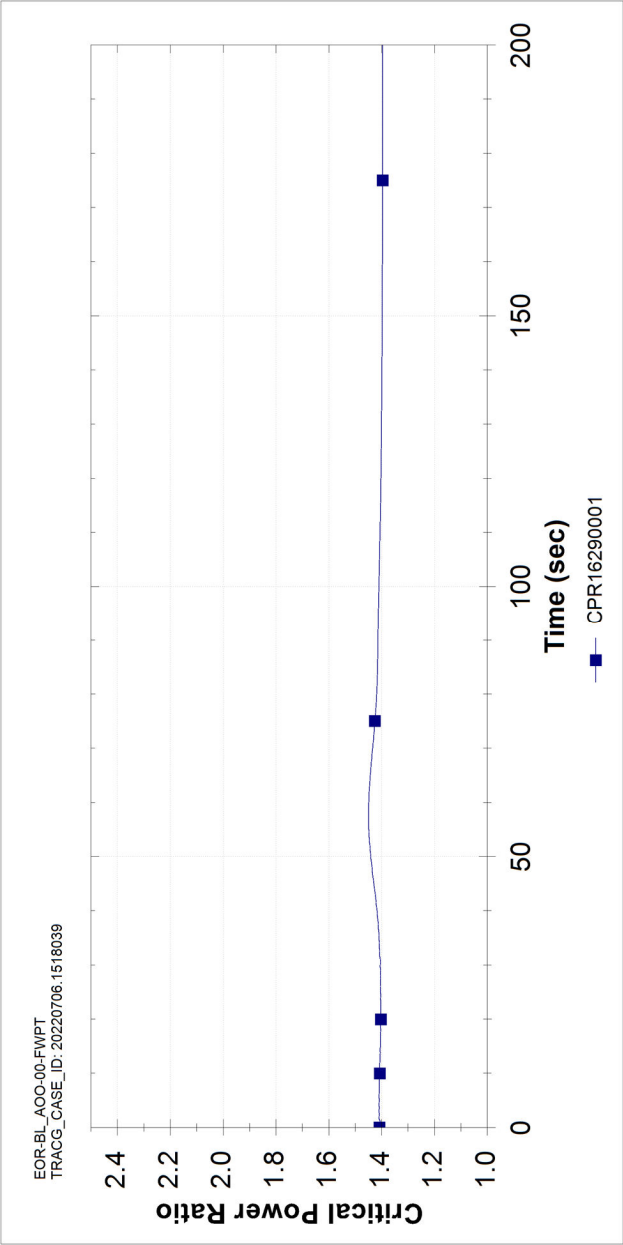


Figure 15.5-39: Feedwater Pump Trip – One Pump (AOG)

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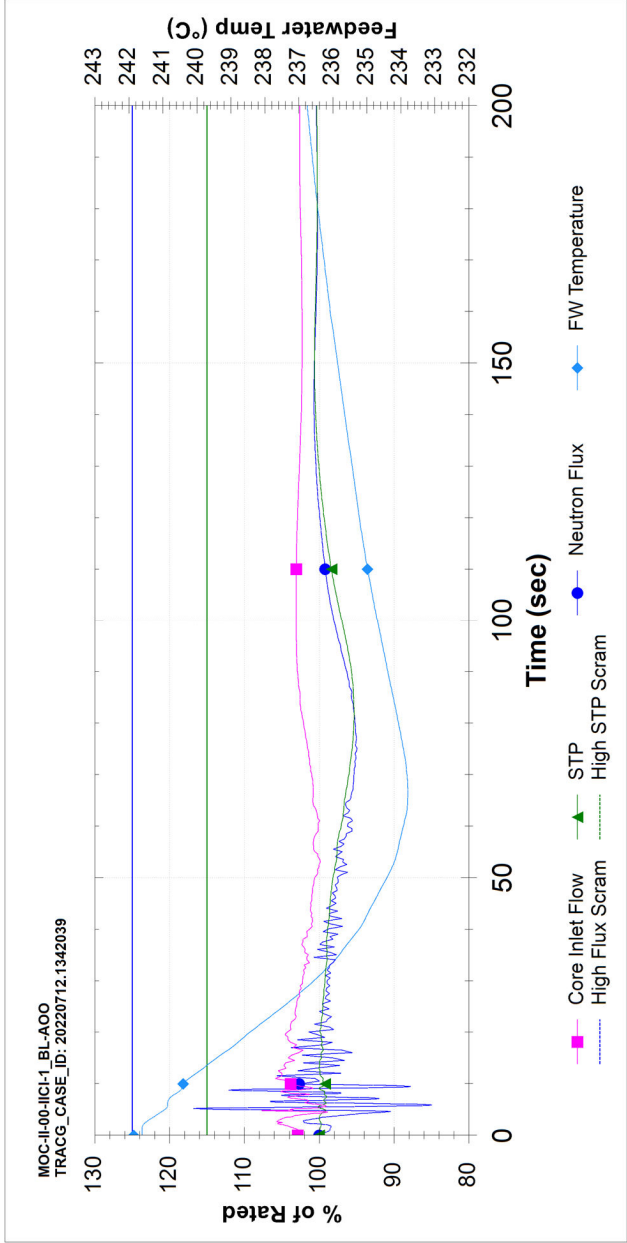


Figure 15.5-40: Inadvertent Isolation Condenser Initiation – One Train (AOO)

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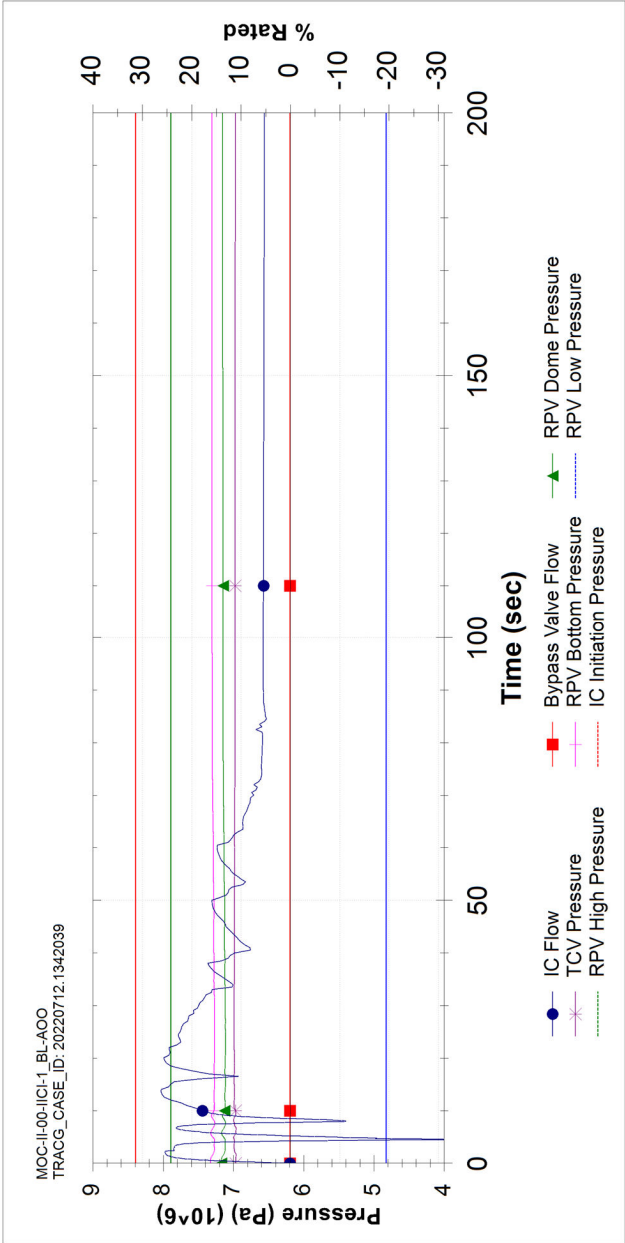


Figure 15.5-41: Inadvertent Isolation Condenser Initiation – One Train (AOO)

NEDO-33965 REVISION 0
NON-PROPRIETARY INFORMATION

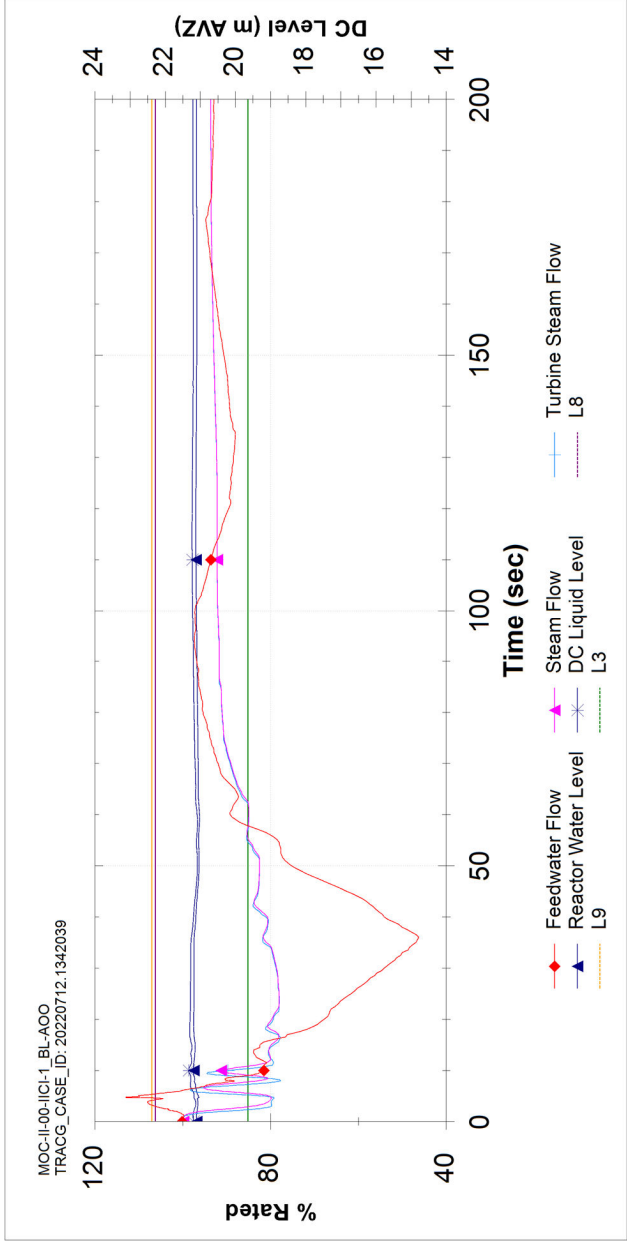


Figure 15.5-42: Inadvertent Isolation Condenser Initiation – One Train (AOO)

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NON-PROPRIETARY INFORMATION

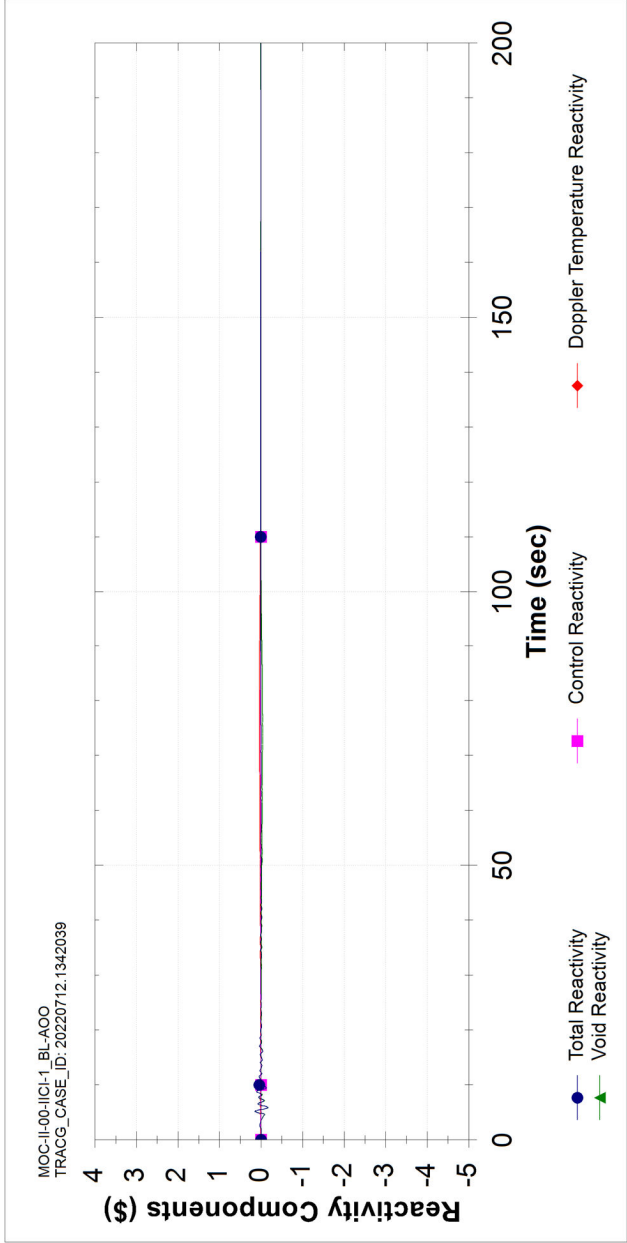


Figure 15.5-43: Inadvertent Isolation Condenser Initiation - One Train (AOO)

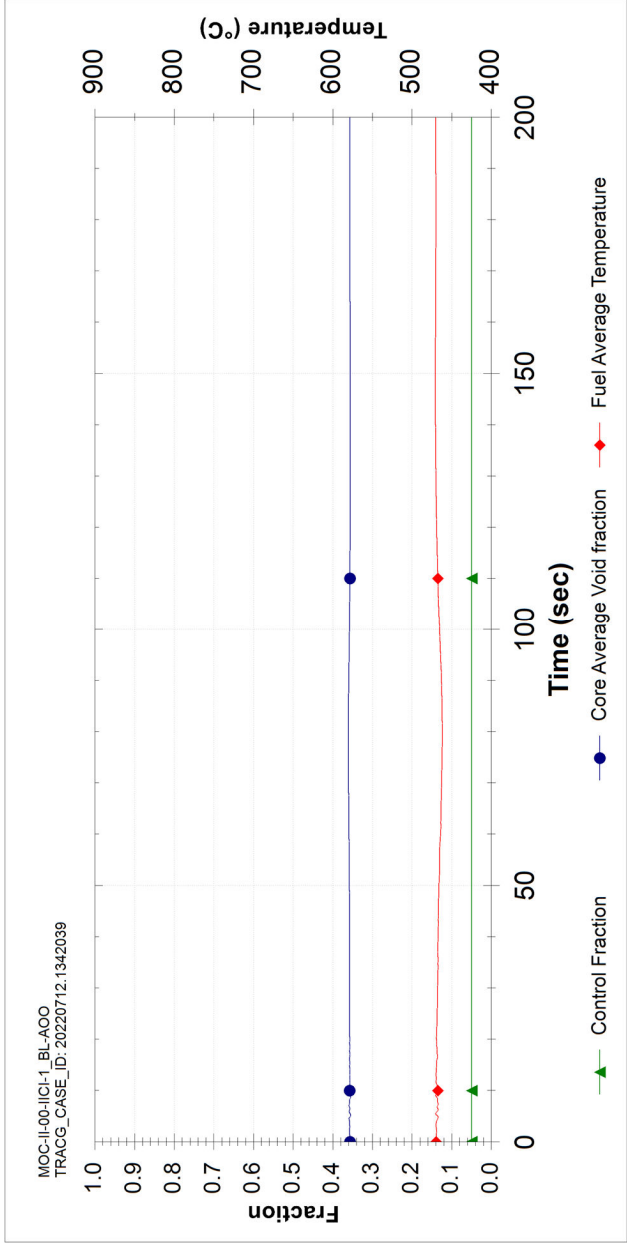


Figure 15.5-44: Inadvertent Isolation Condenser Initiation - One Train (AOO)

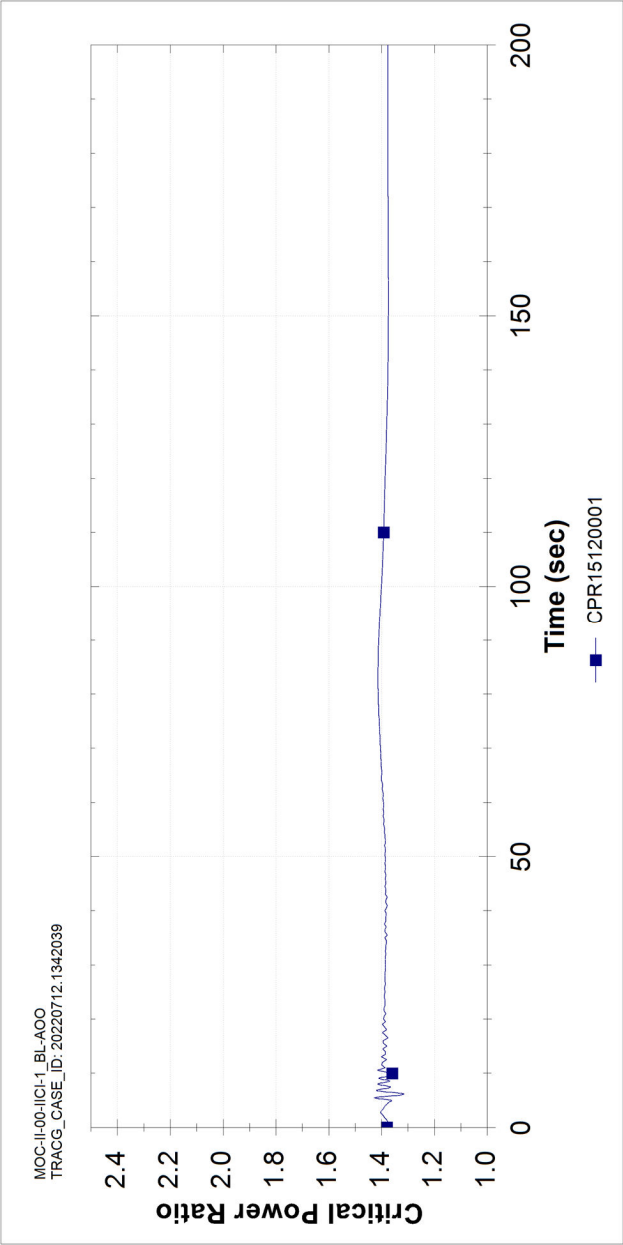


Figure 15.5-45: Inadvertent Isolation Condenser Initiation - One Train (AOO)

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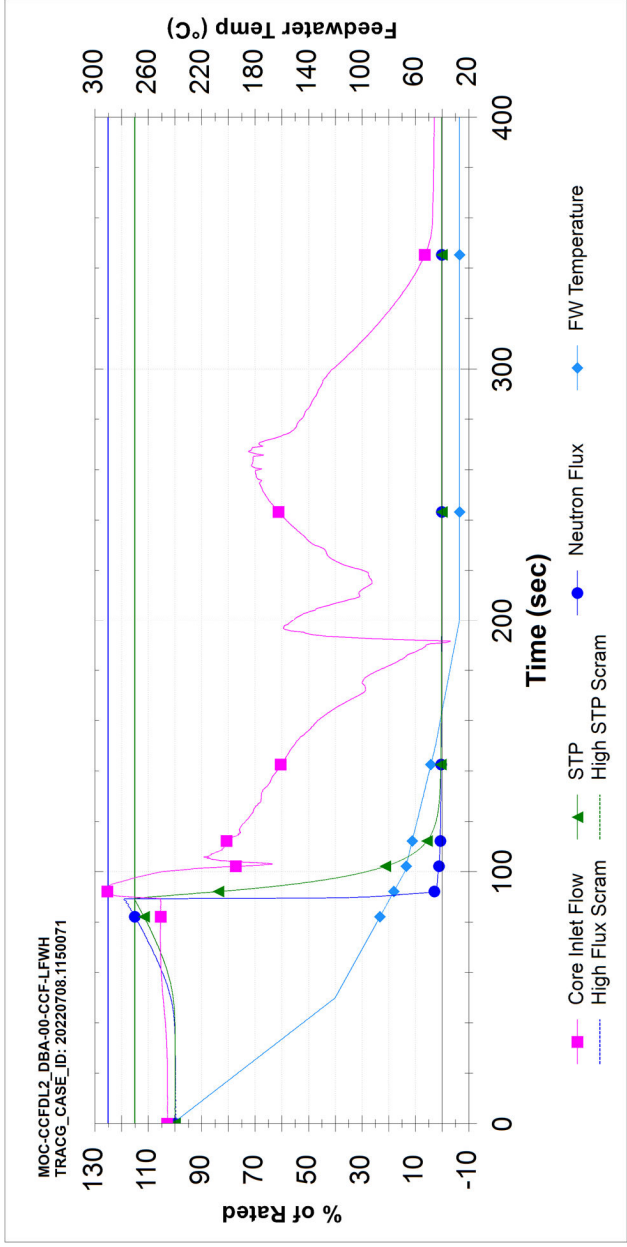


Figure 15.5-46: Loss of Feedwater Heating (DBA)

NEDO-33965 REVISION 0
NON-PROPRIETARY INFORMATION

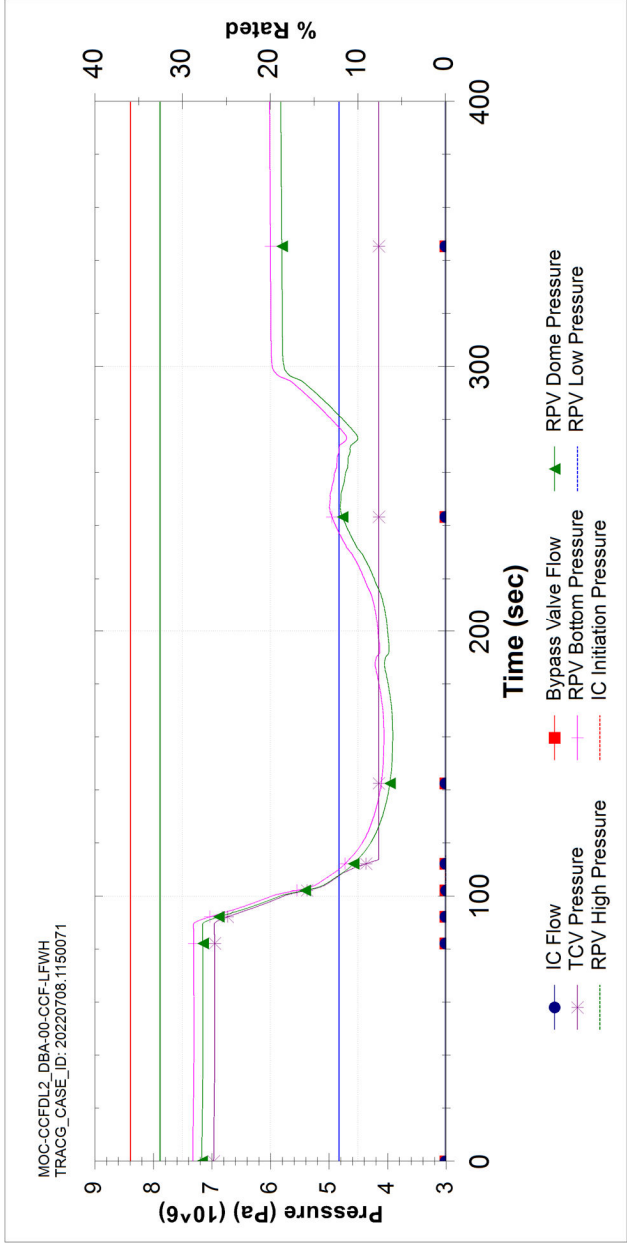


Figure 15.5-47: Loss of Feedwater Heating (DBA)

NEDO-33965 REVISION 0
NON-PROPRIETARY INFORMATION

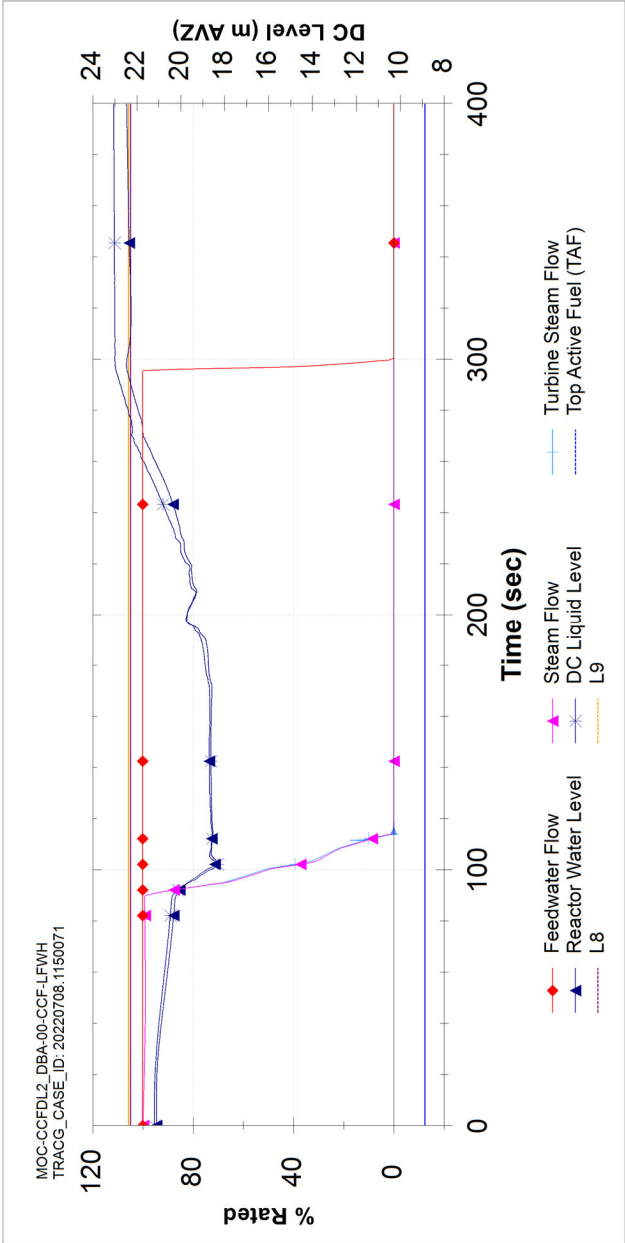


Figure 15.5-48: Loss of Feedwater Heating (DBA)

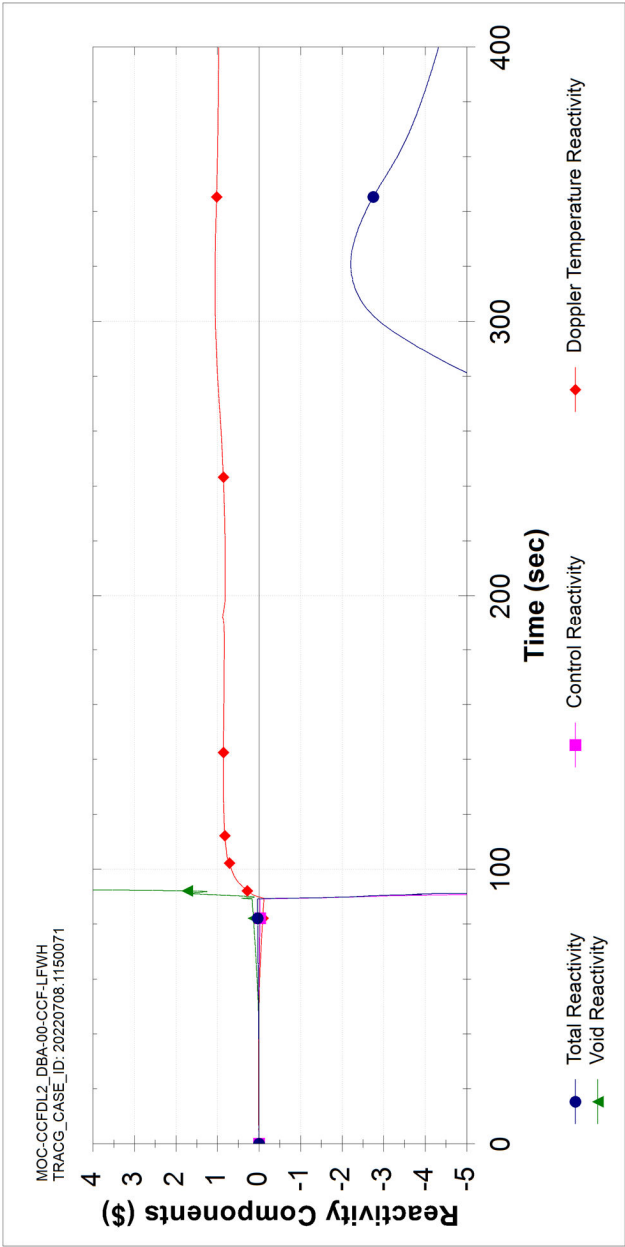


Figure 15.5-49: Loss of Feedwater Heating (DBA)

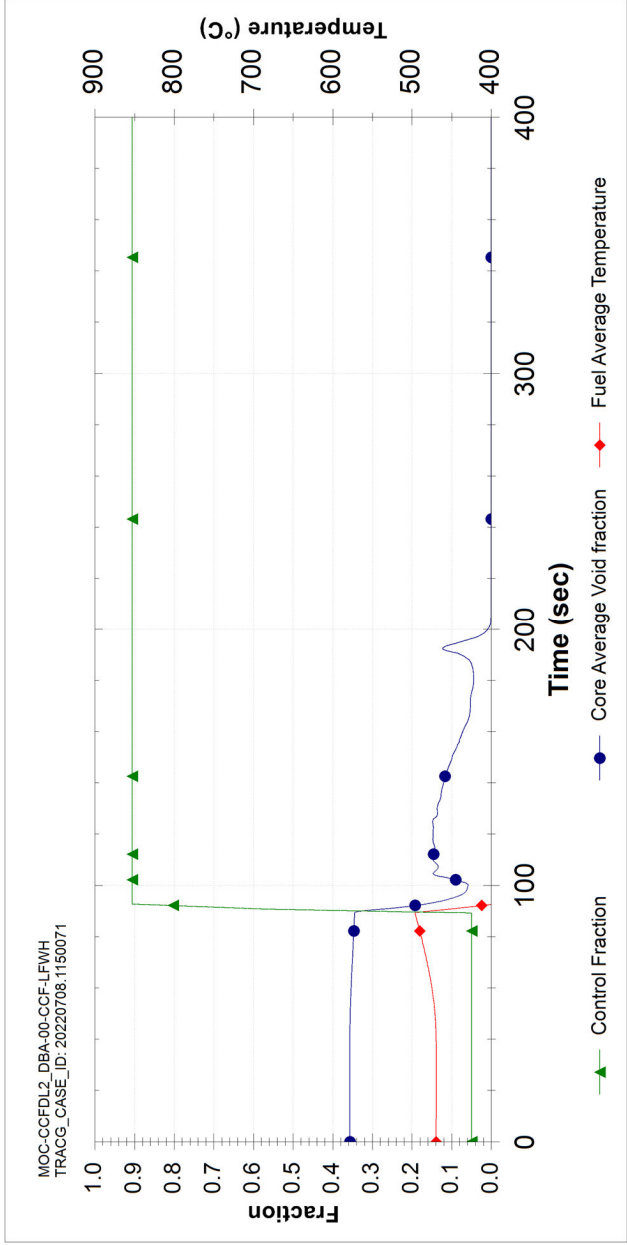


Figure 15.5-50: Loss of Feedwater Heating (DBA)

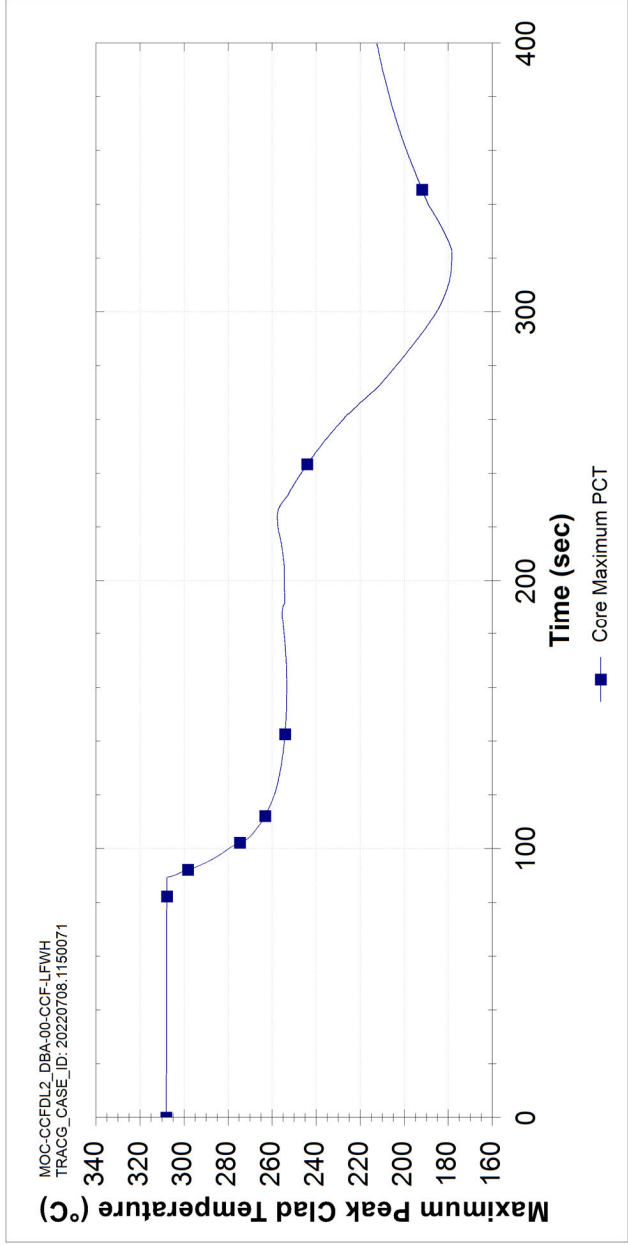


Figure 15.5-51: Loss of Feedwater Heating (DBA)

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NON-PROPRIETARY INFORMATION

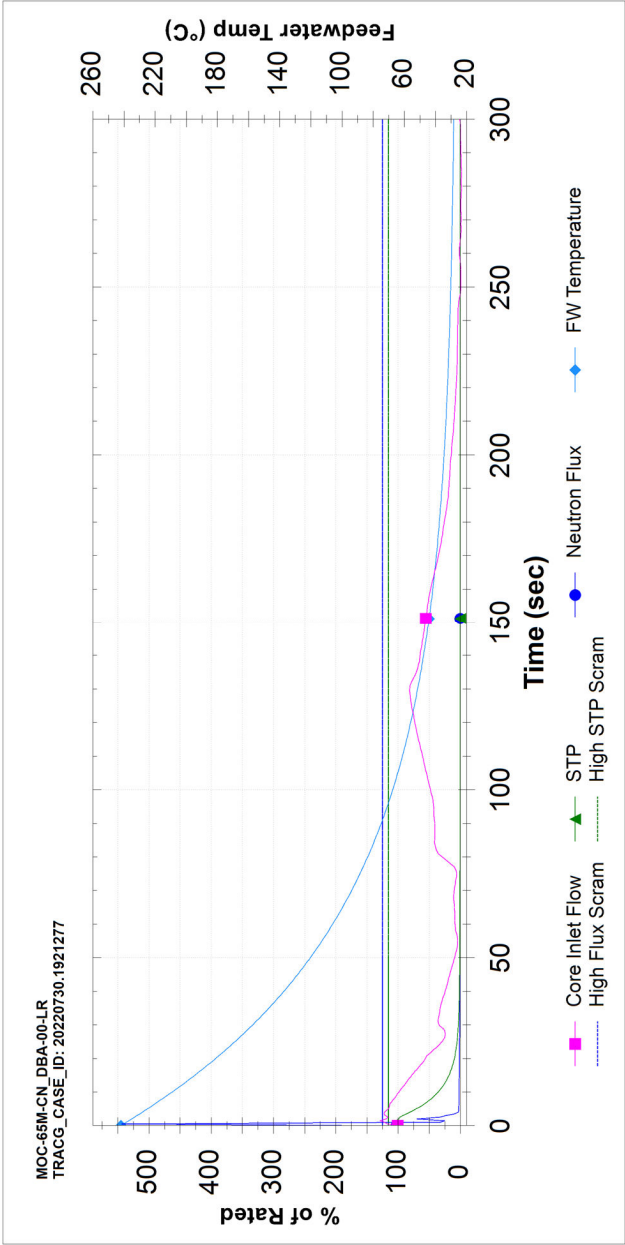


Figure 15.5-52: Generator Load Rejection (DBA)

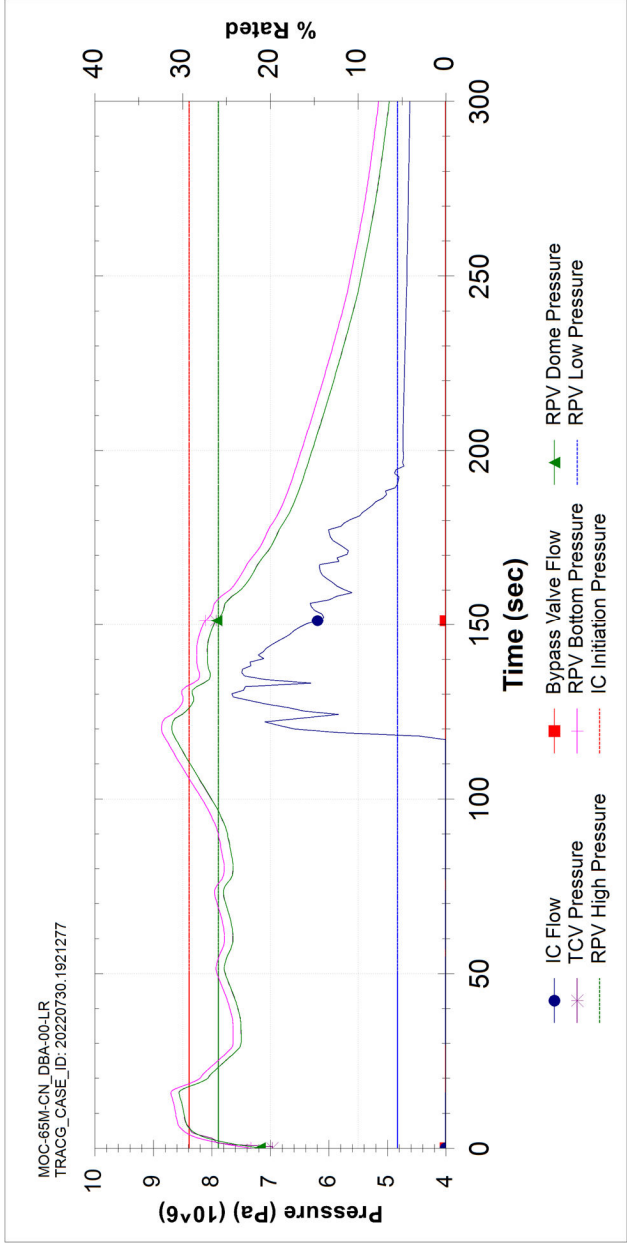


Figure 15.5-53: Generator Load Rejection (DBA)

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NON-PROPRIETARY INFORMATION

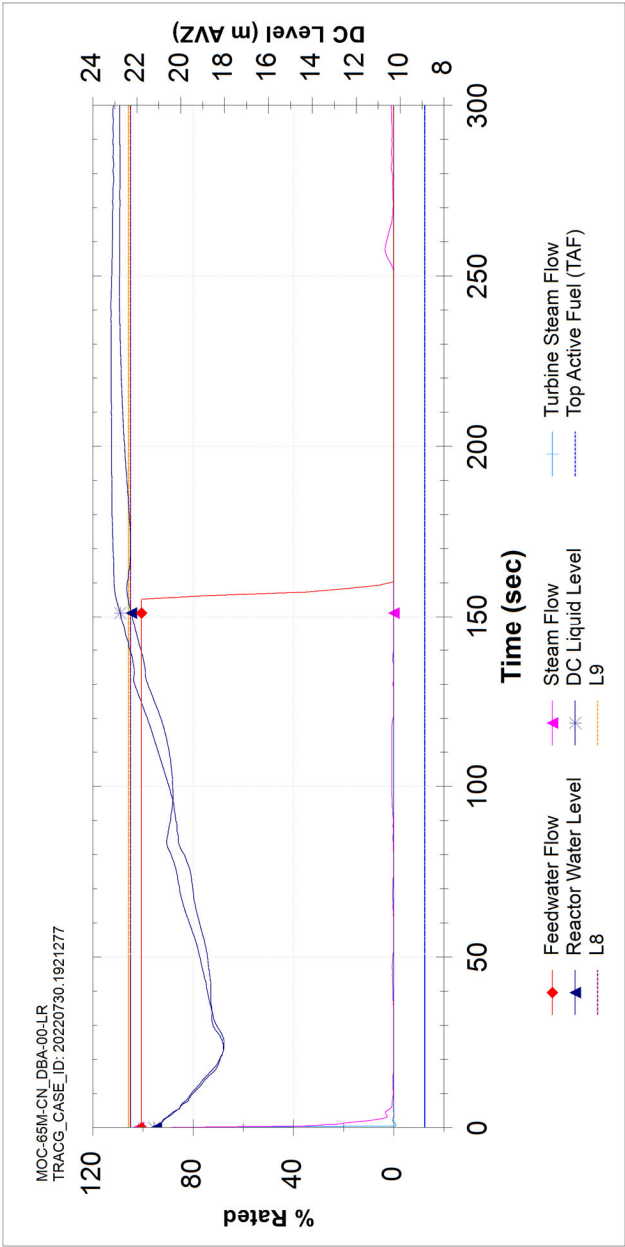


Figure 15.5-54: Generator Load Rejection (DBA)

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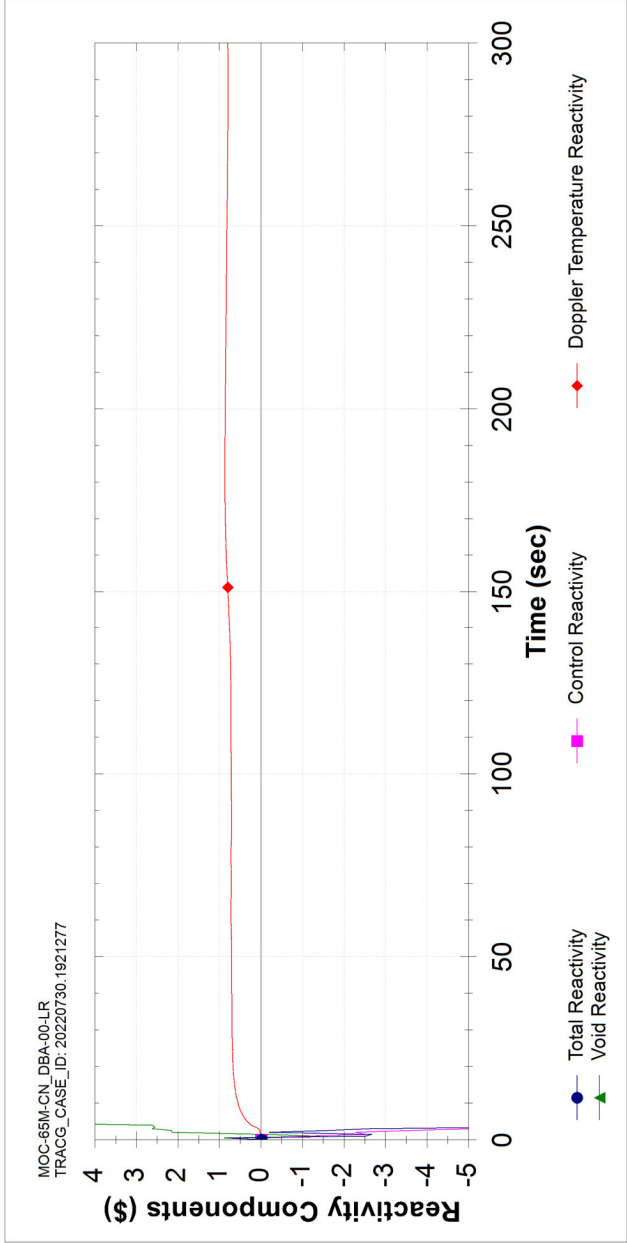


Figure 15.5-55: Generator Load Rejection (DBA)

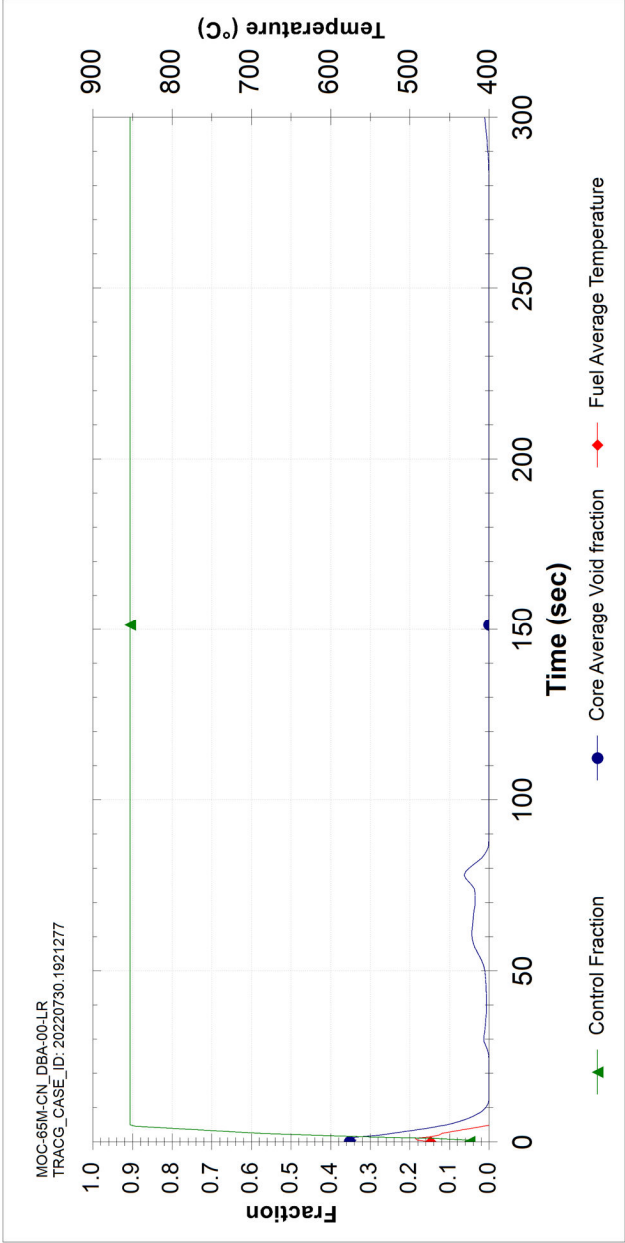


Figure 15.5-56: Generator Load Rejection (DBA)

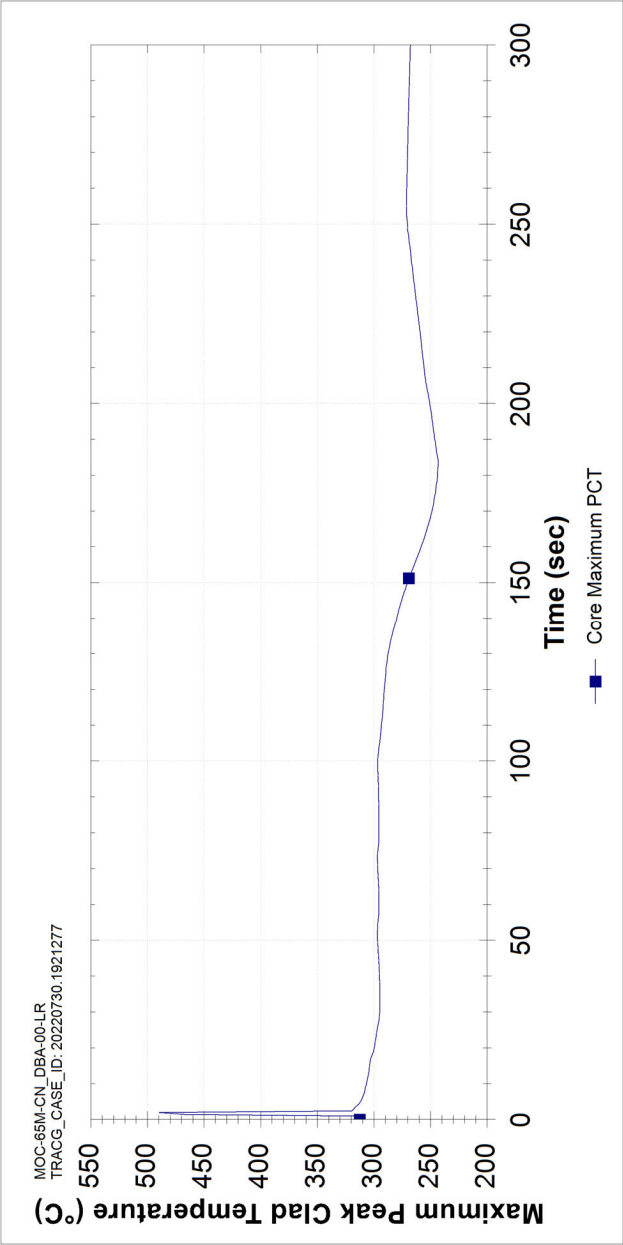


Figure 15.5-57: Generator Load Rejection (DBA)

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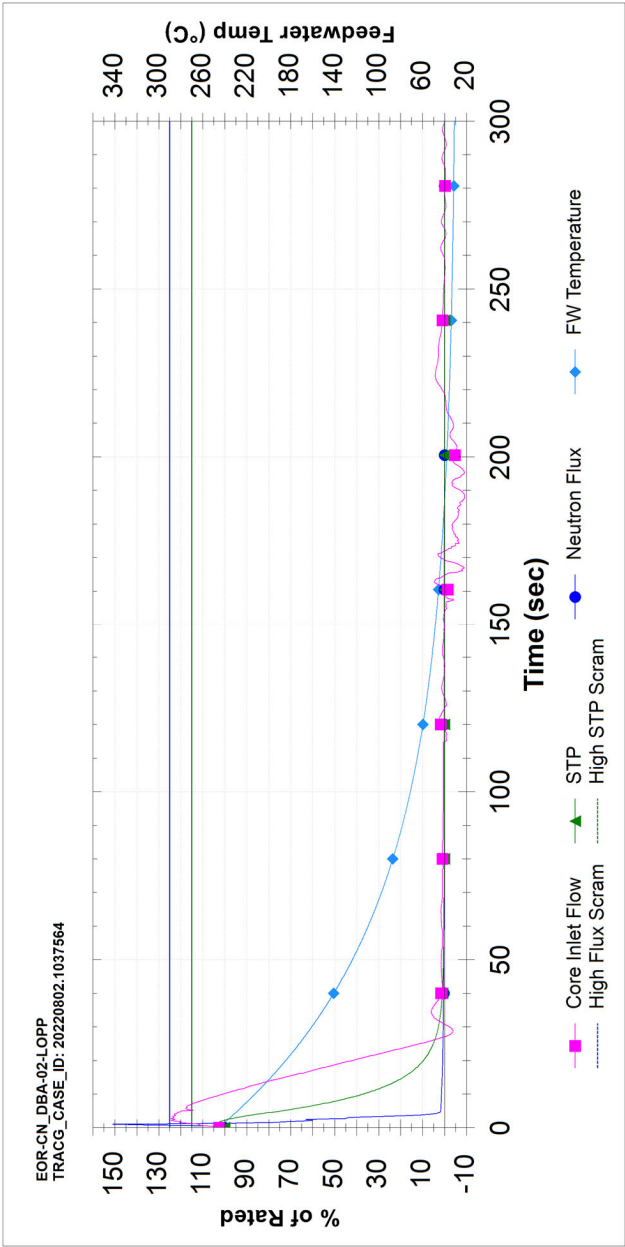


Figure 15.5-58: Loss of Preferred Power (DBA)

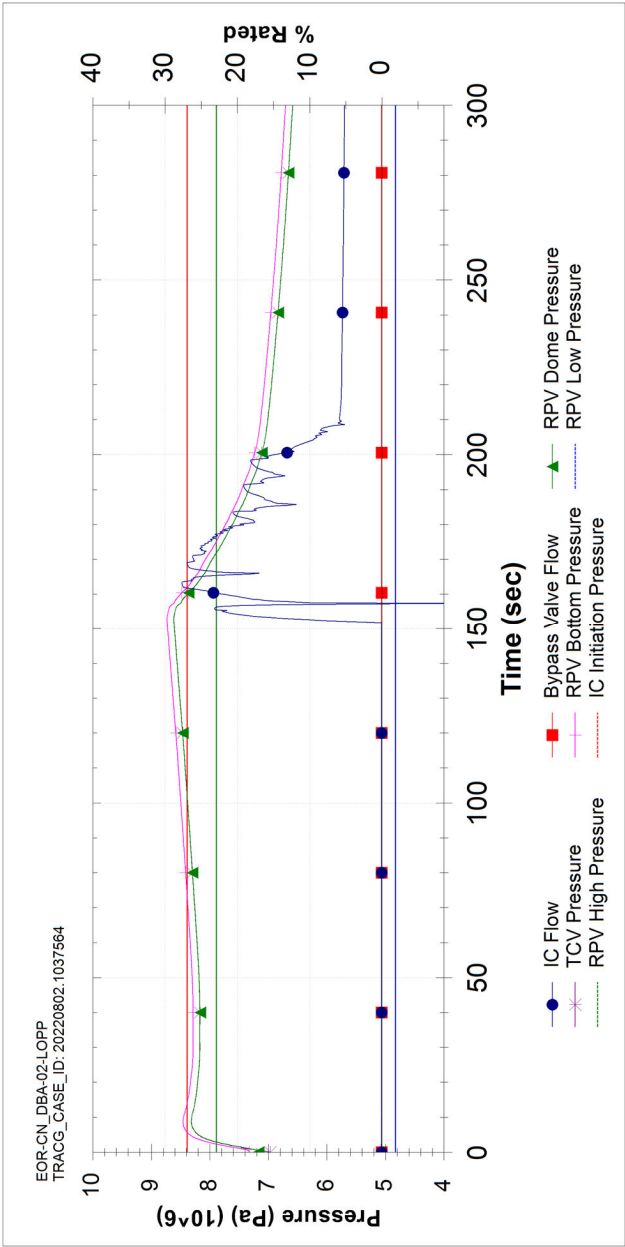


Figure 15.5-59: Loss of Preferred Power (DBA)

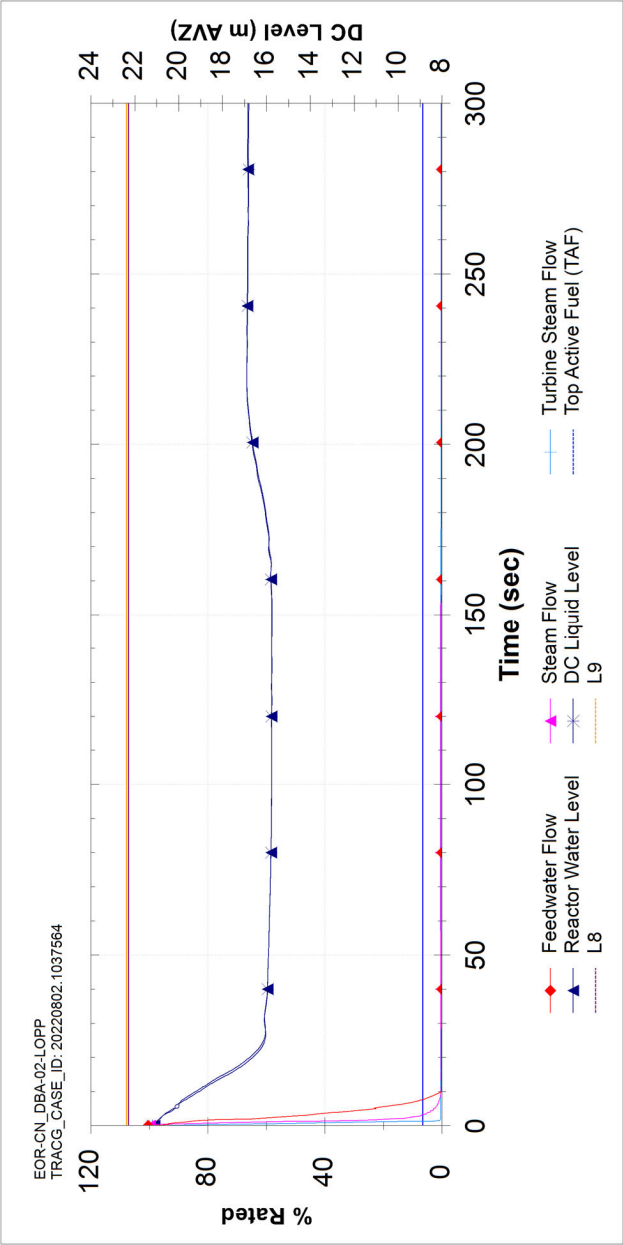


Figure 15.5-60: Loss of Preferred Power (DBA)

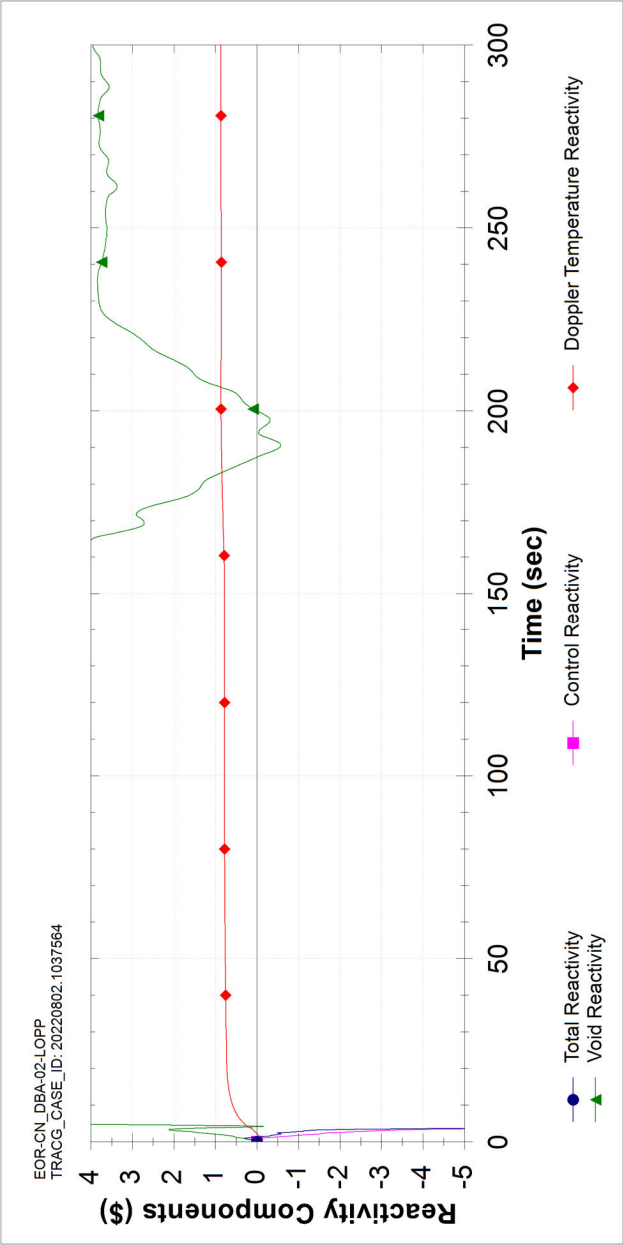


Figure 15.5-61: Loss of Preferred Power (DBA)

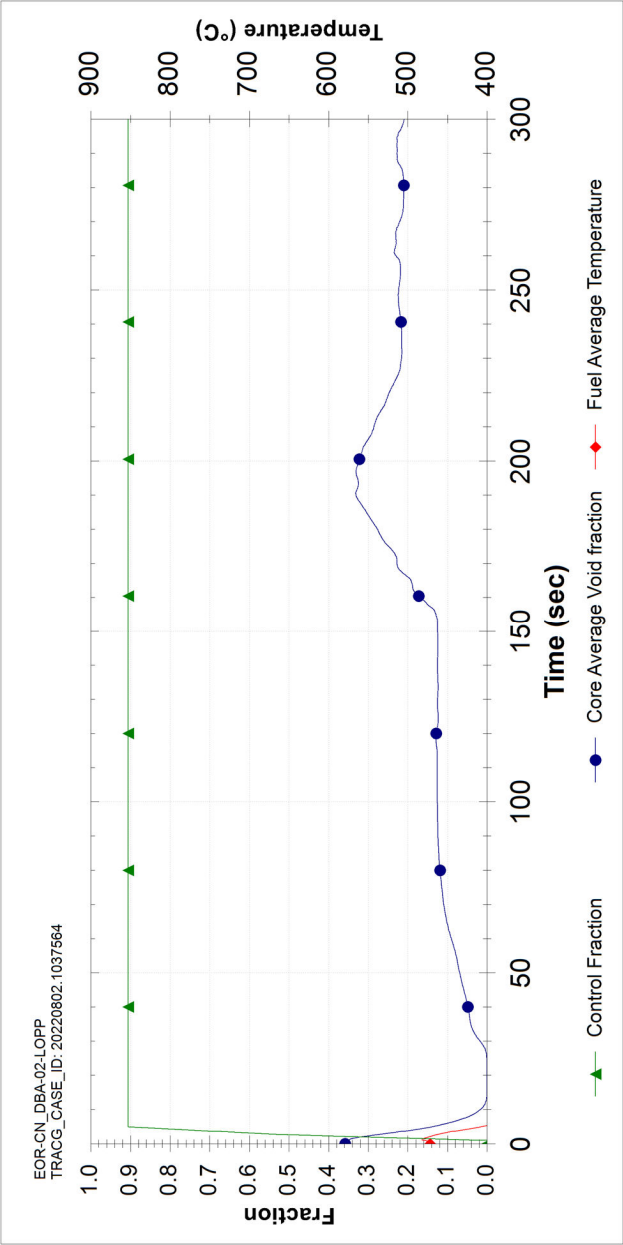


Figure 15.5-62: Loss of Preferred Power (DBA)

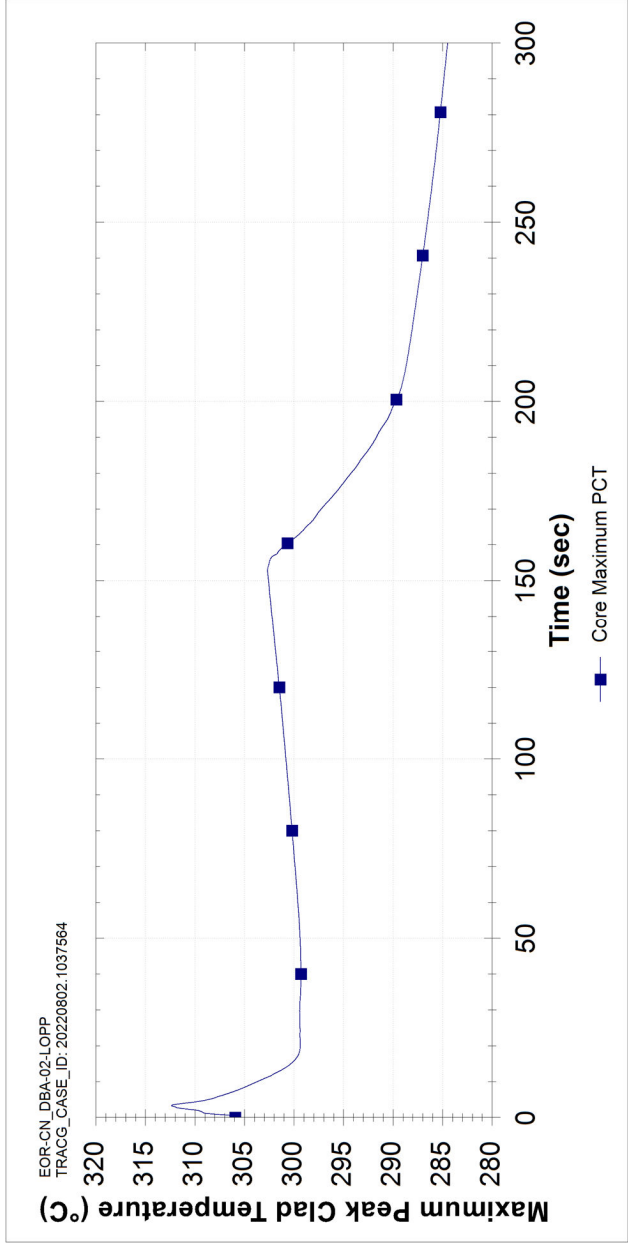


Figure 15.5-63: Loss of Preferred Power (DBA)

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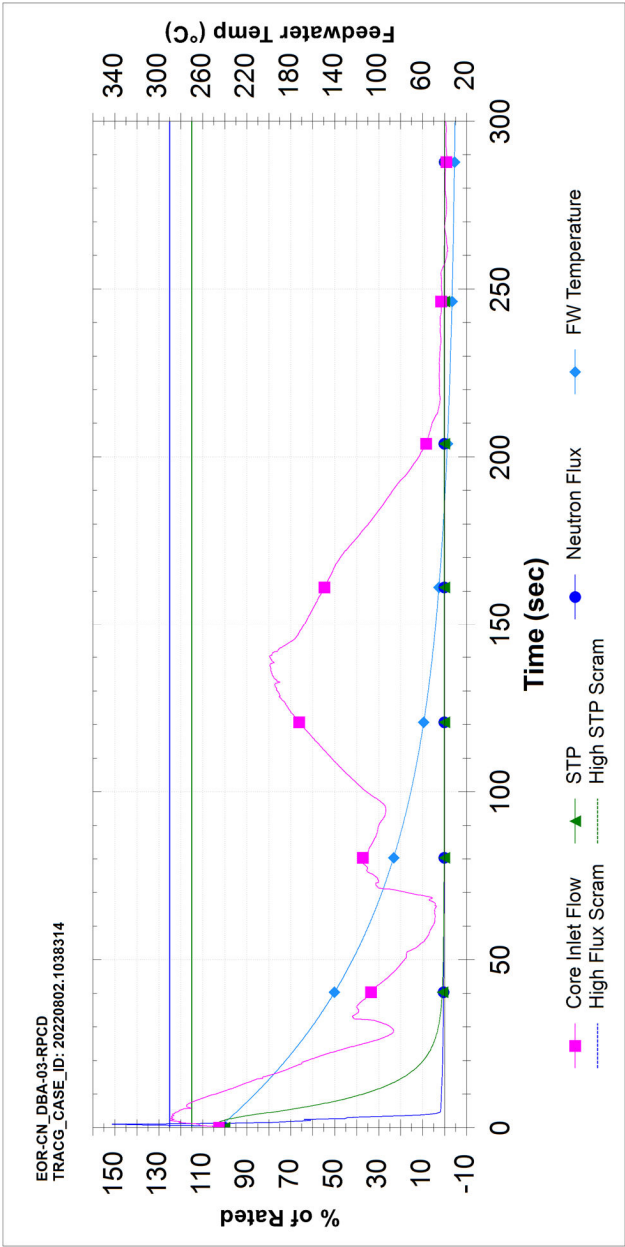


Figure 15.5-64: RPV Pressure Control Downscale (DBA)

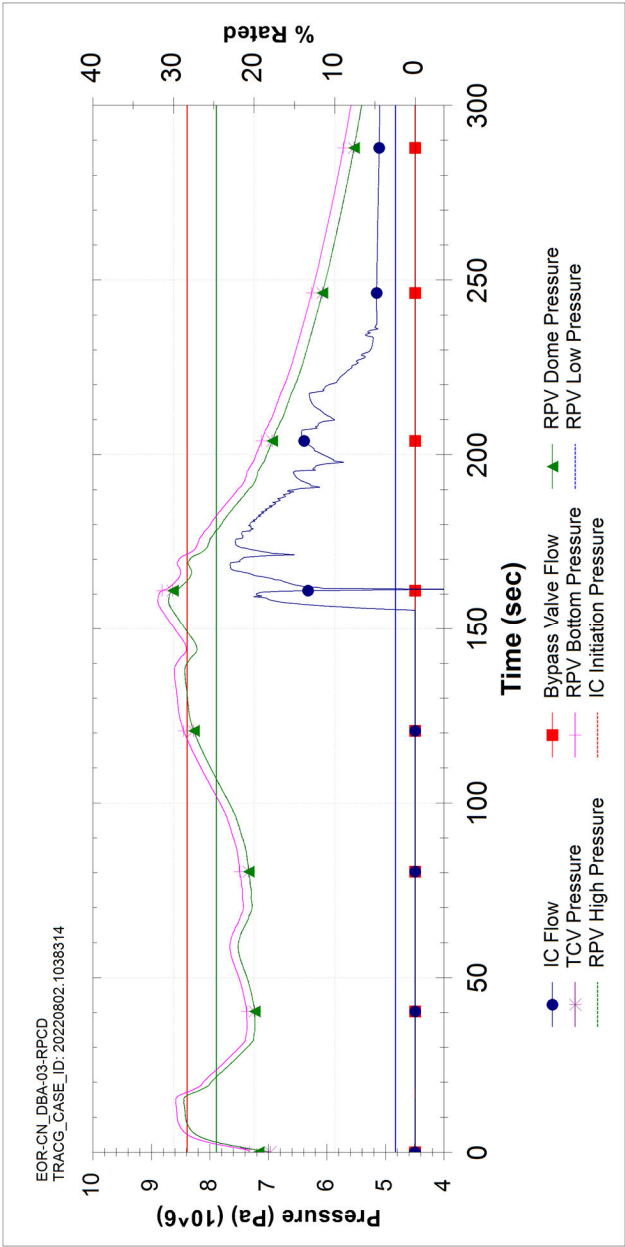


Figure 15.5-65: RPV Pressure Control Downslope (DBA)

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NON-PROPRIETARY INFORMATION

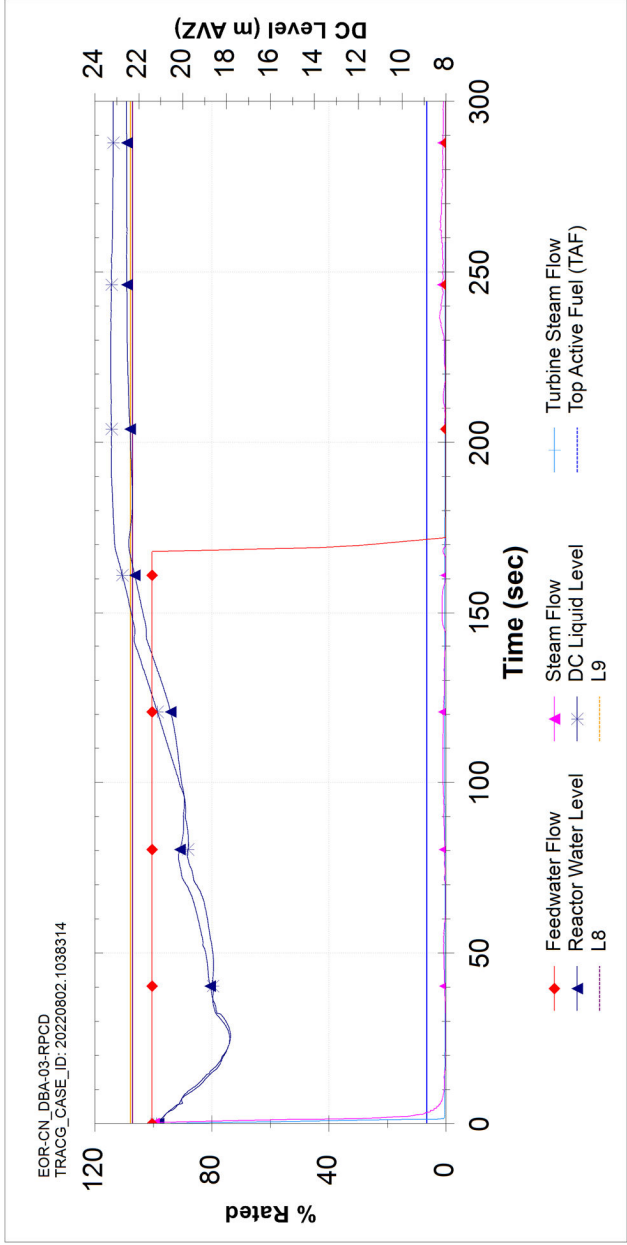


Figure 15.5-66: RPV Pressure Control Downscale (DBA)

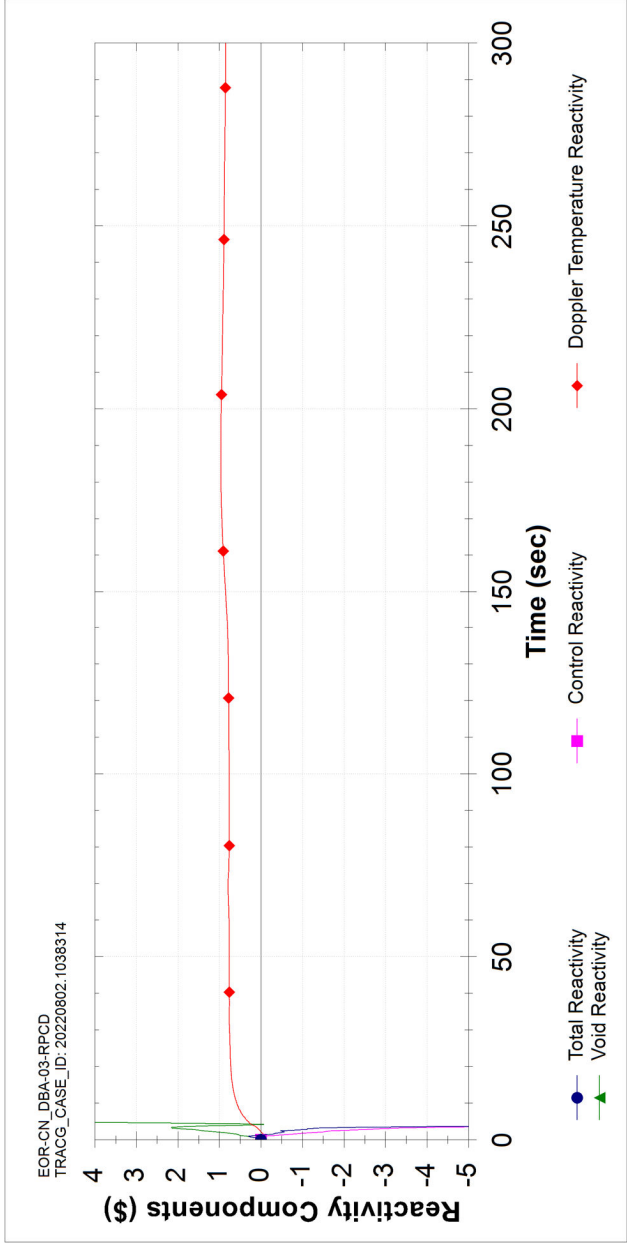


Figure 15.5-67: RPV Pressure Control Downscale (DBA)

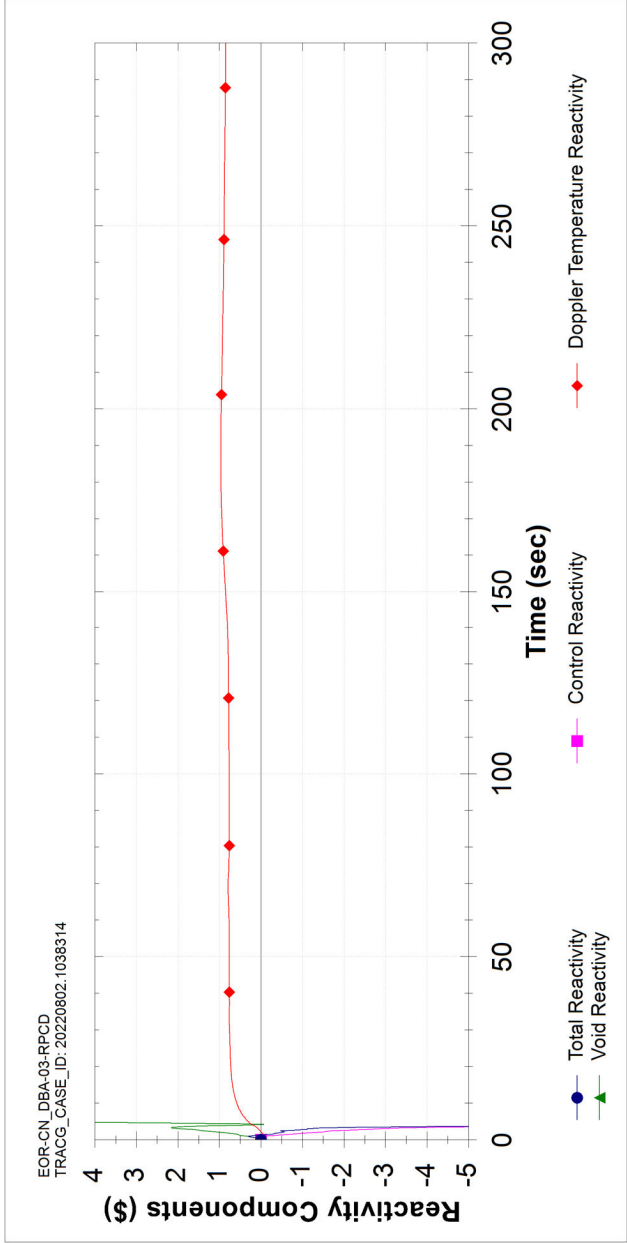


Figure 15.6-68a: RPV Pressure Control Downscale (DBA)

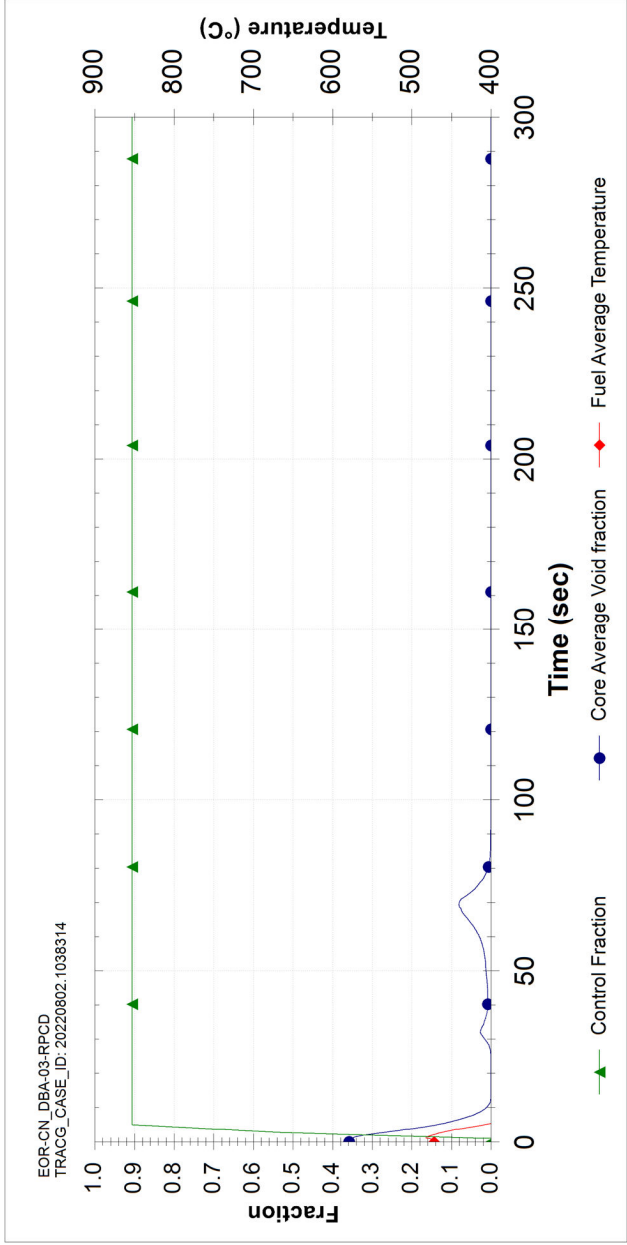


Figure 15.5-68b: RPV Pressure Control Downslope (DBA)

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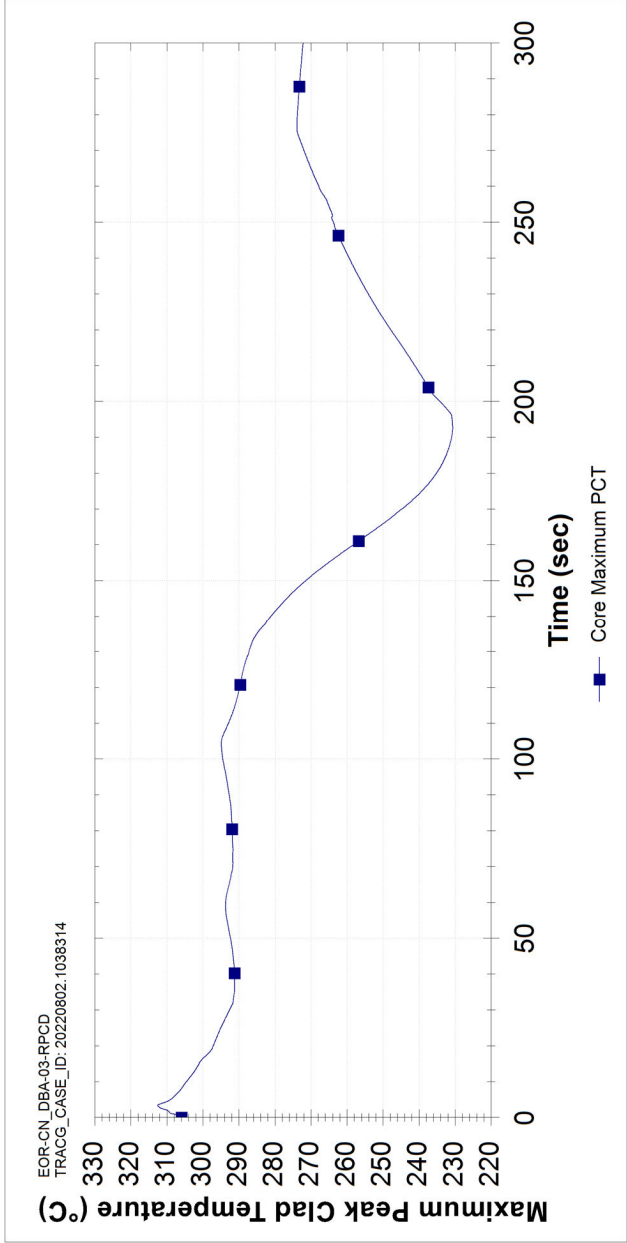


Figure 15.5-69: RPV Pressure Control Downscale (DBA)

NEDO-33965 REVISION 0
NON-PROPRIETARY INFORMATION

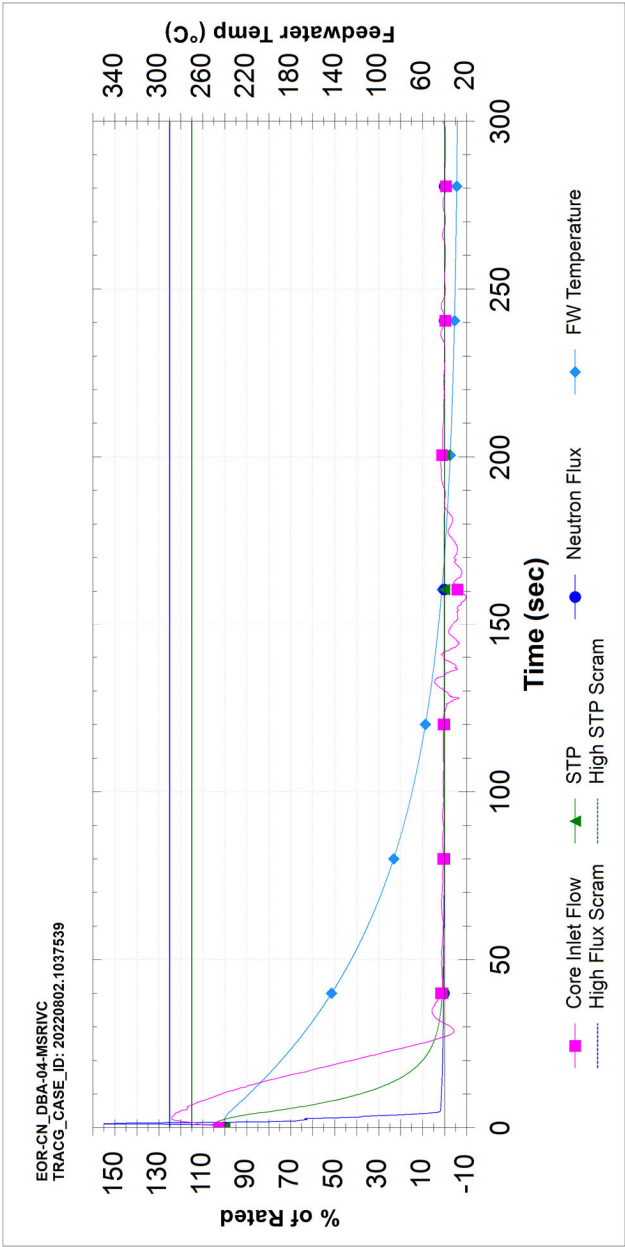


Figure 15.5-70: Closure of All MSRIVs and FW Isolation Valves (DBA)

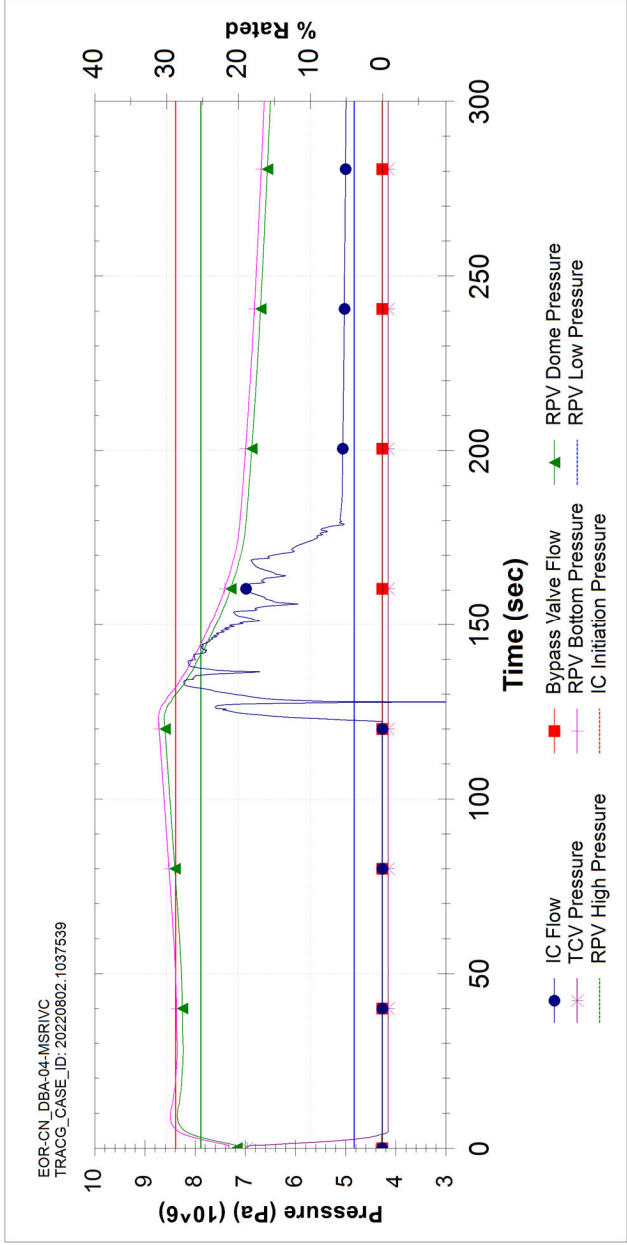


Figure 15.5-71: Closure of All MSRIVs and FW Isolation Valves (DBA)

NEDO-33965 REVISION 0
NON-PROPRIETARY INFORMATION

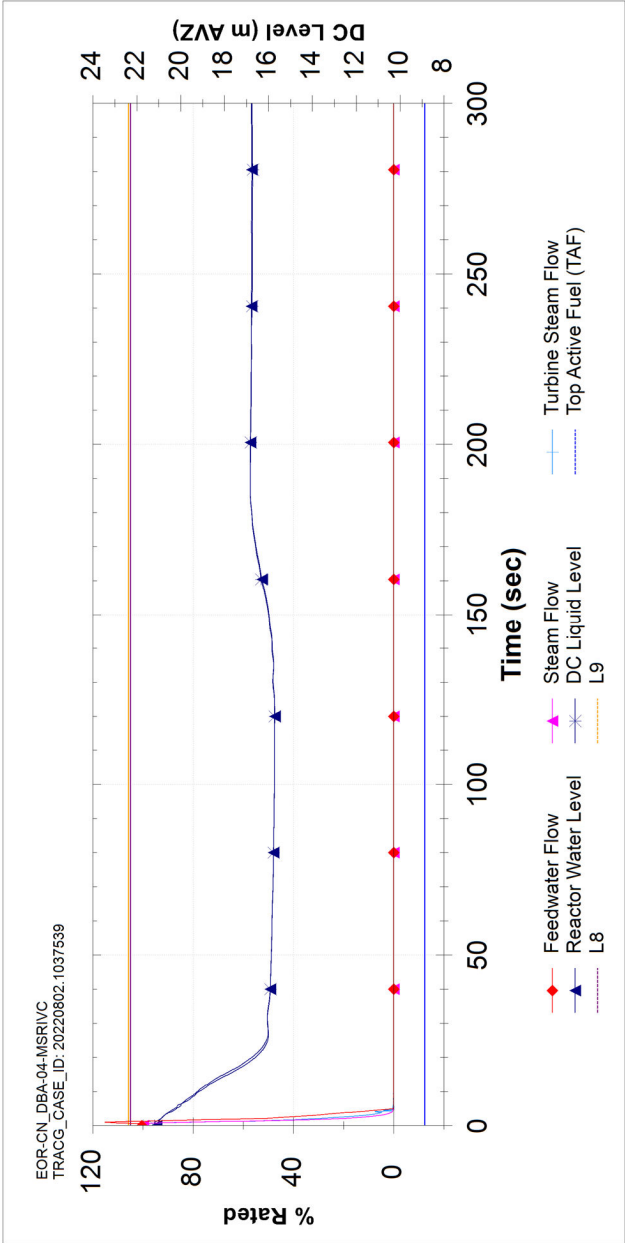


Figure 15.5-72: Closure of All MSRIVs and FW Isolation Valves (DBA)

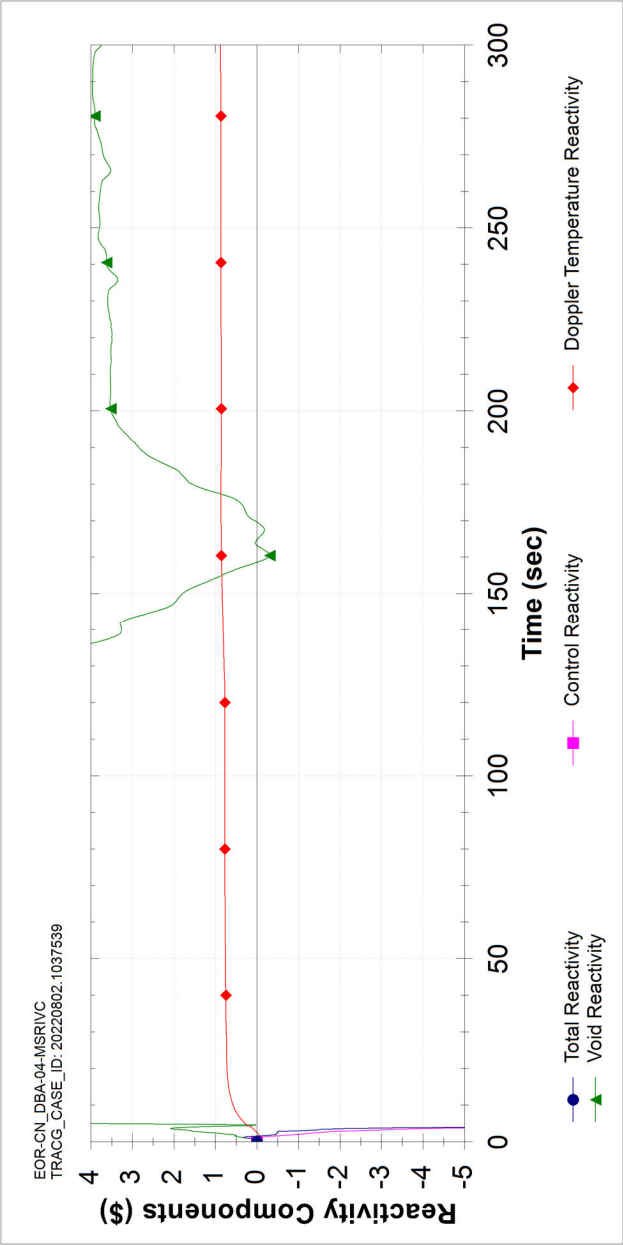


Figure 15.5-73: Closure of All MSRIVs and FW Isolation Valves (DBA)

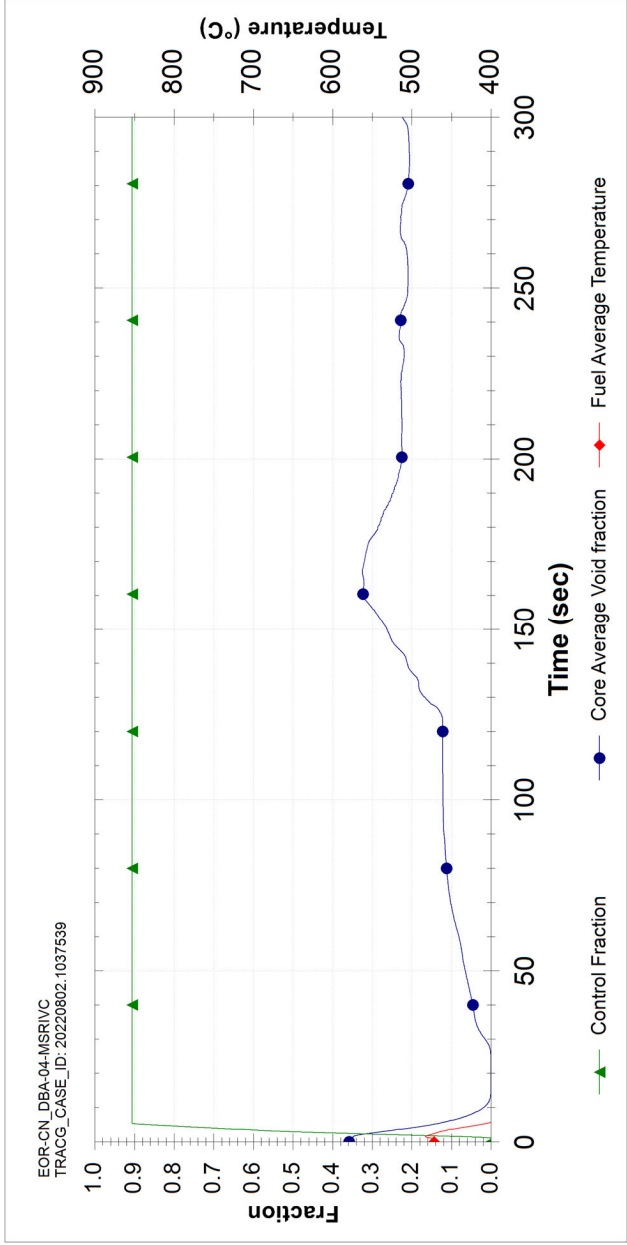


Figure 15.5-74: Closure of All MSRIVs and FW Isolation Valves (DBA)

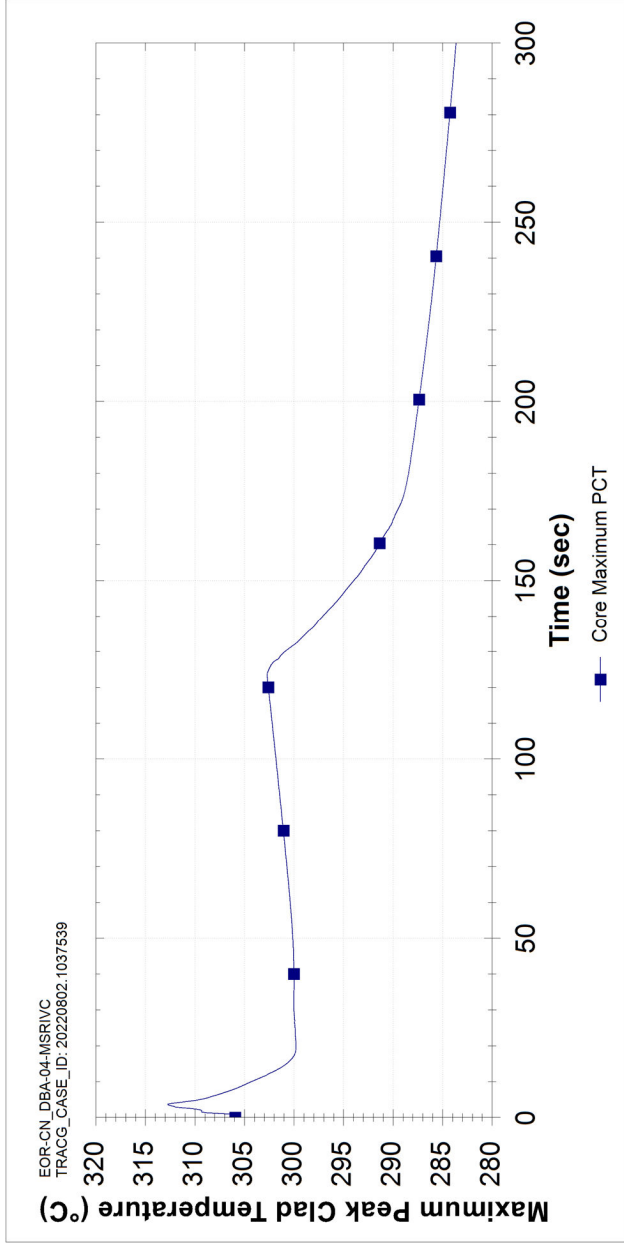


Figure 15.5-75: Closure of All MSRIVs and FW Isolation Valves (DBA)

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NON-PROPRIETARY INFORMATION

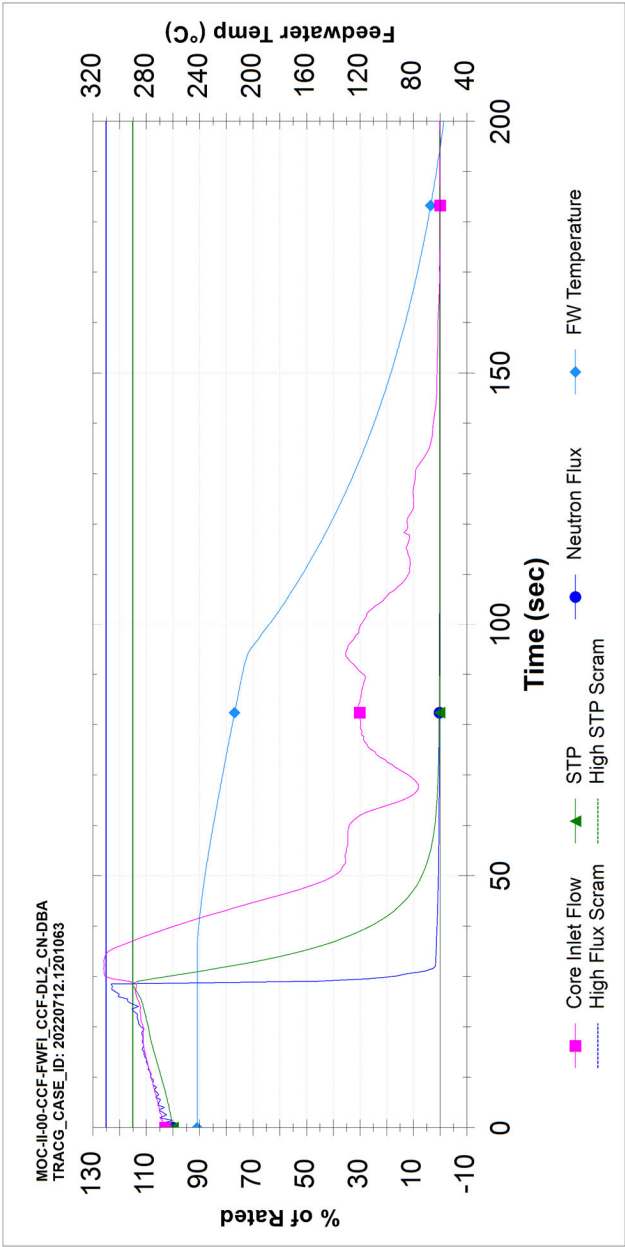


Figure 15.5-76: Feedwater Flow Increase (DBA)

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NON-PROPRIETARY INFORMATION

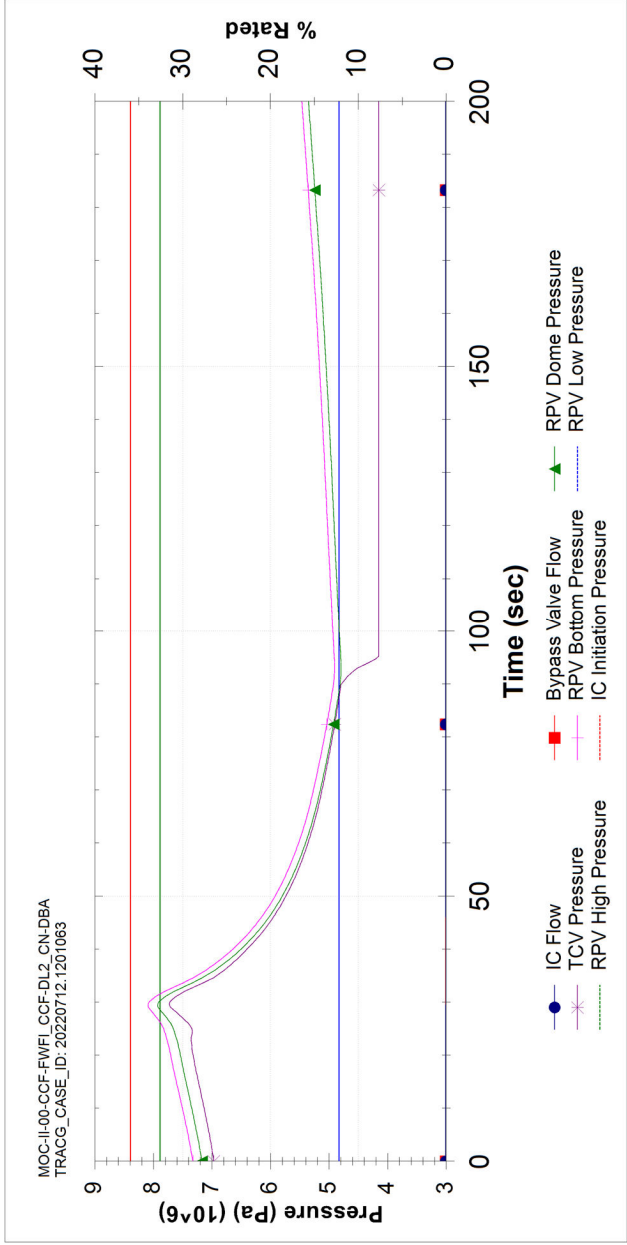


Figure 15.5-77: Feedwater Flow Increase (DBA)

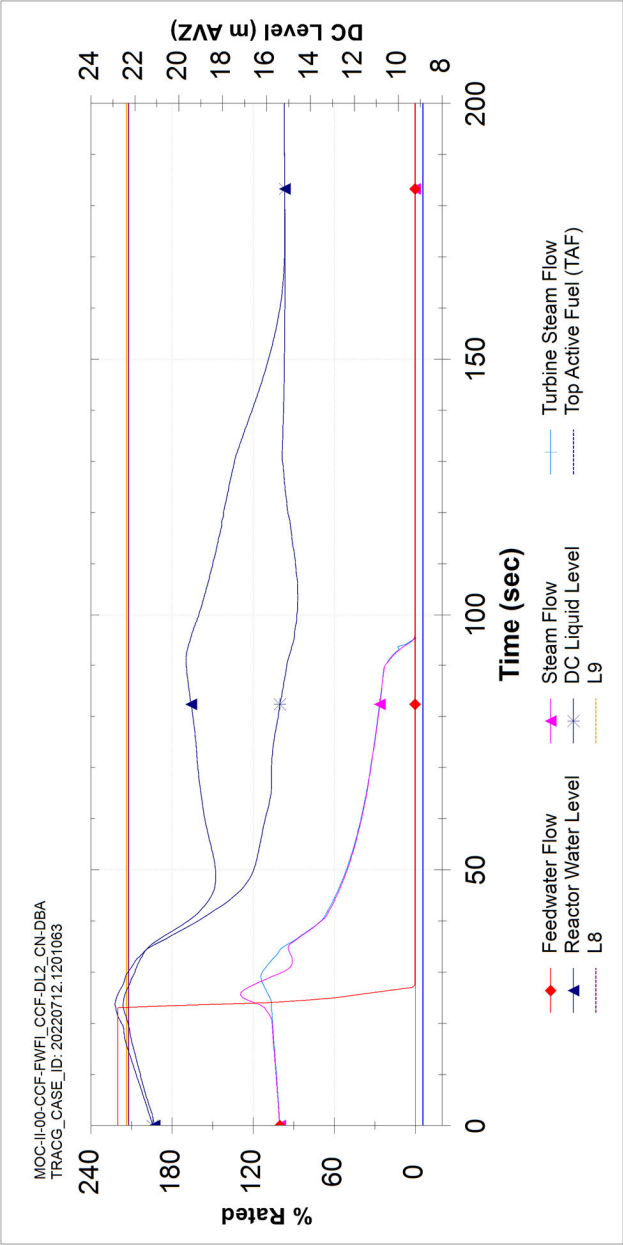


Figure 15.5-78: Feedwater Flow Increase (DBA)

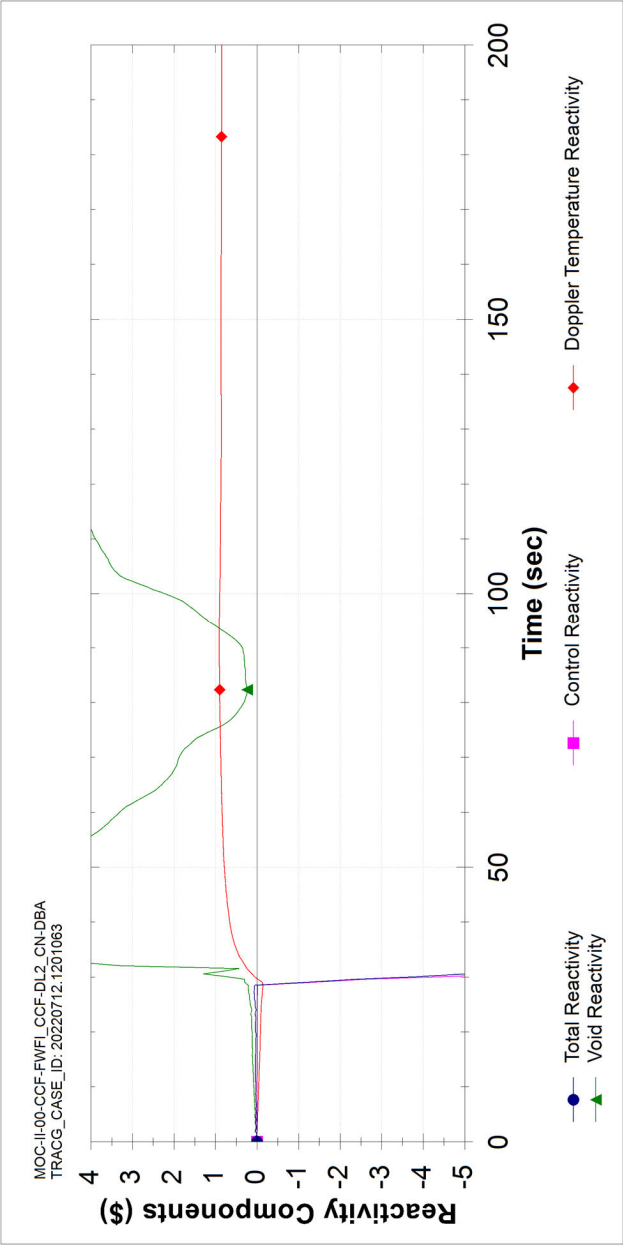


Figure 15.5-79: Feedwater Flow Increase (DBA)

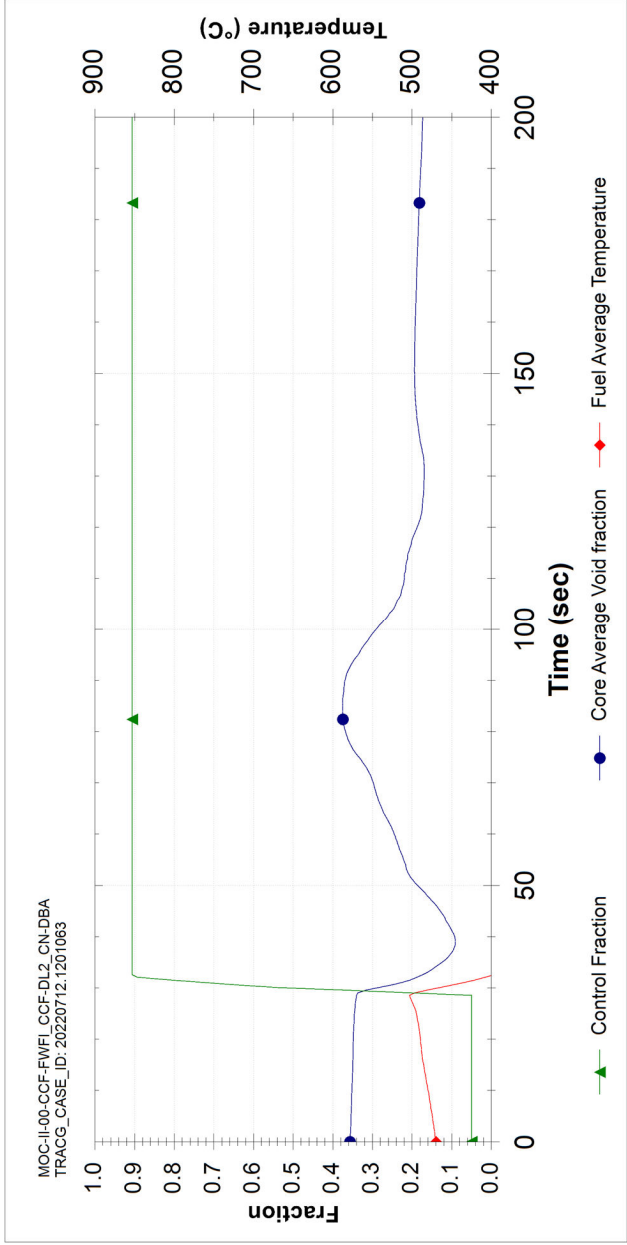


Figure 15.5-80: Feedwater Flow Increase (DBA)

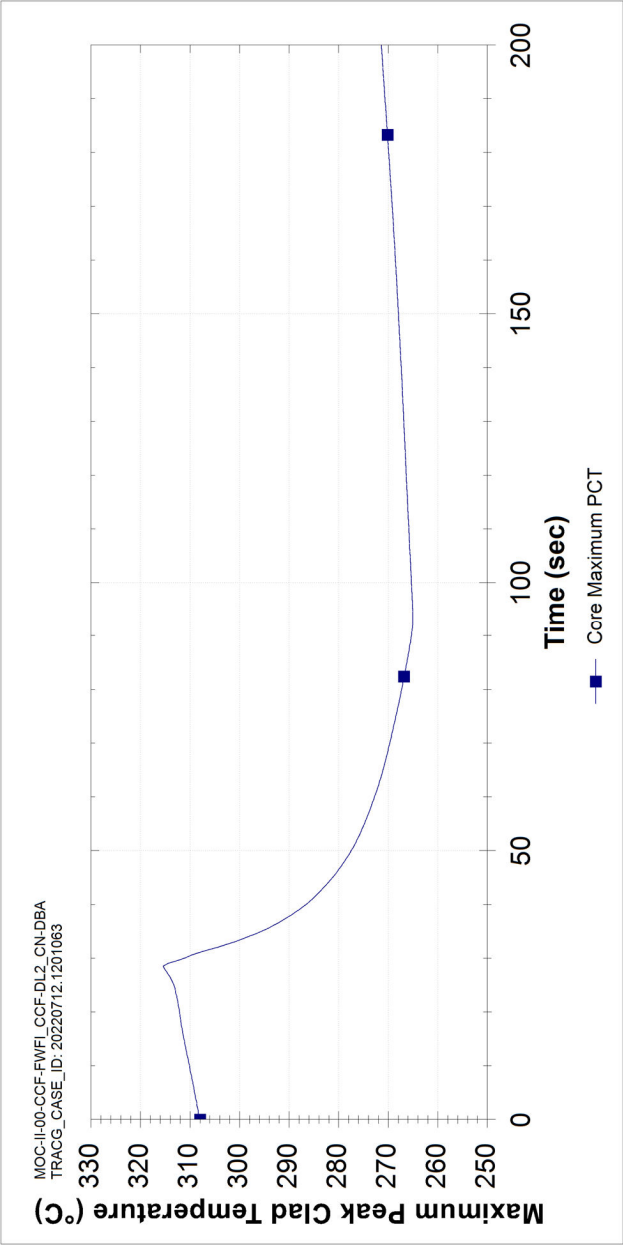


Figure 15.5-81: Feedwater Flow Increase (DBA)

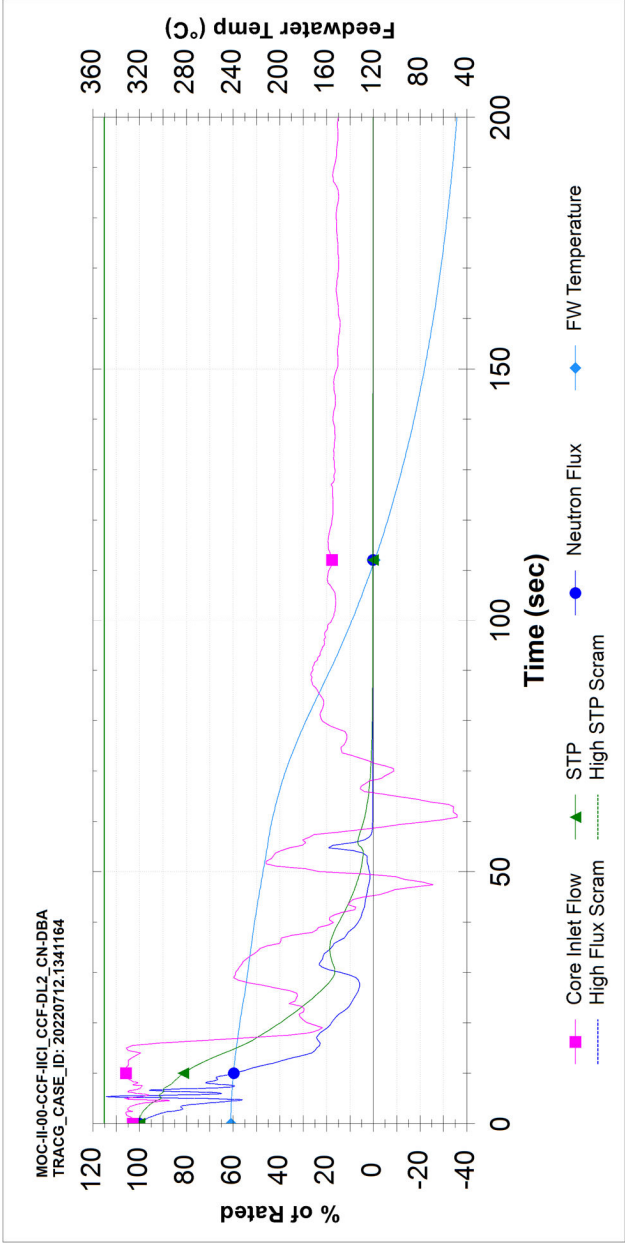


Figure 15.5-82: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

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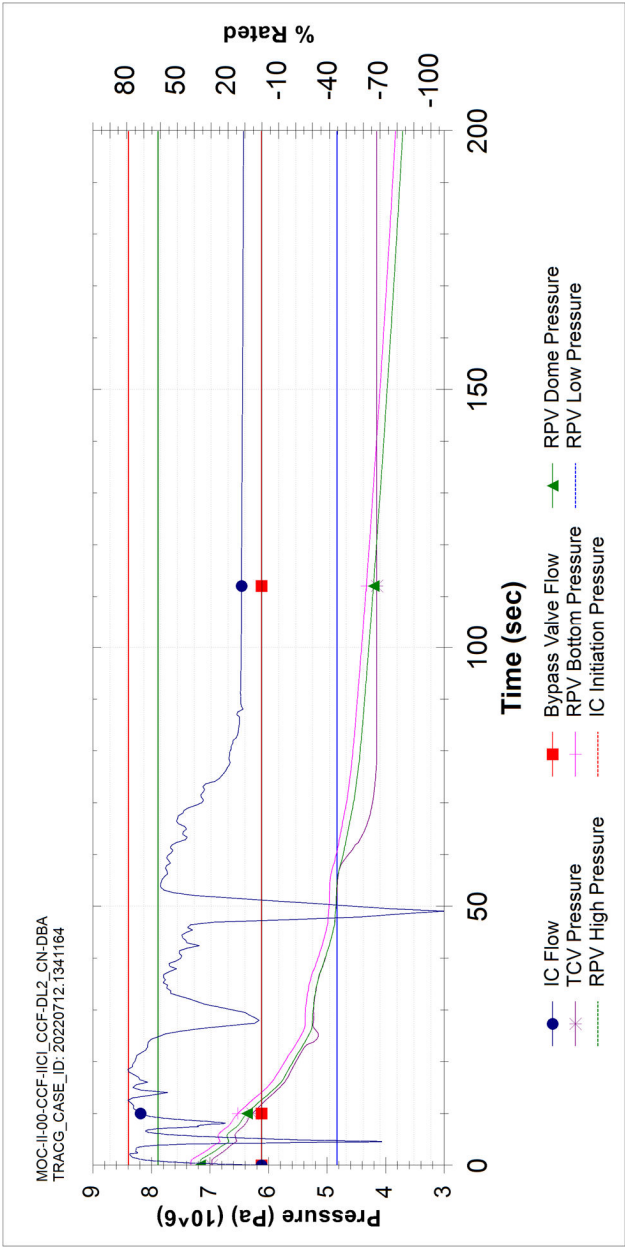


Figure 15.5-83: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

NEDO-33965 REVISION 0
NON-PROPRIETARY INFORMATION

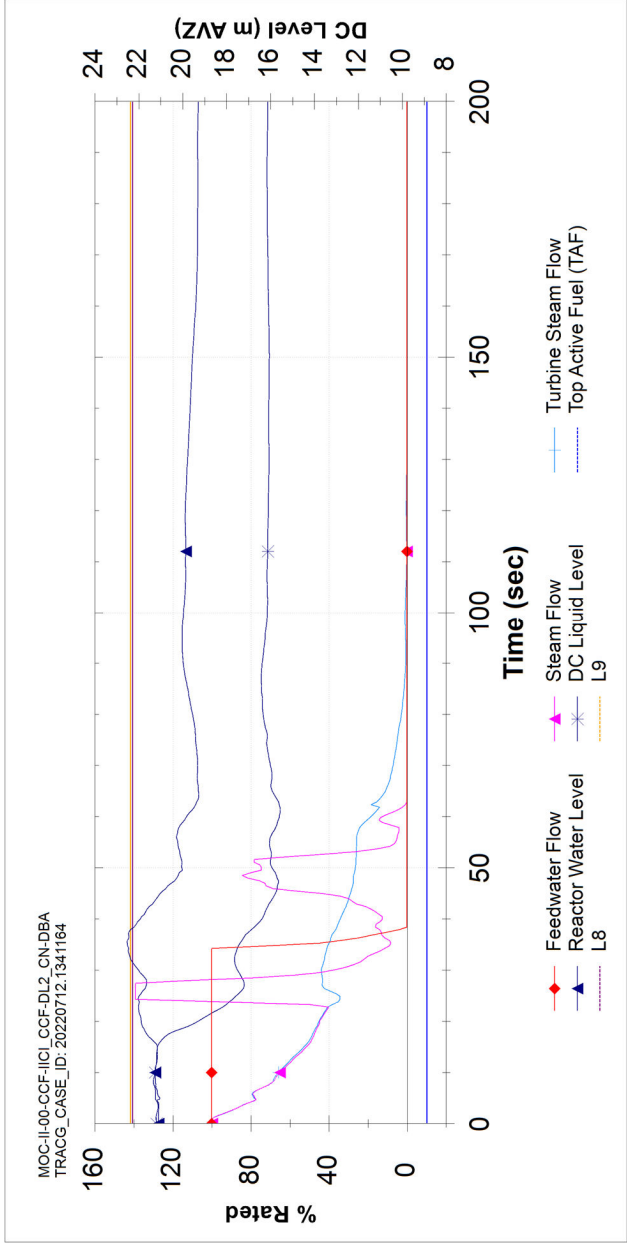


Figure 15.5-84: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

NEDO-33965 REVISION 0
NON-PROPRIETARY INFORMATION

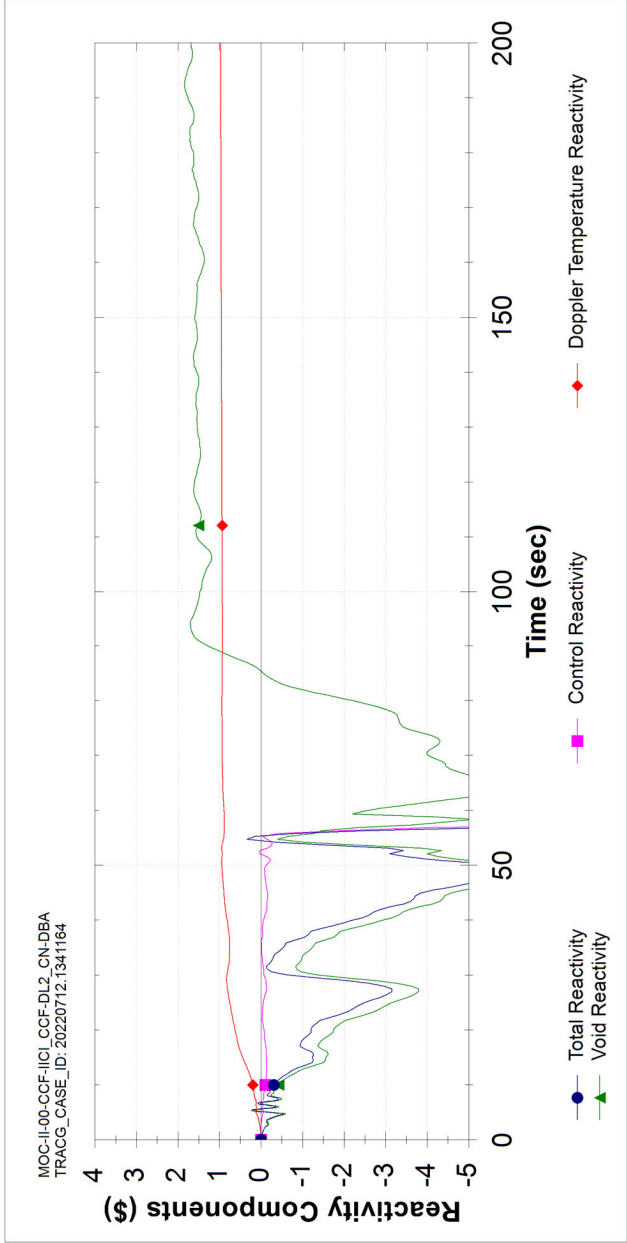


Figure 15.5-85: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

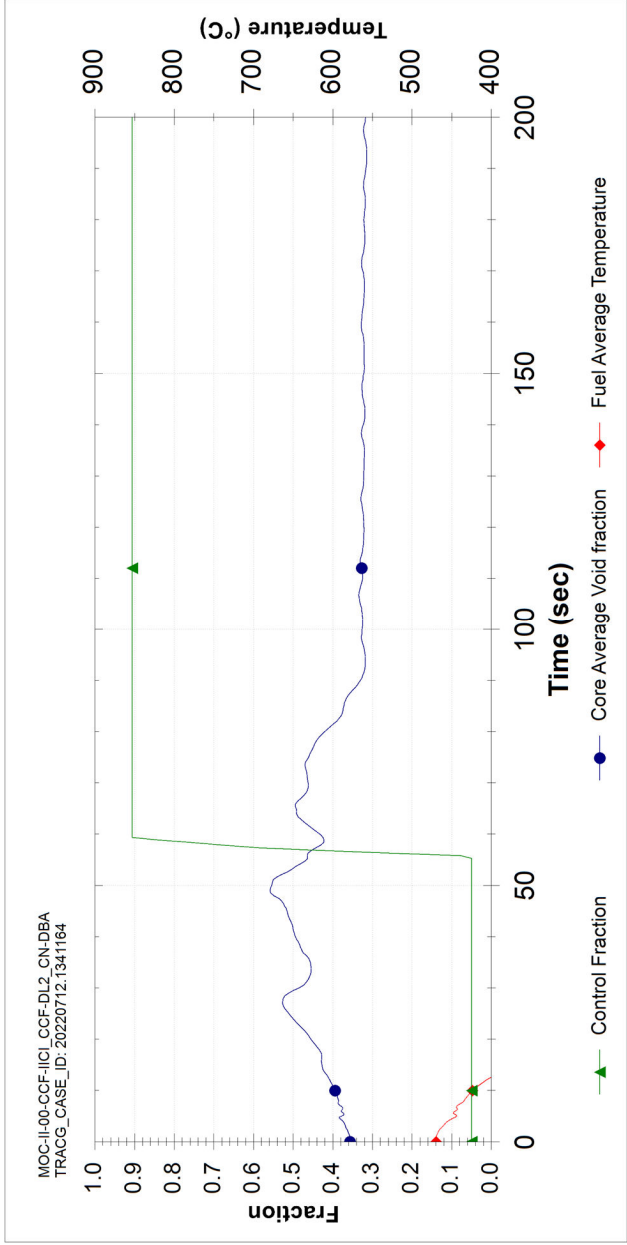


Figure 15.5-86: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

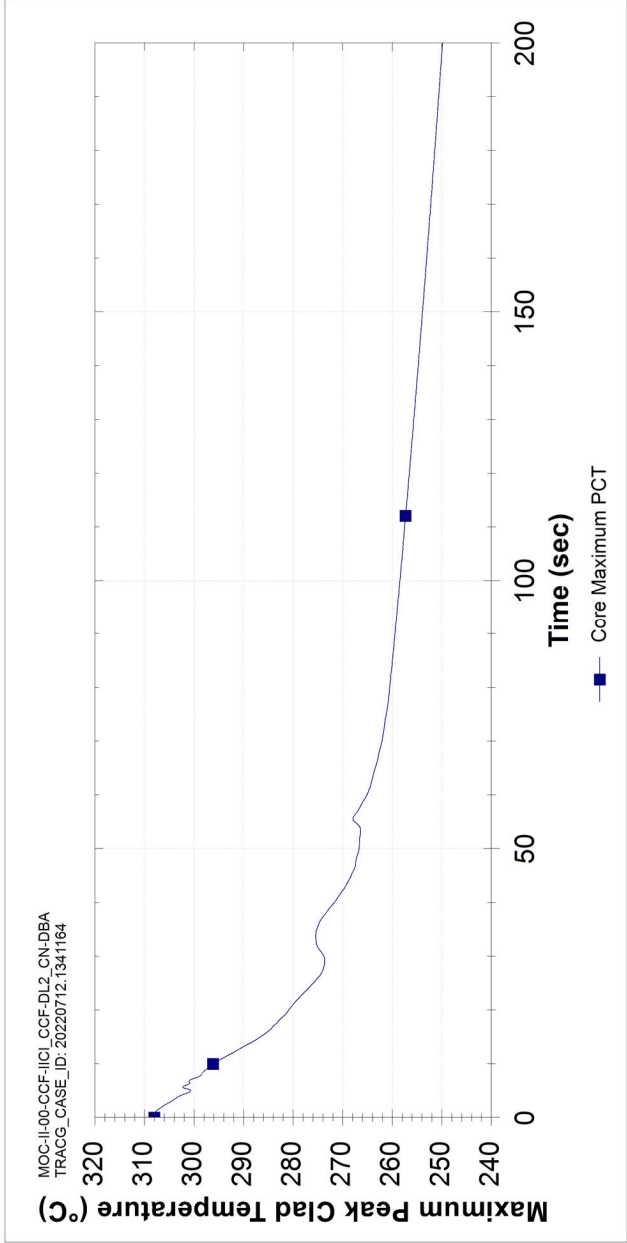


Figure 15.5-87: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

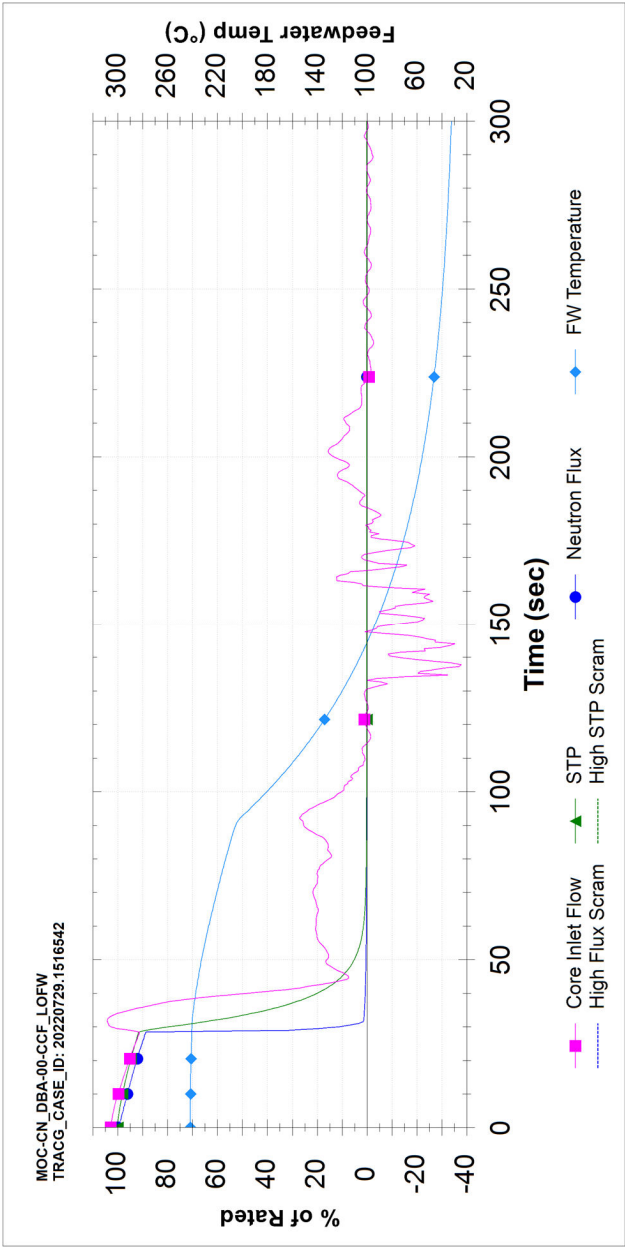


Figure 15.5-88: Loss of Feedwater Flow (DBA)

NEDO-33965 REVISION 0
NON-PROPRIETARY INFORMATION

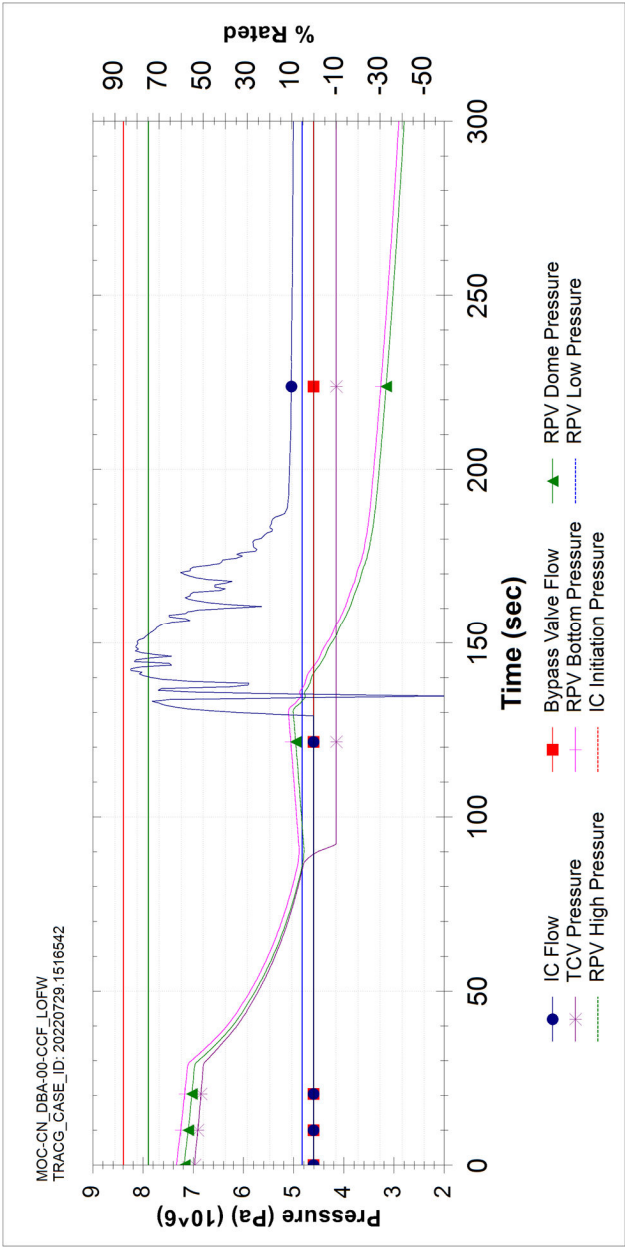


Figure 15.5-89: Loss of Feedwater Flow (DBA)

NEDO-33965 REVISION 0
NON-PROPRIETARY INFORMATION

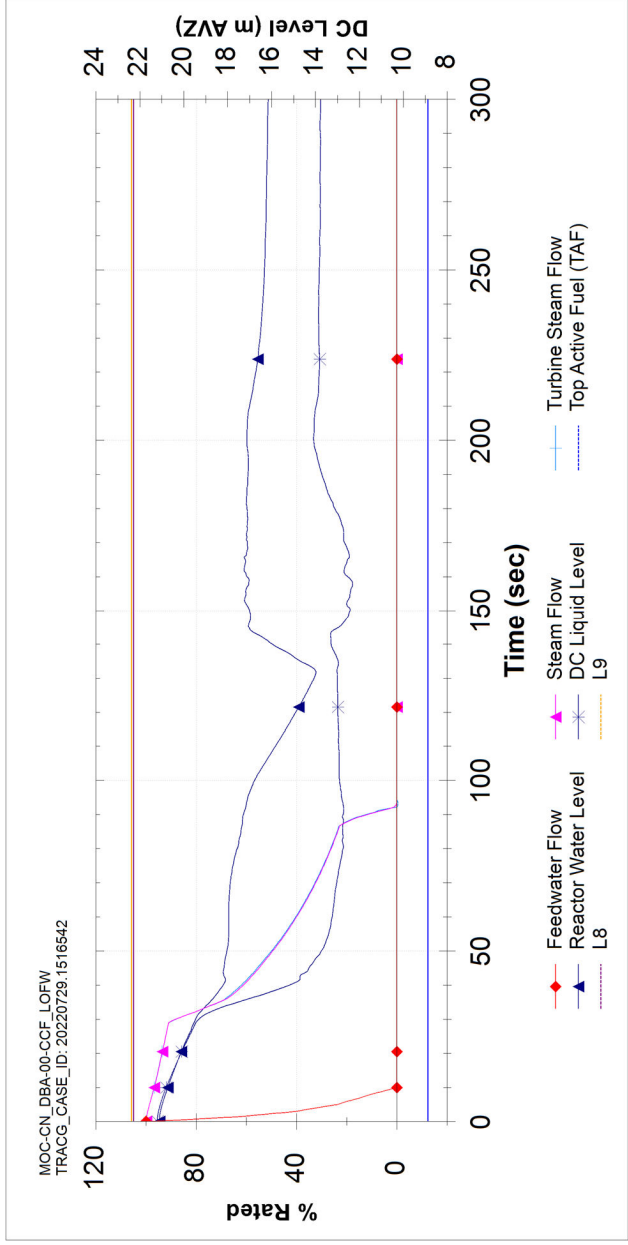


Figure 15.5-90: Loss of Feedwater Flow (DBA)

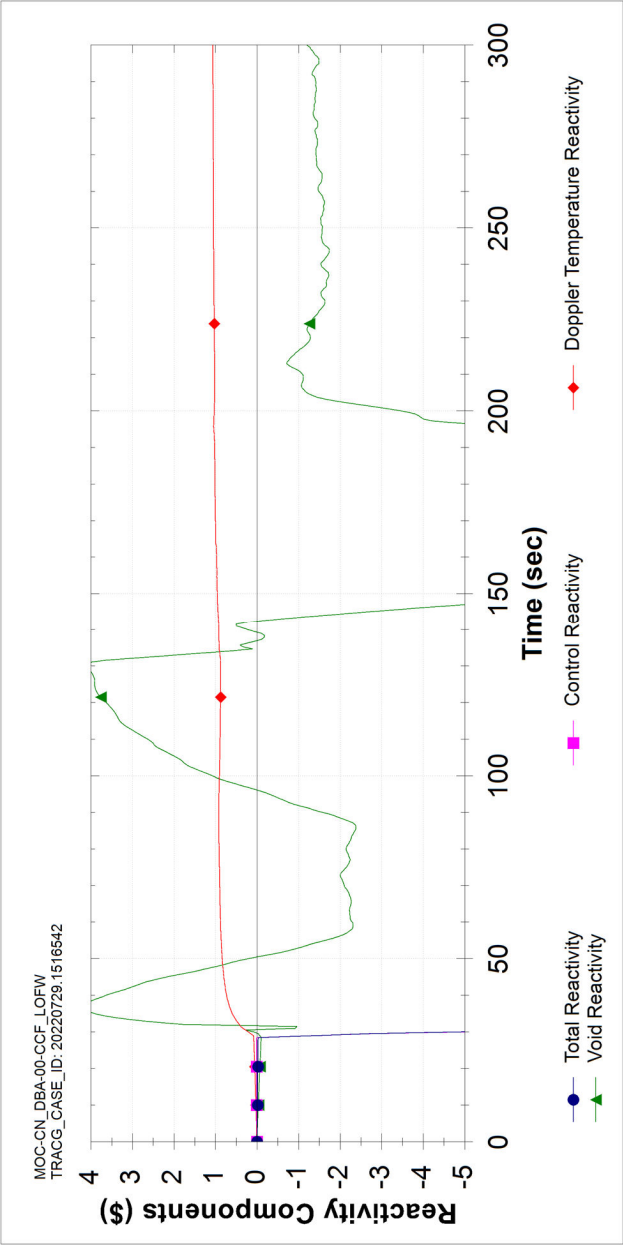


Figure 15.5-91: Loss of Feedwater Flow (DBA)

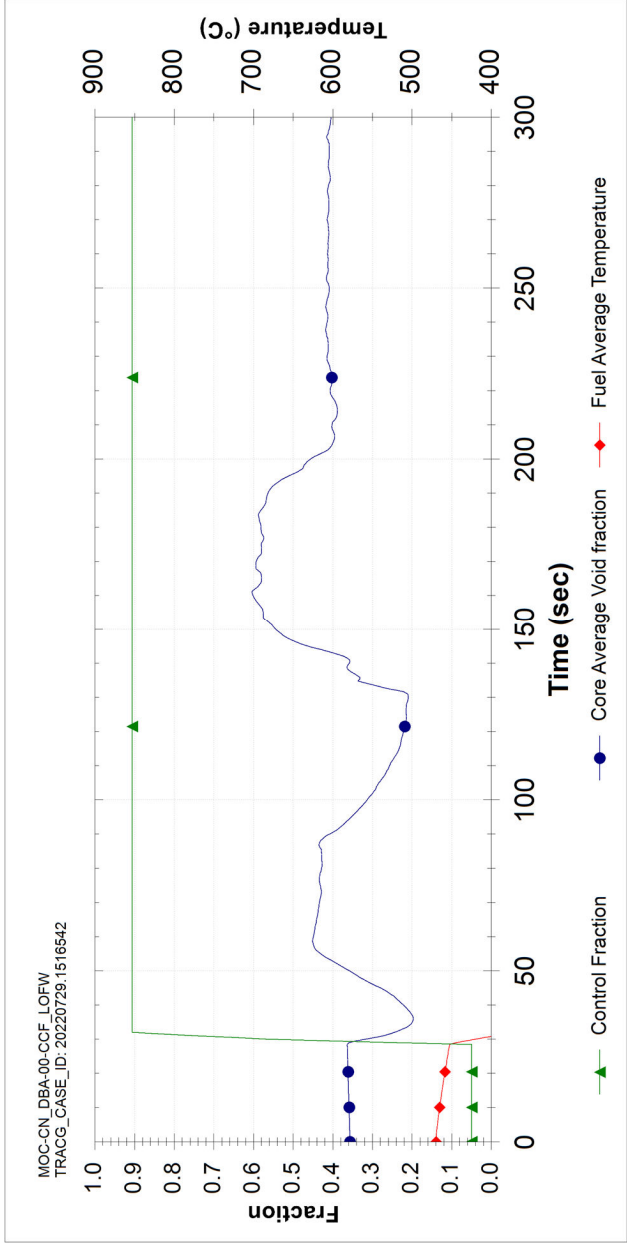


Figure 15.5-92: Loss of Feedwater Flow (DBA)

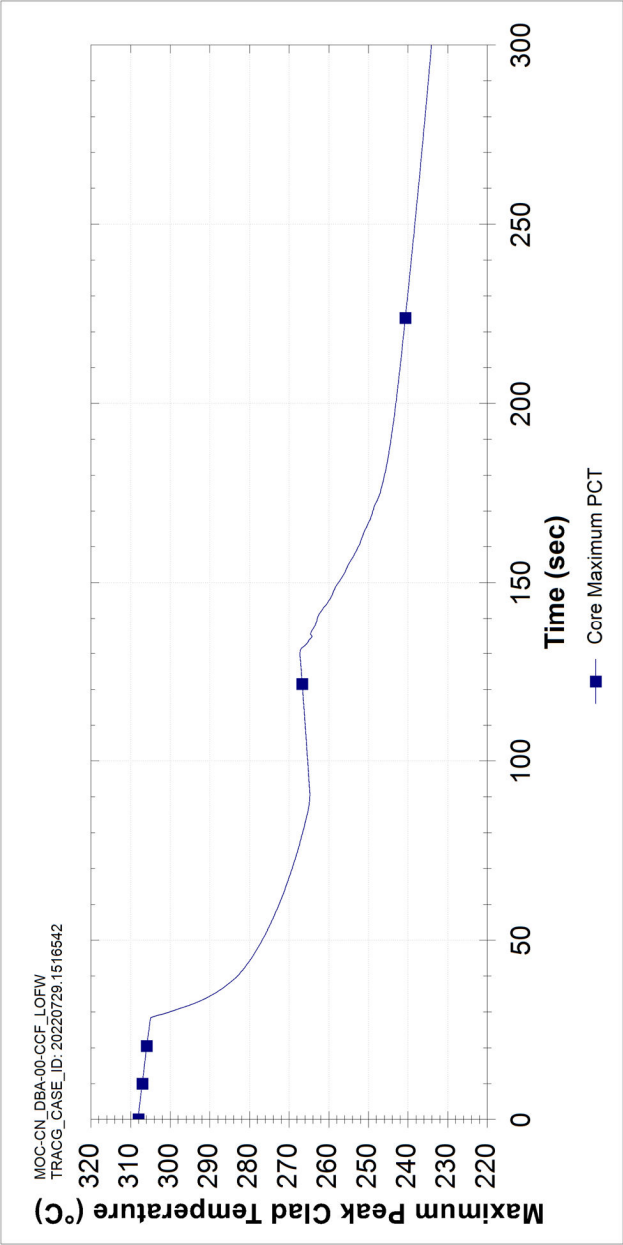


Figure 15.5-93: Loss of Feedwater Flow (DBA)

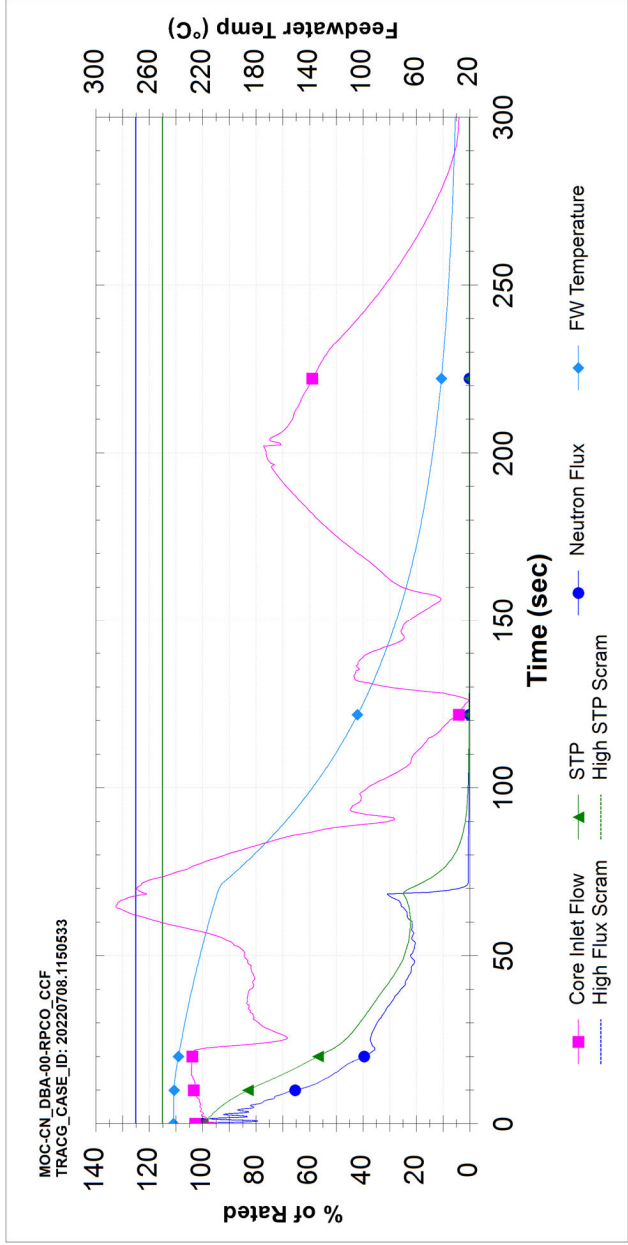


Figure 15.5-94: RPV Pressure Control Open (DBA)

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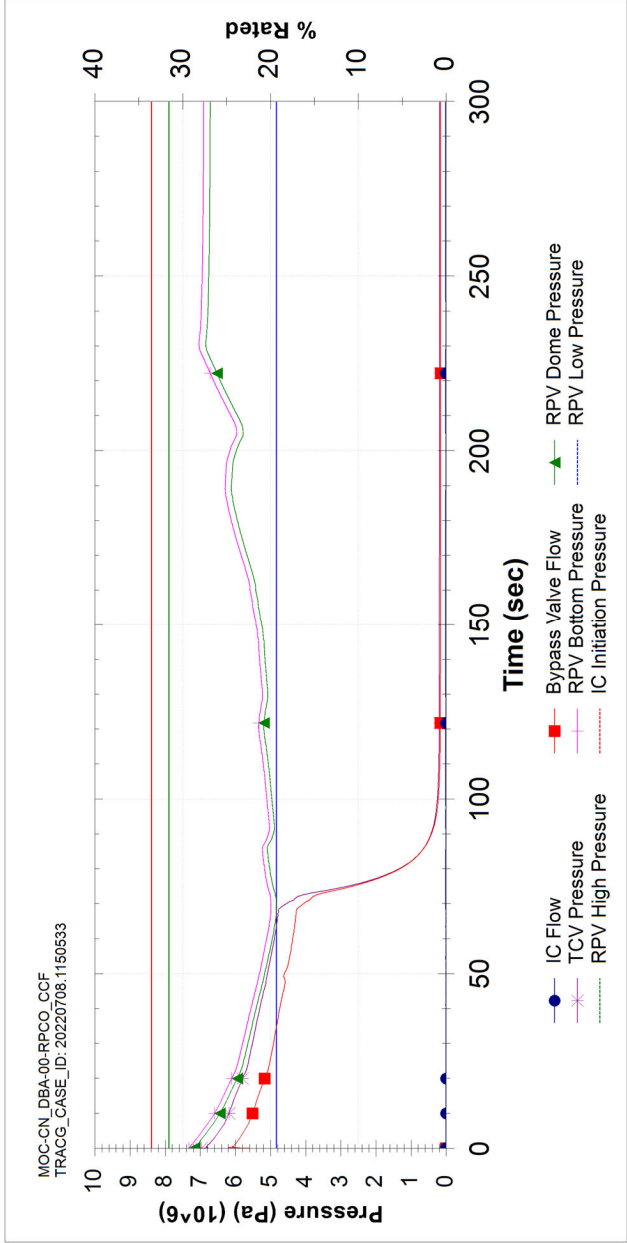


Figure 15.5-95: RPV Pressure Control Open (DBA)

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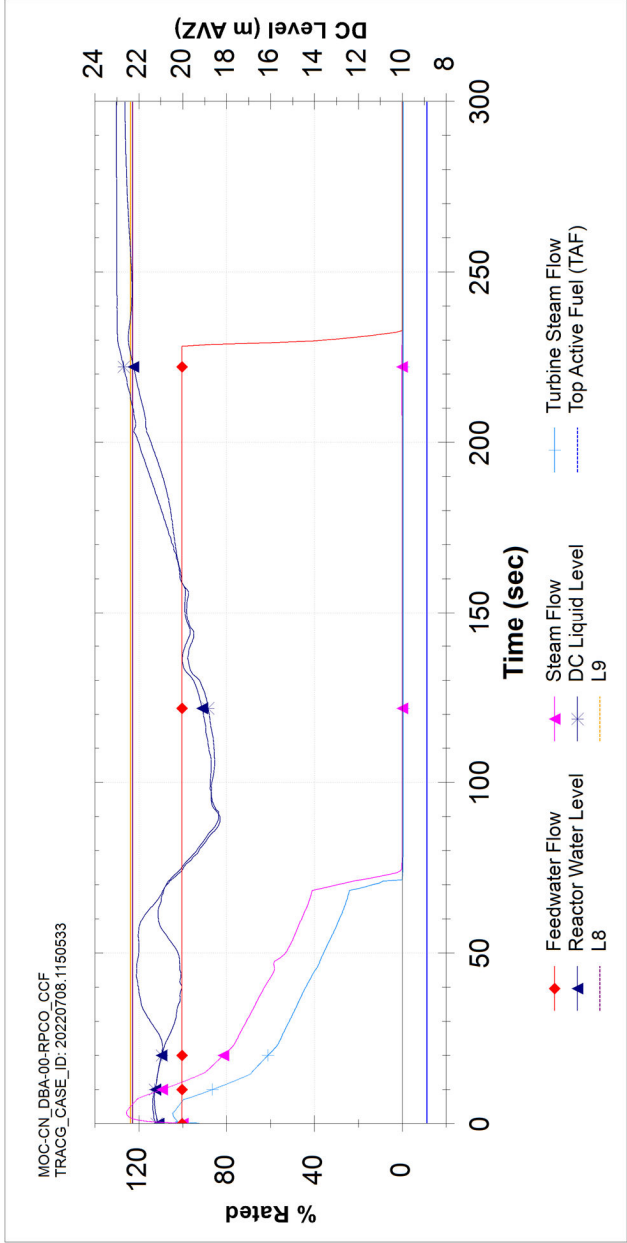


Figure 15.5-96: RPV Pressure Control Open (DBA)

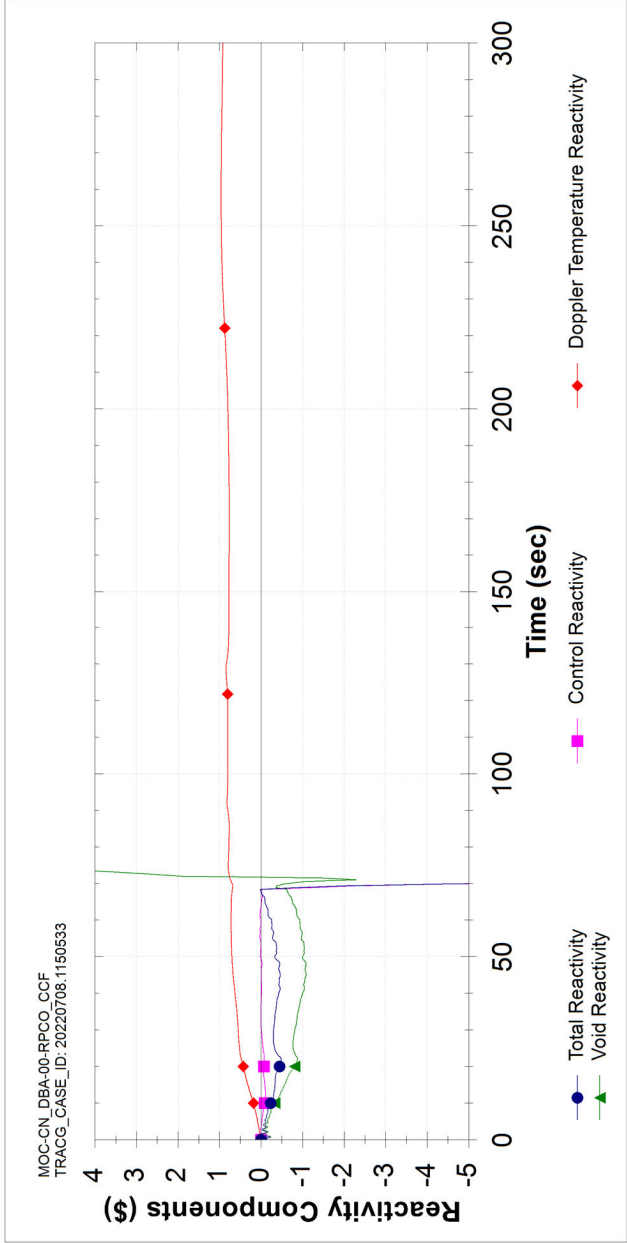


Figure 15.5-97: RPV Pressure Control Open (DBA)

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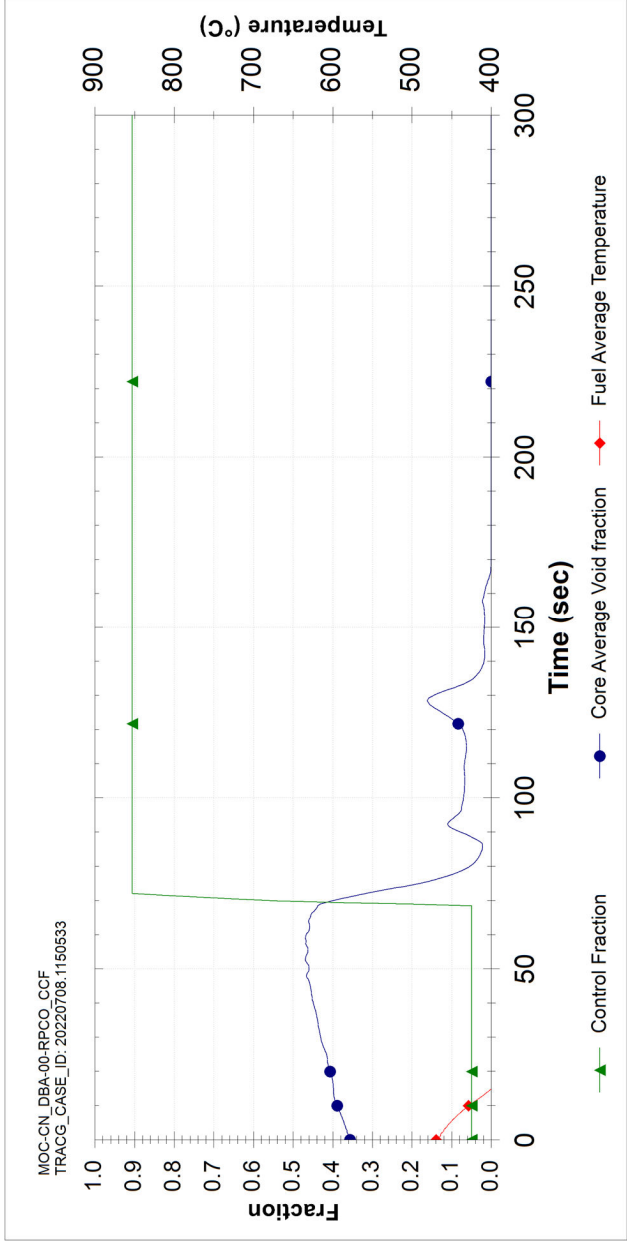


Figure 15.5-98: RPV Pressure Control Open (DBA)

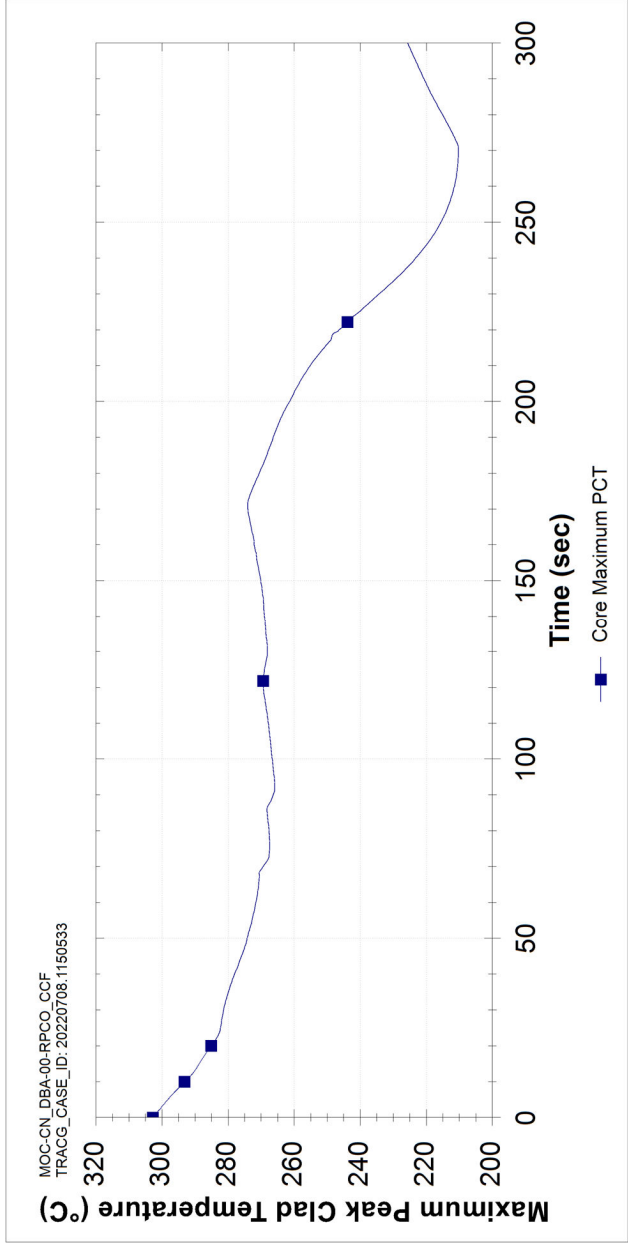


Figure 15.5-99: RPV Pressure Control Open (DBA)

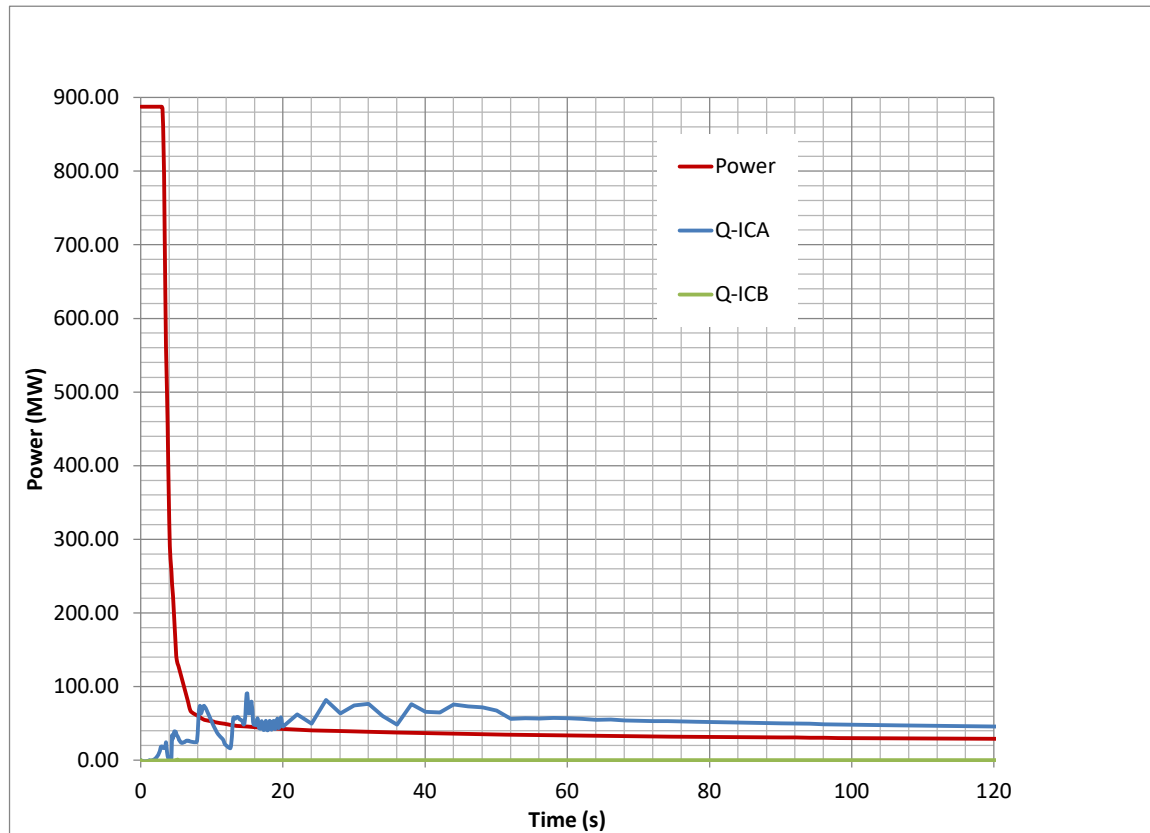


Figure 15.5-100: Reactor Power, Large Main Steam Pipe Break, Conservative Case

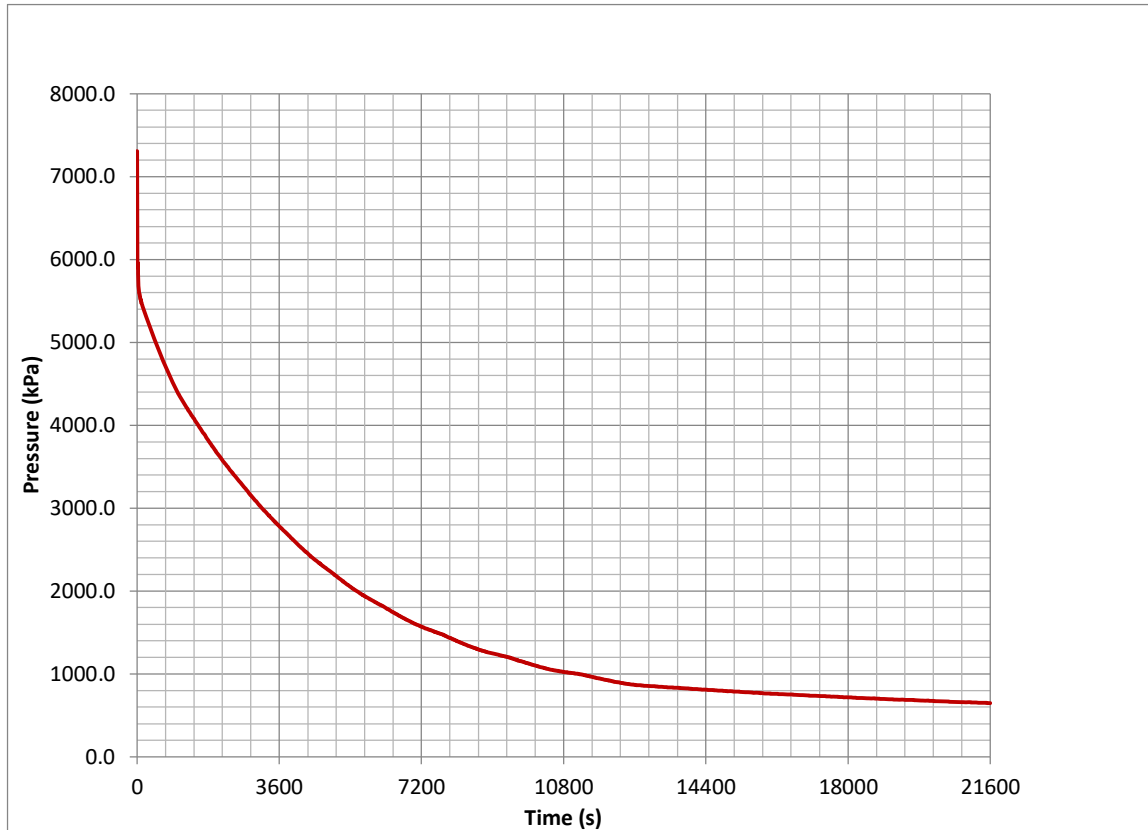


Figure 15.5-101: Reactor Pressure, Large Main Steam Pipe Break, Conservative Case

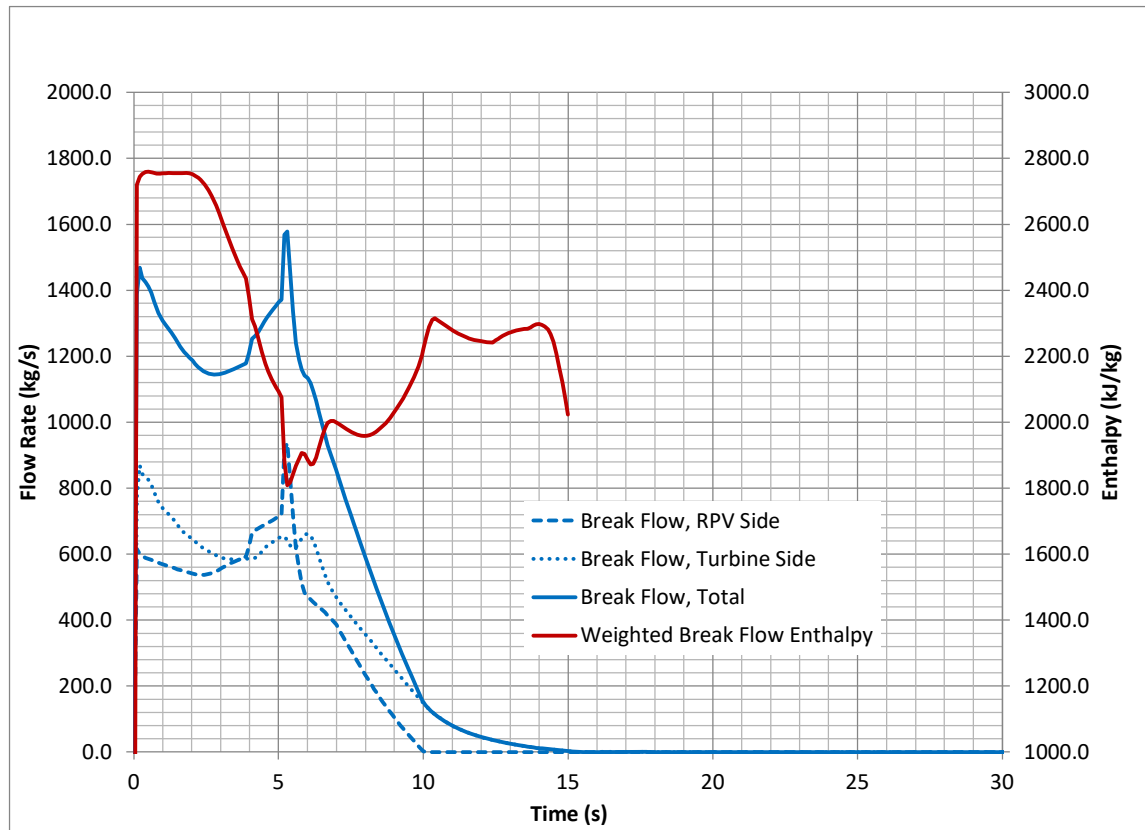


Figure 15.5-102: Break Flow Rate and Enthalpy, Large Main Steam Pipe Break, Conservative Case

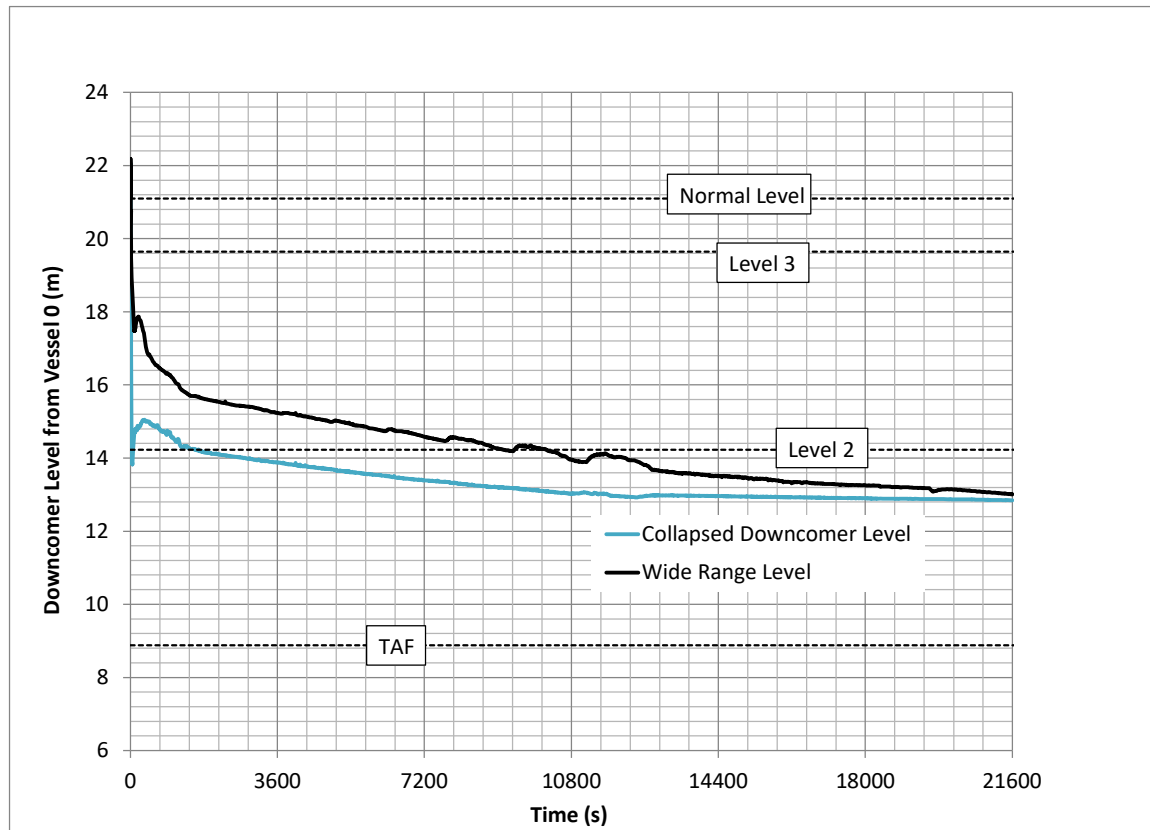


Figure 15.5-103: Reactor Water Level, Large Main Steam Pipe Break, Conservative Case

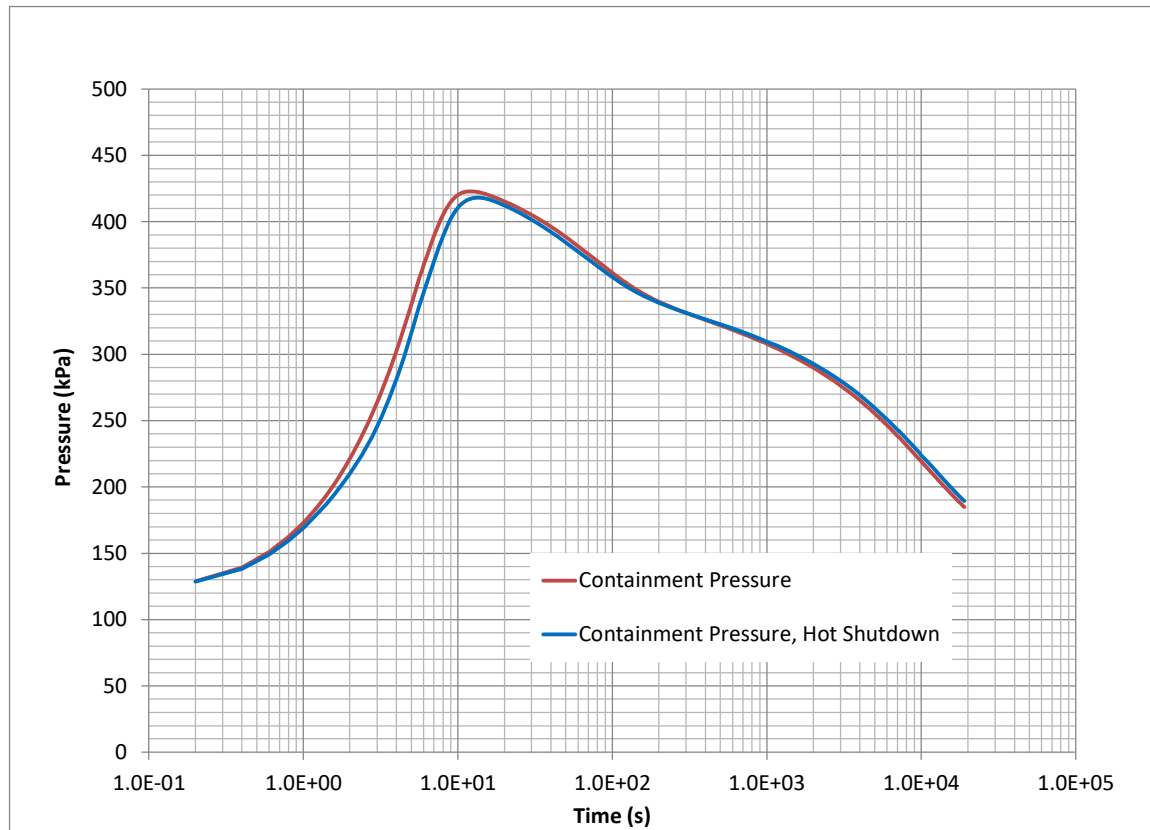


Figure 15.5-104: Containment Pressure, Large Main Steam Pipe Break, Conservative Case

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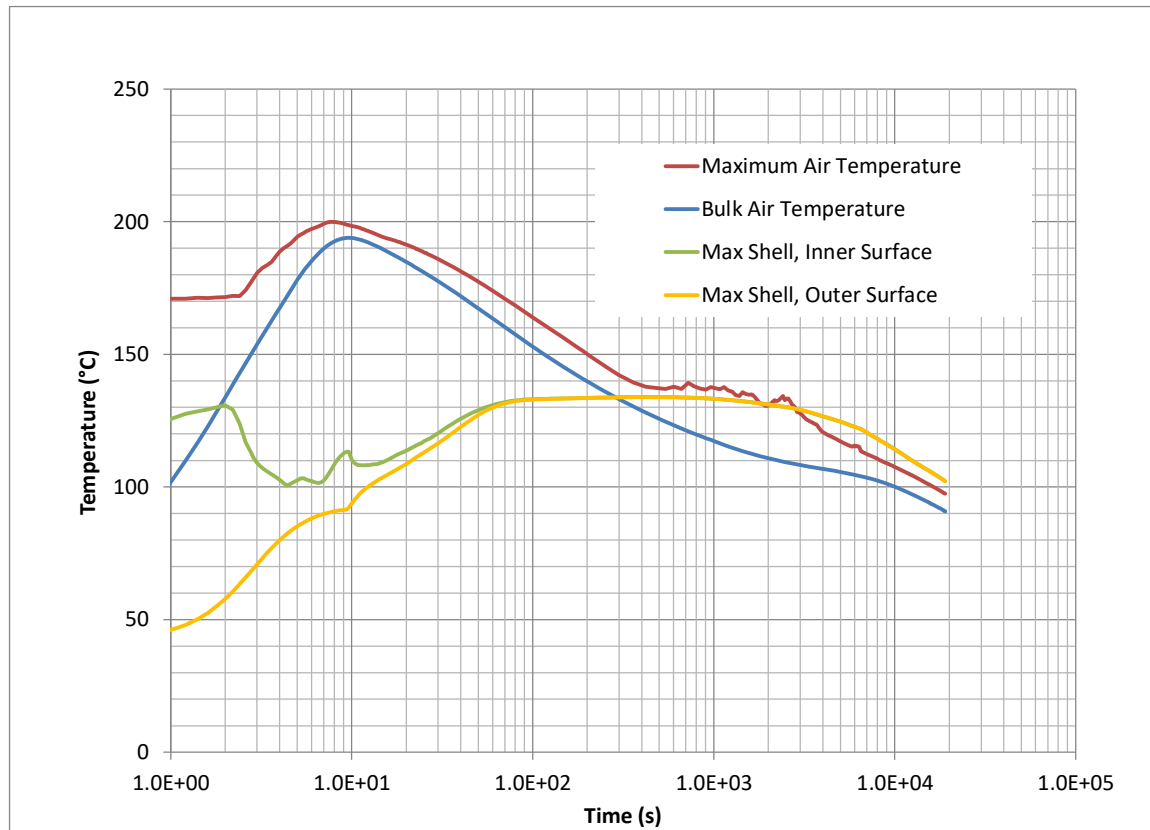


Figure 15.5-105: Containment Temperatures, Large Main Steam Pipe Break, Conservative Case

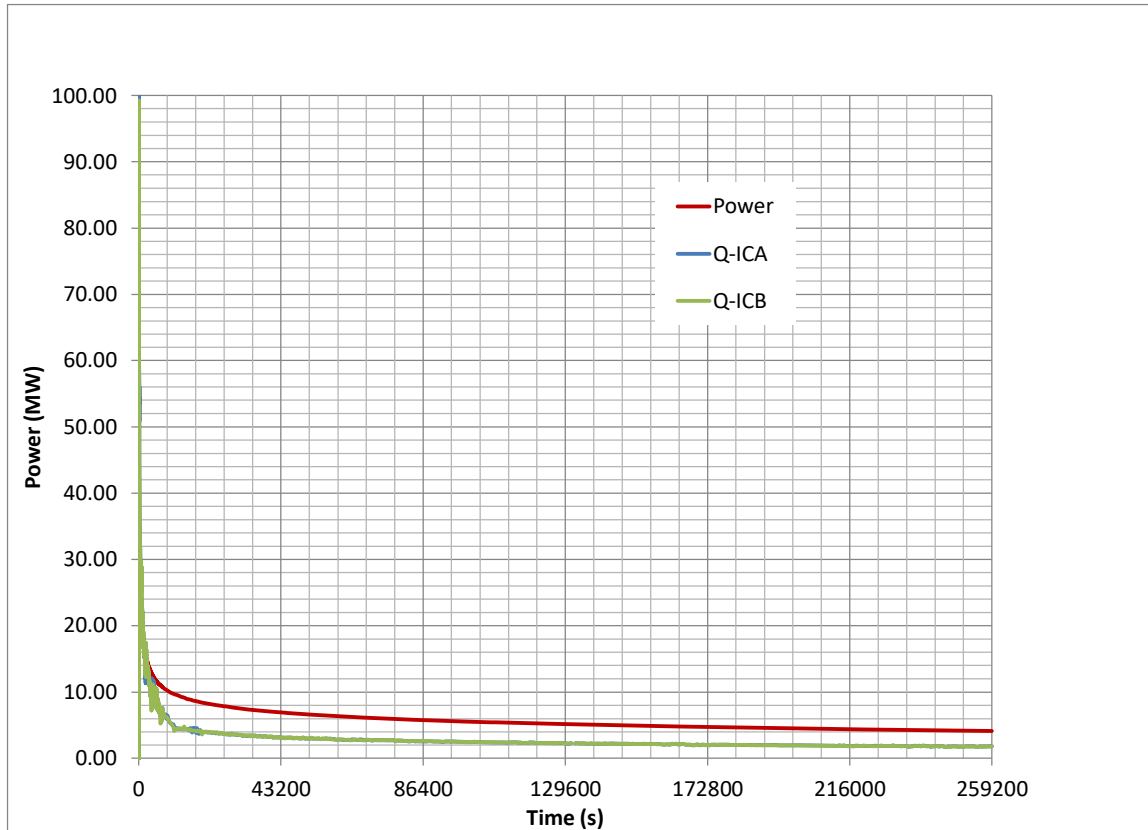


Figure 15.5-106: Reactor Power, Small Steam Break With LOPP, Conservative Case

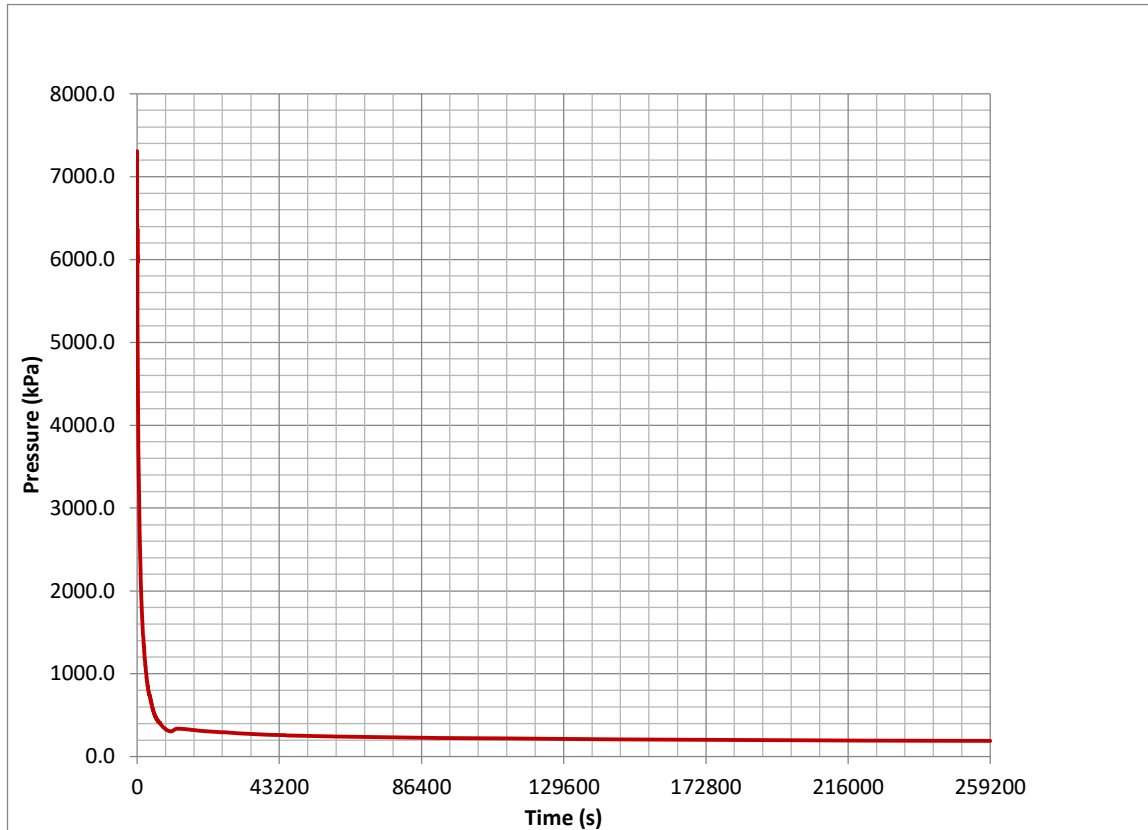


Figure 15.5-107: Reactor Pressure, Small Steam Pipe Break With LOPP, Conservative Case

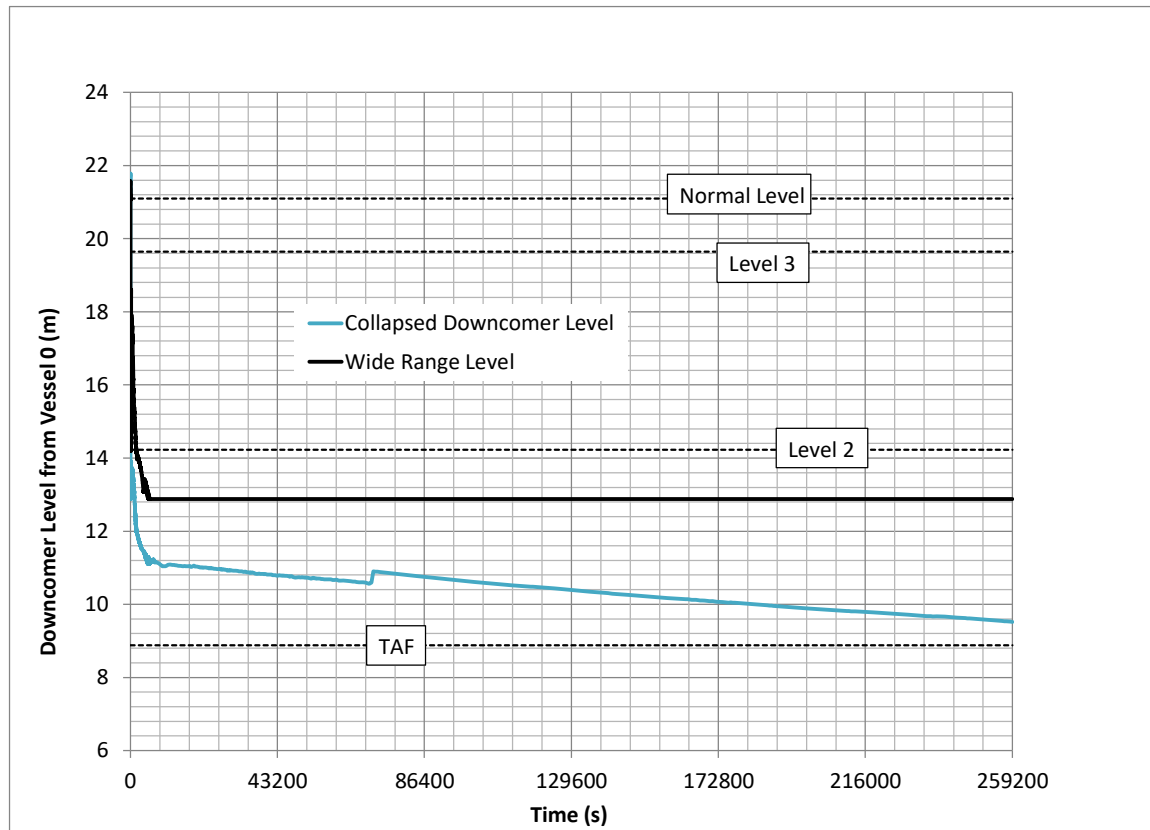


Figure 15.5-108: Reactor Water Level, Small Steam Pipe Break With LOPP, Conservative Case

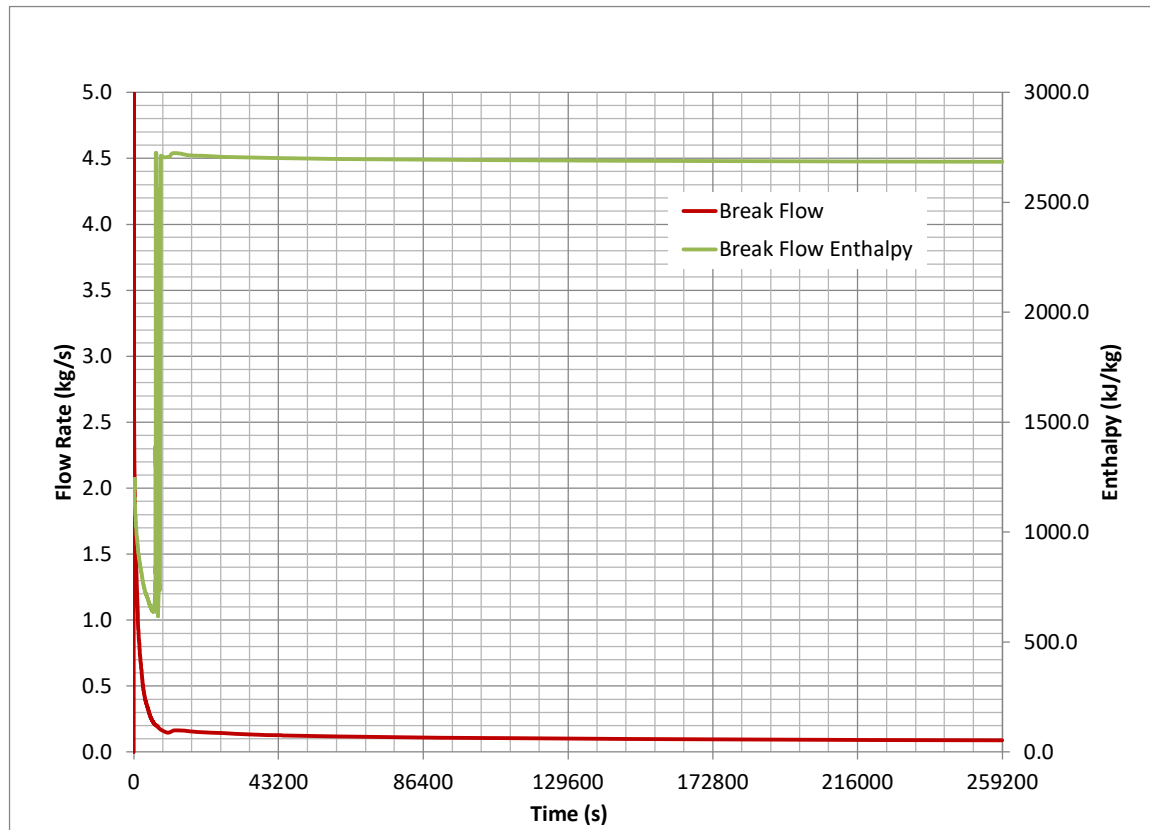


Figure 15.5-109: Break Flow Rate and Enthalpy, Small Steam Pipe Break With LOPP, Conservative Case

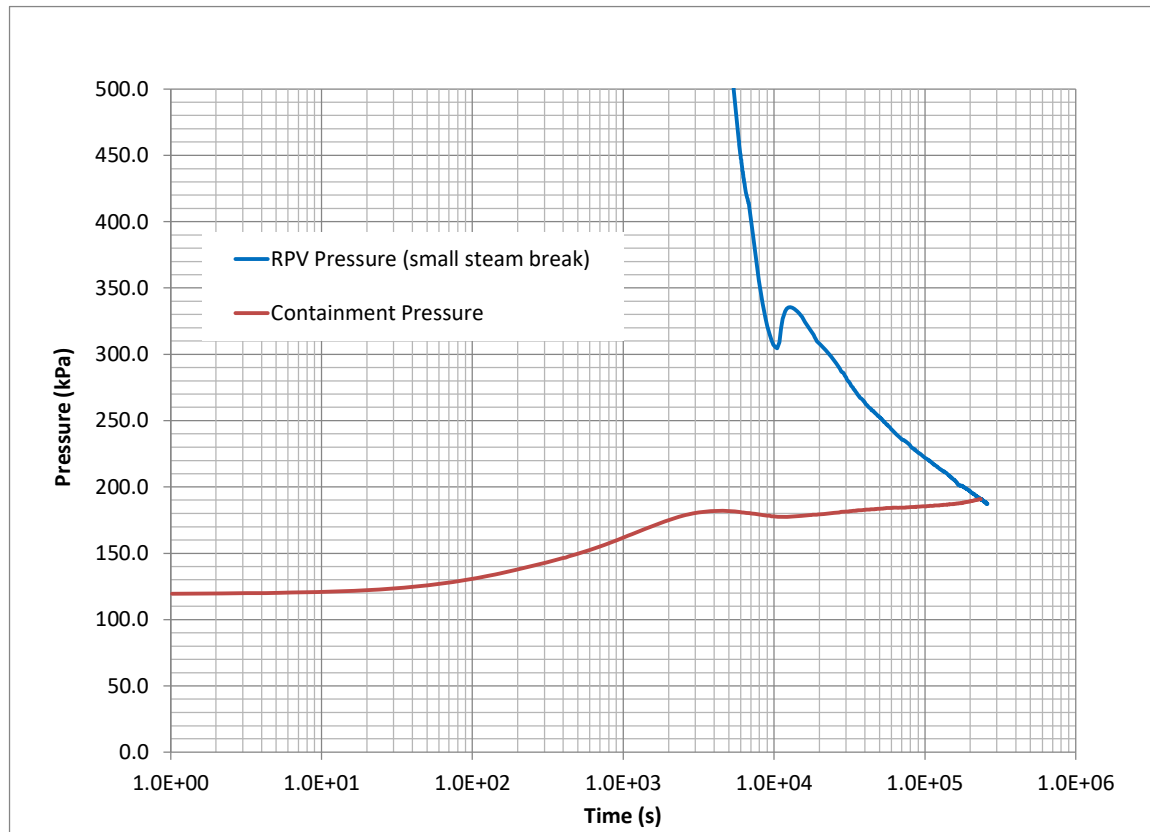


Figure 15.5-110: Containment Pressure, Small Steam Pipe Break with LOPP 2 ICS Trains, Conservative Case

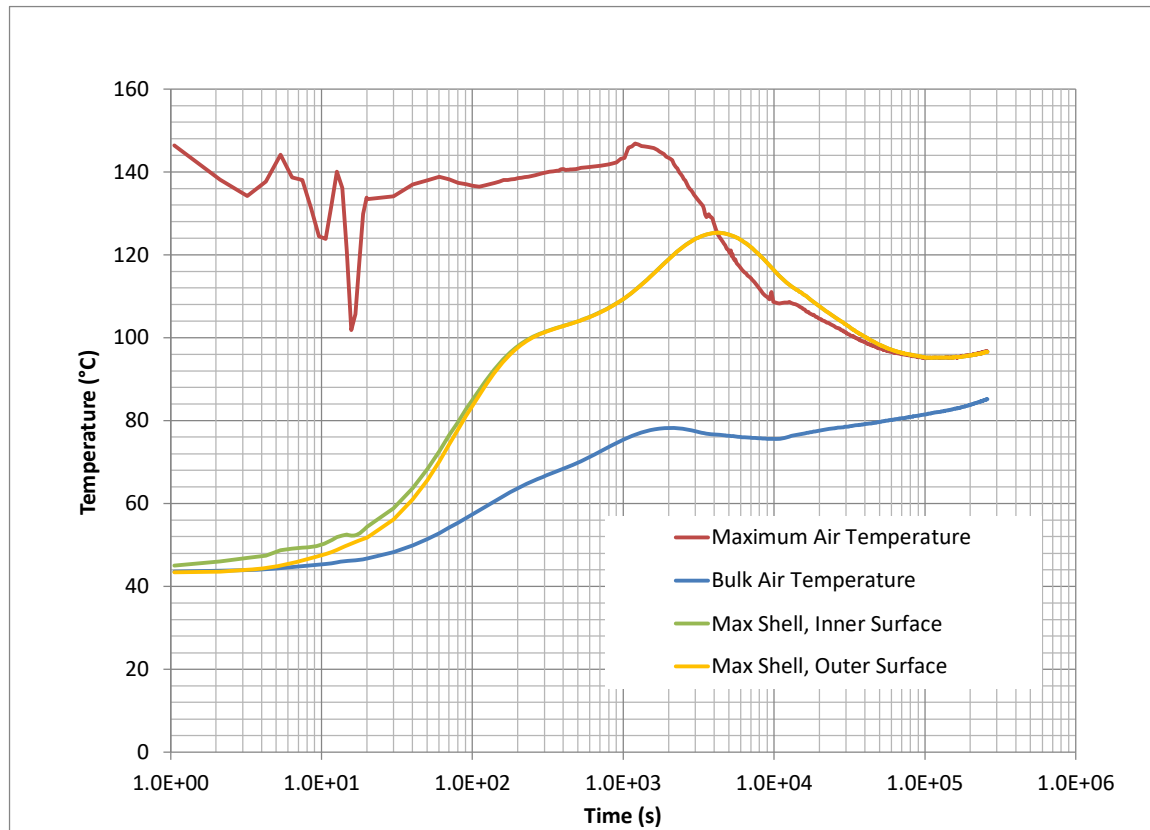


Figure 15.5-111: Containment Temperature, Small Steam Pipe Break, Conservative Case

Figure 15.5-112: Not Used

Figure 15.5-113: Not Used

Figure 15.5-114: Not Used

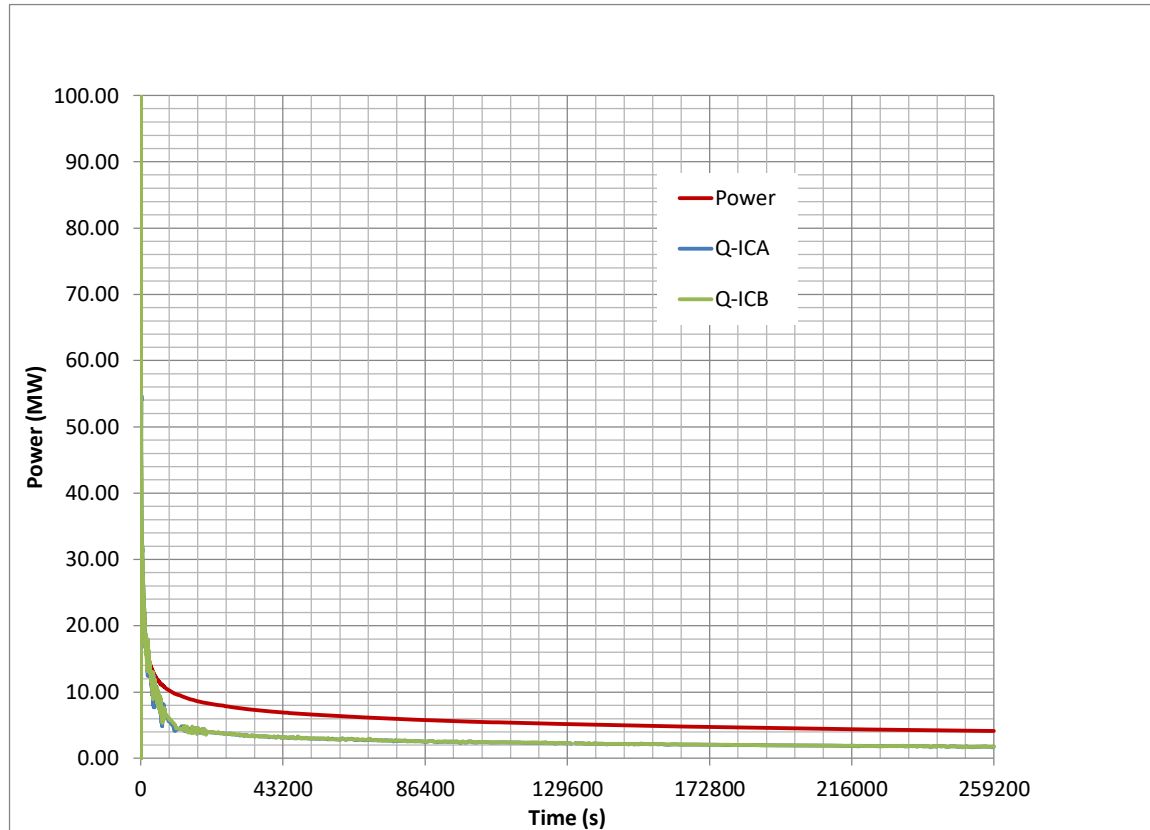


Figure 15.5-115: Reactor Power, Small Liquid Pipe Break, Conservative Case

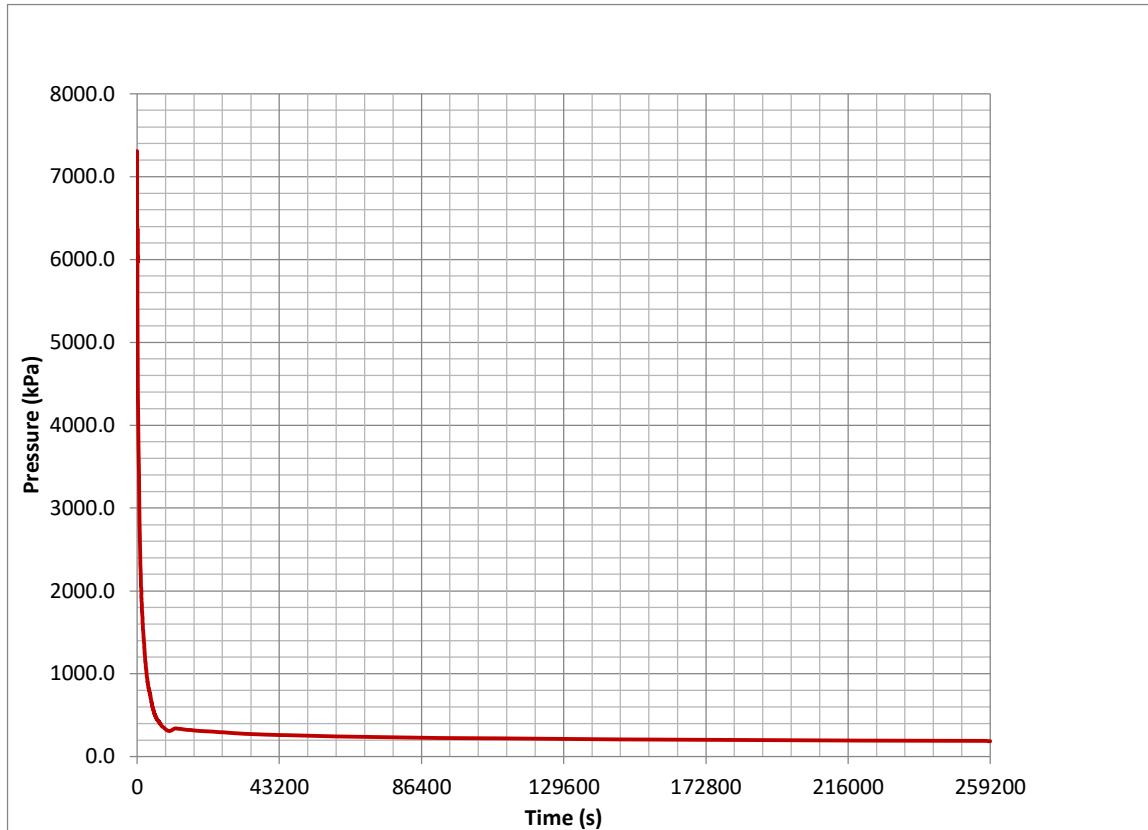


Figure 15.5-116: Reactor Pressure, Small Liquid Pipe Break, Conservative Case

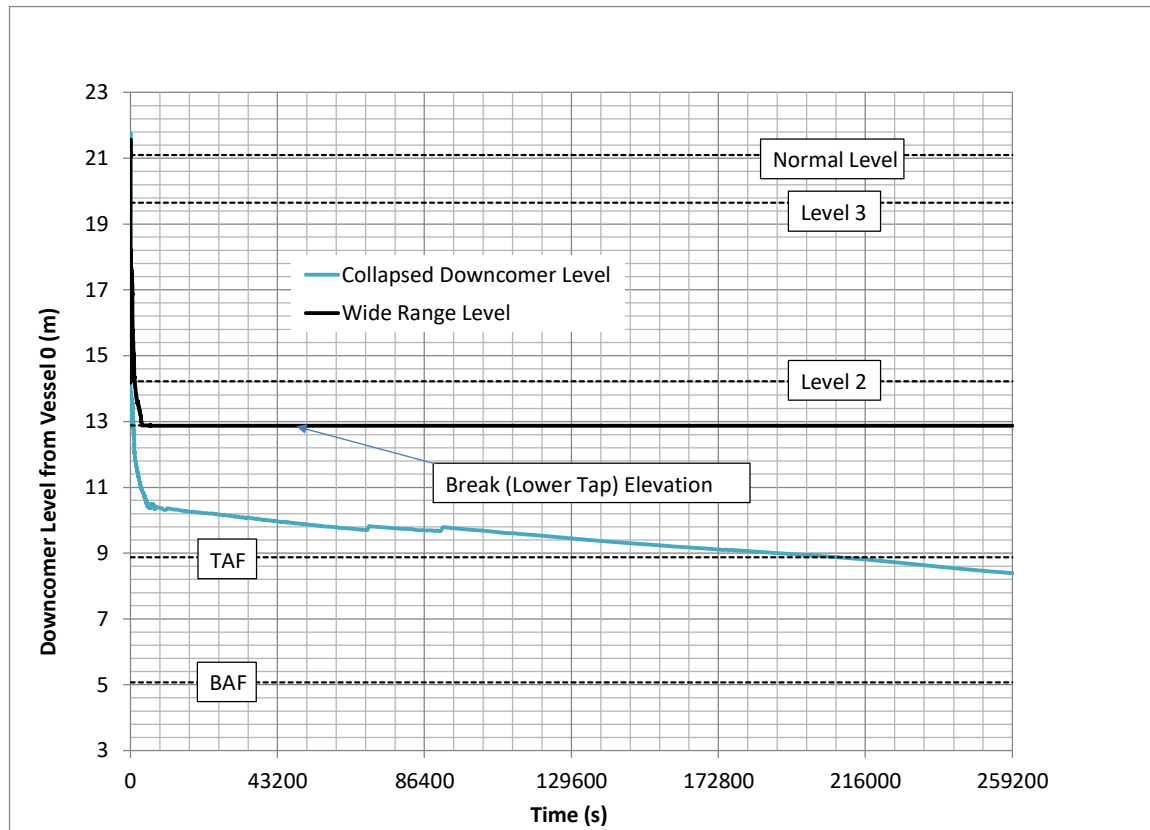


Figure 15.5-117: Reactor Water Level, Small Liquid Break, Conservative Case

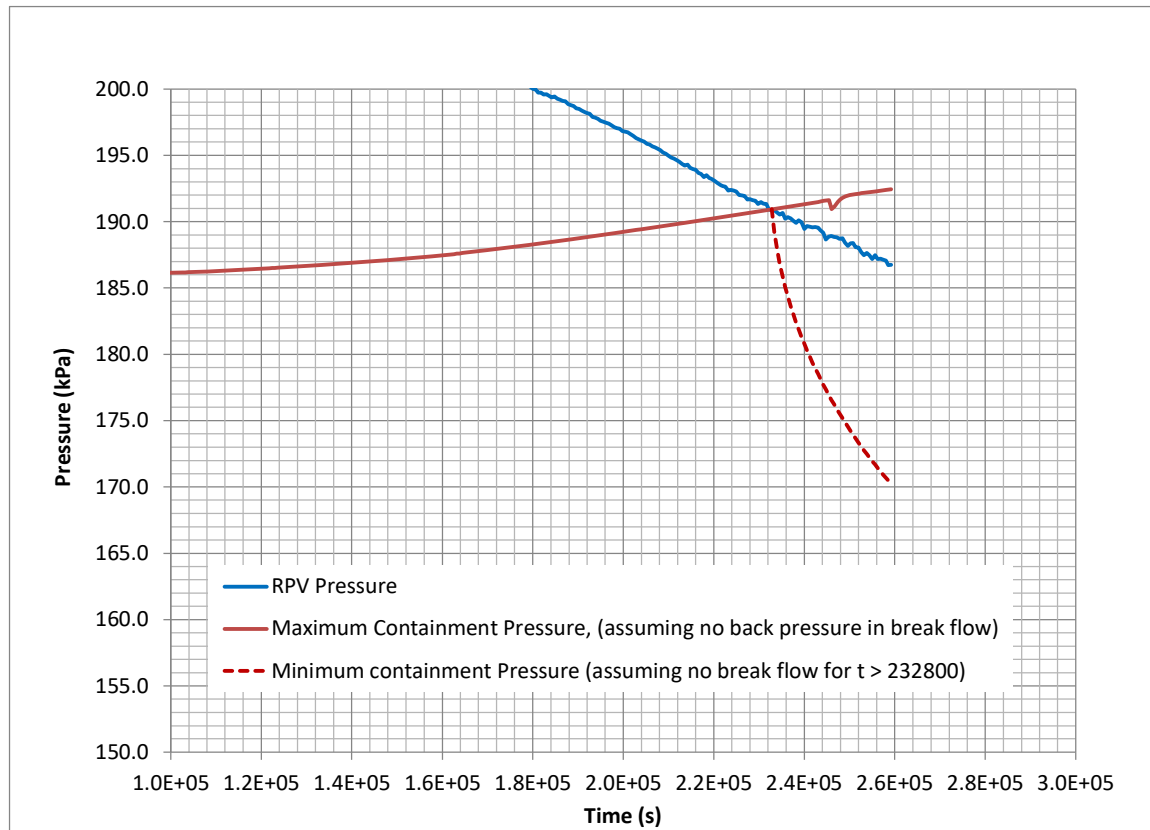


Figure 15.5-118: Containment Pressure After RVP Depressurization, Small Liquid Pipe Break, Conservative Case

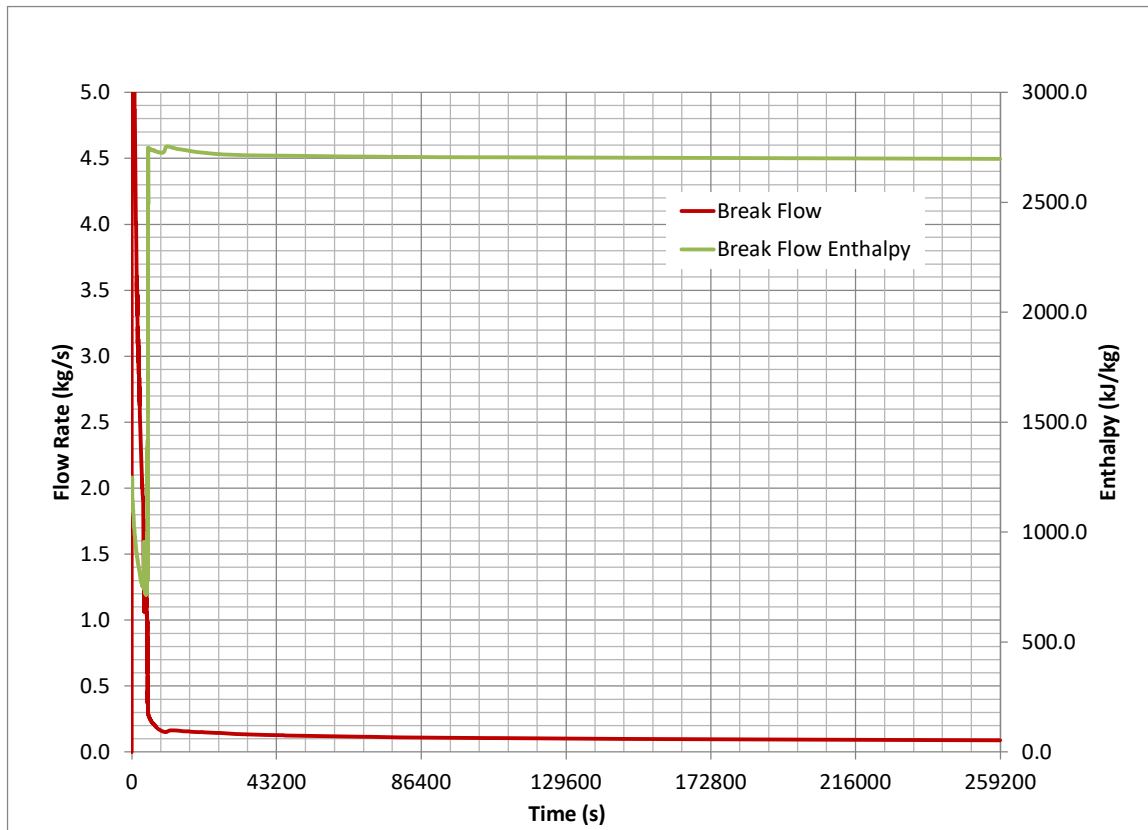


Figure 15.5-119: Break Flow Rate and Enthalpy, Small Liquid Pipe Break, Conservative Case

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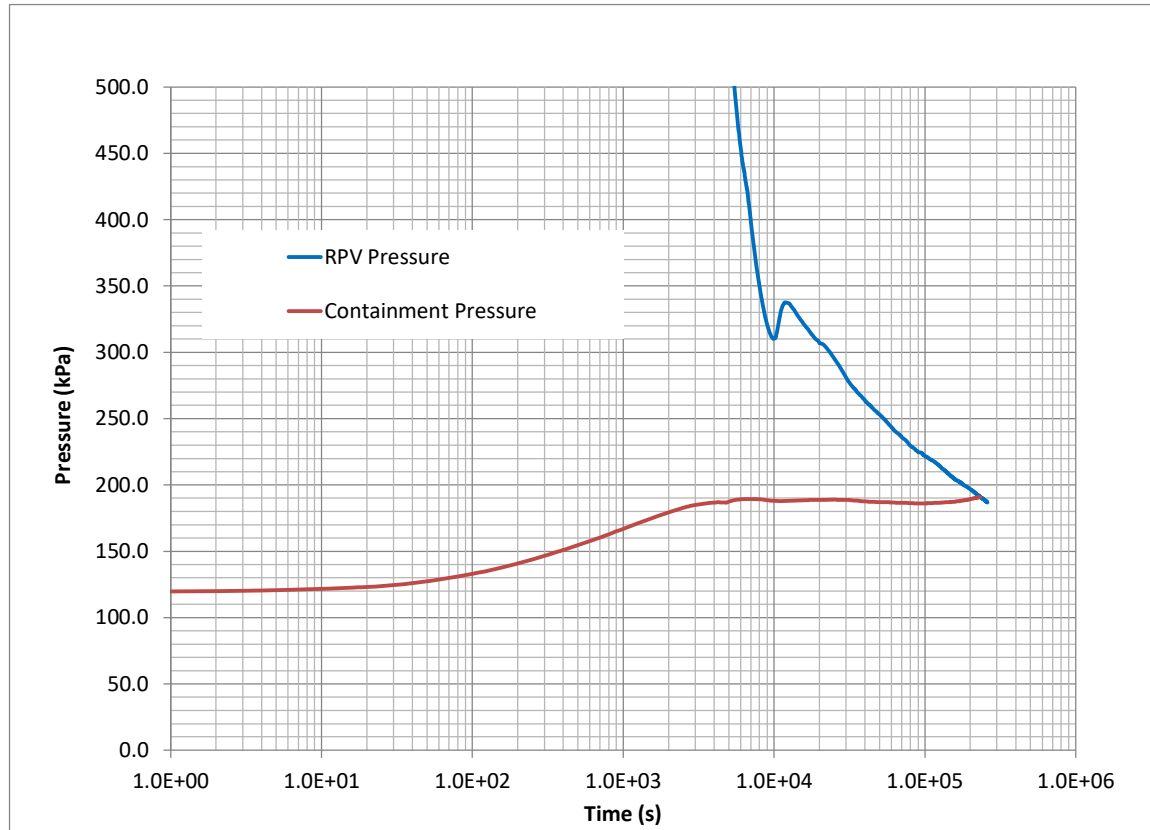


Figure 15.5-120: Containment Pressure, Small Liquid Pipe Break, Conservative Case

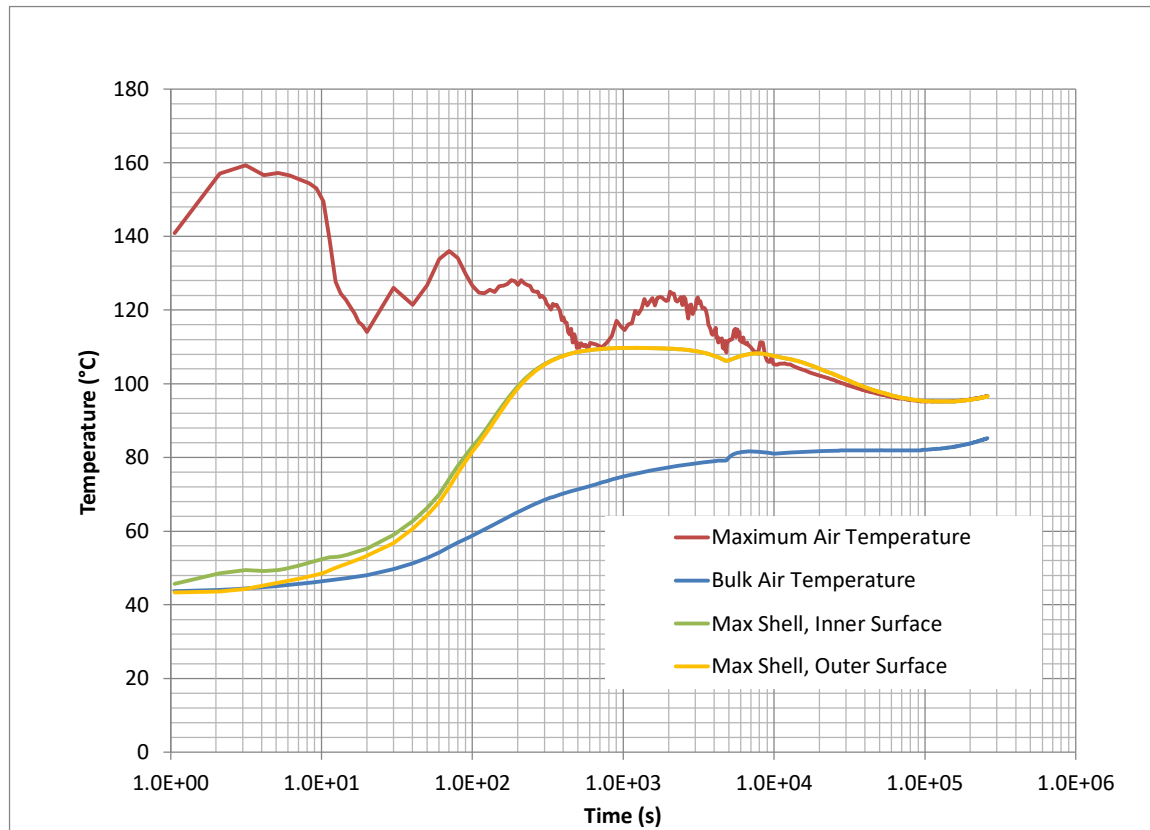


Figure 15.5-121: Containment Temperature, Small Liquid Pipe Break, Conservative Case

Figure 15.5-122: Not Used

Figure 15.5-123: Not Used

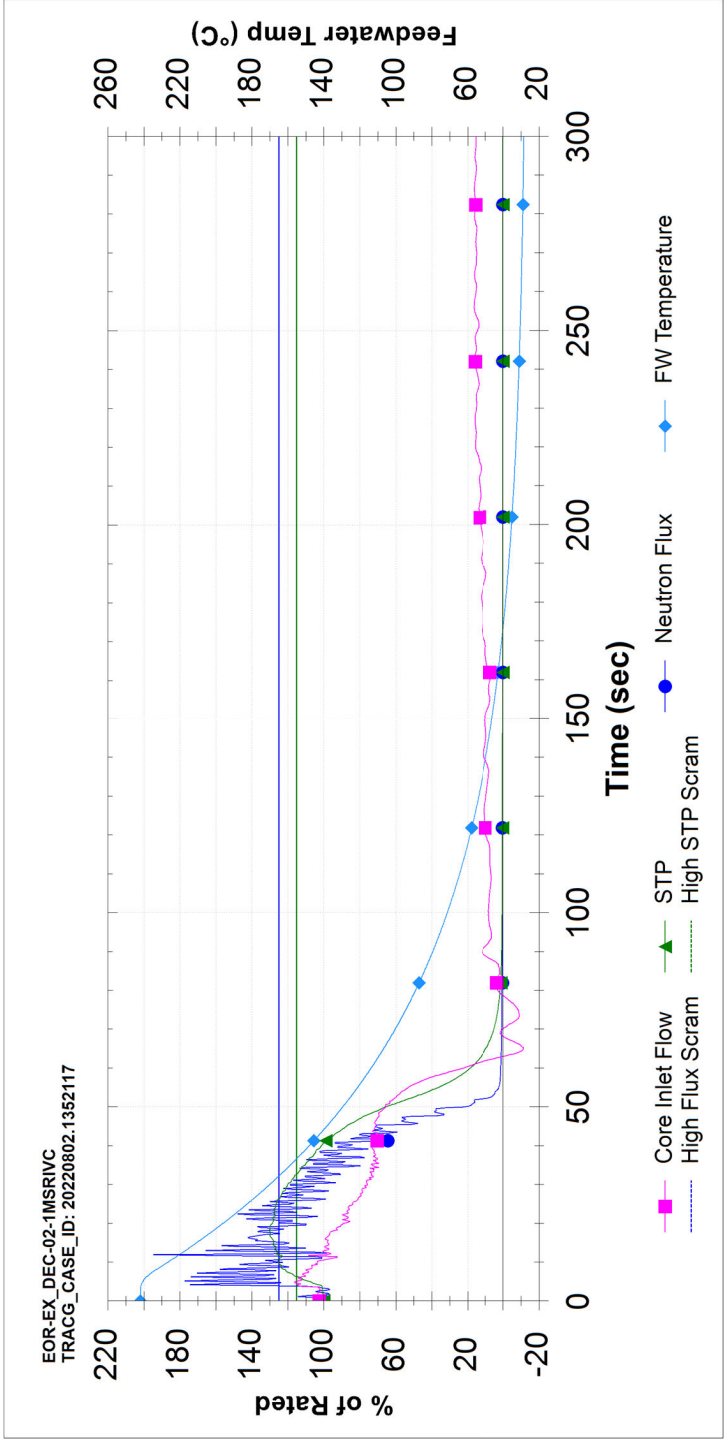


Figure 15.5-124: Closure of One Main Steam Reactor Isolation
Valve with Failure to Scram (DEC)

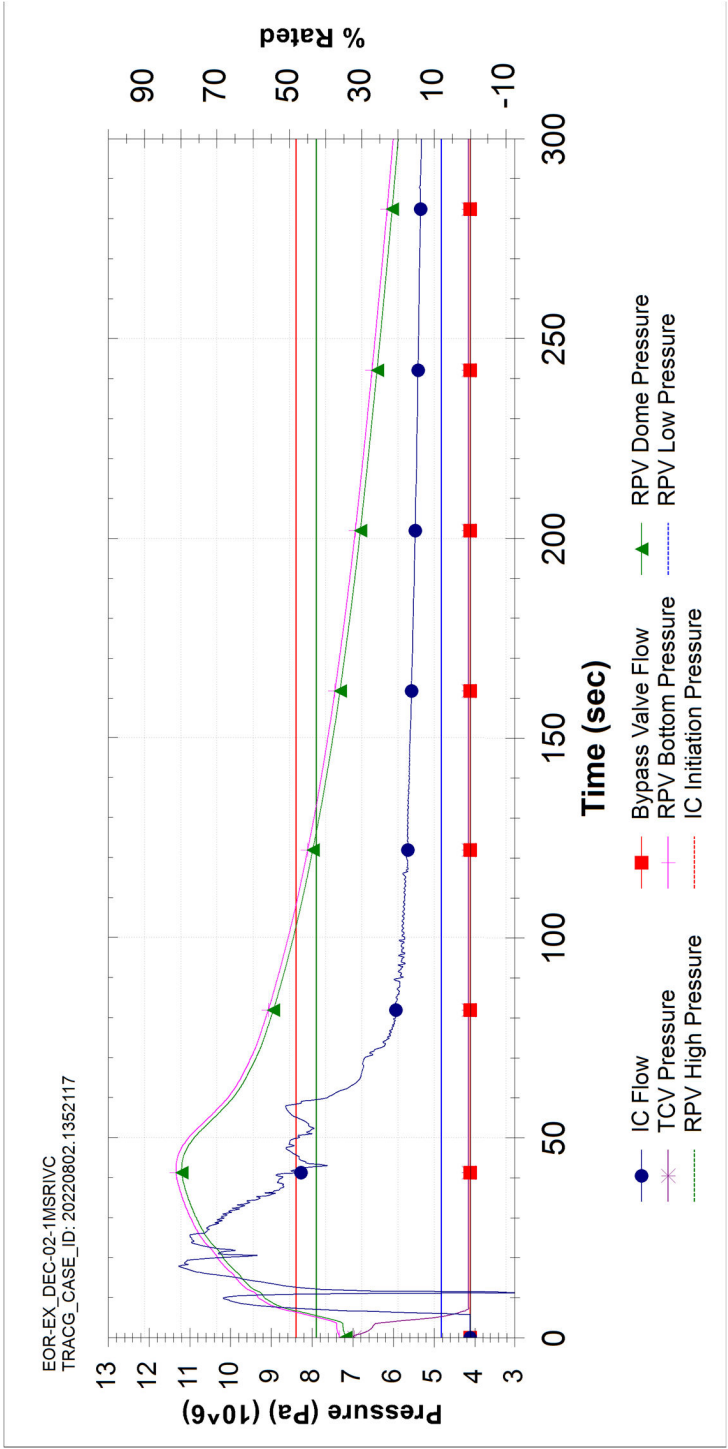


Figure 15.5-125: Closure of One Main Steam Reactor Isolation
Valve with Failure to Scram (DEC)

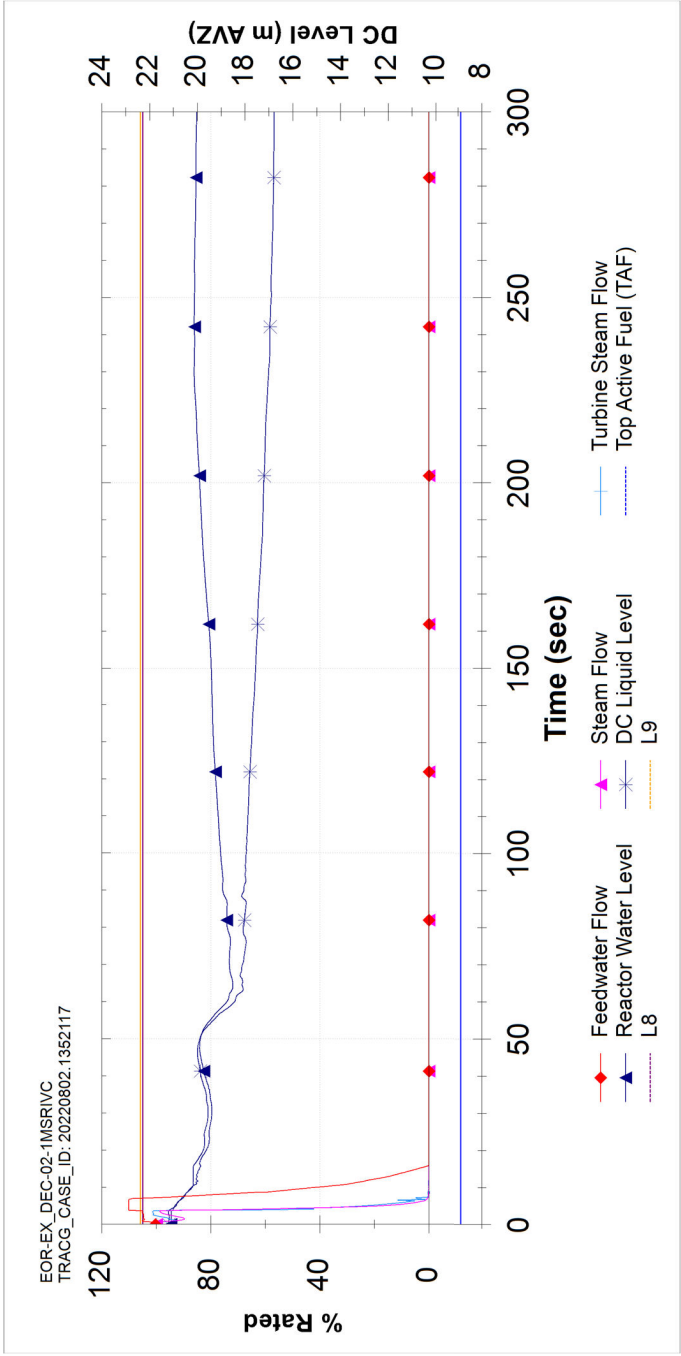
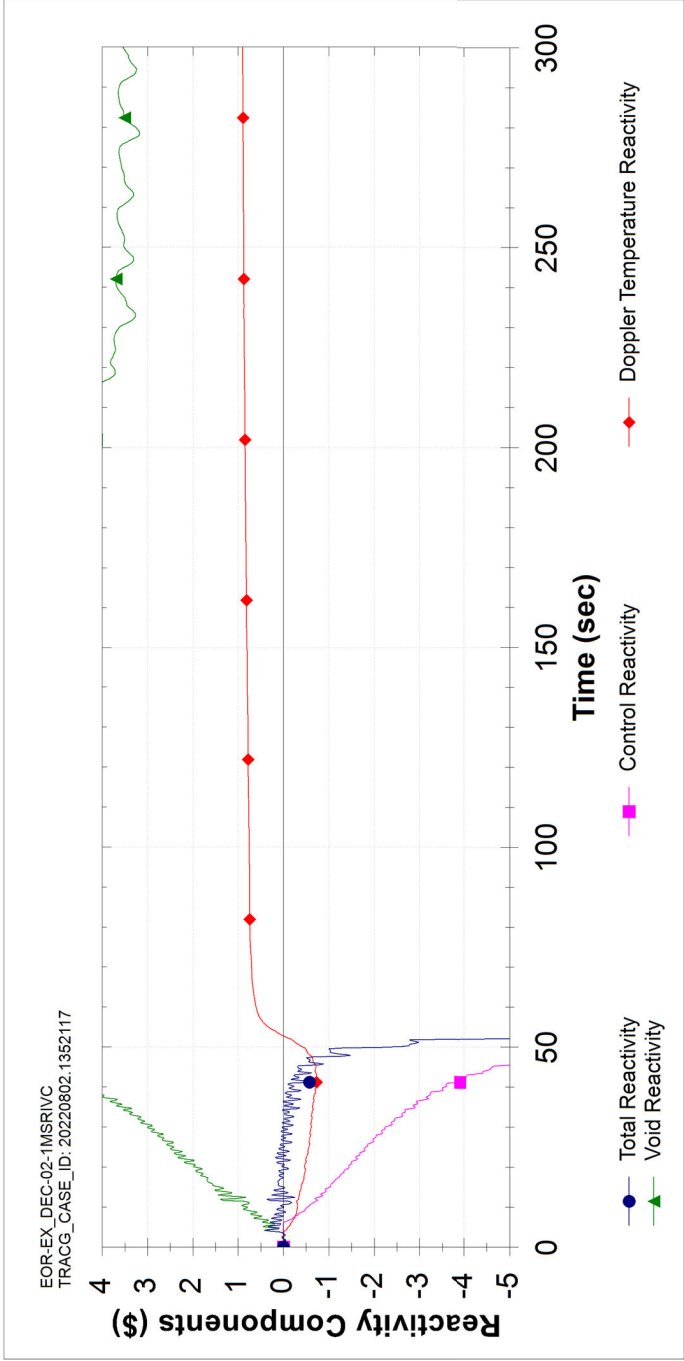
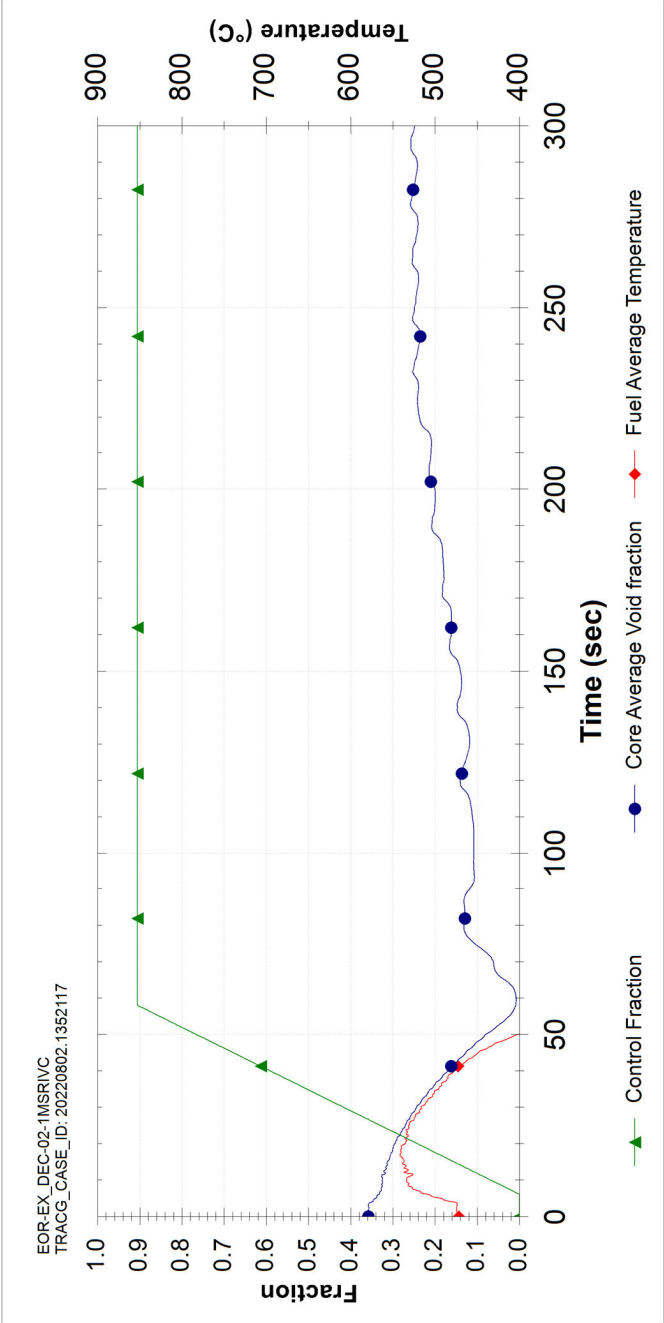


Figure 15.5-126: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)



**Figure 15.5-127: Closure of One Main Steam Reactor Isolation
Valve with Failure to Scram (DEC)**



**Figure 15.5-128: Closure of One Main Steam Reactor Isolation
Valve with Failure to Scram (DEC)**

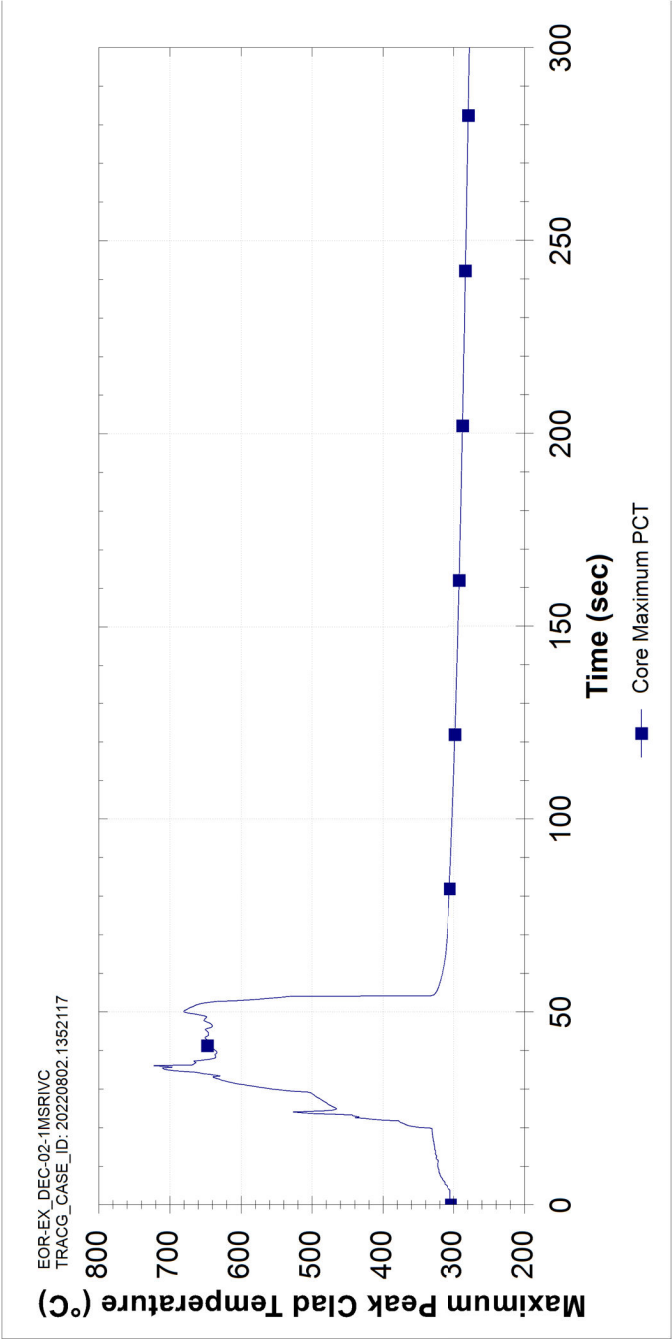


Figure 15.5-129: Closure of One Main Steam Reactor Isolation
Valve with Failure to Scram (DEC)

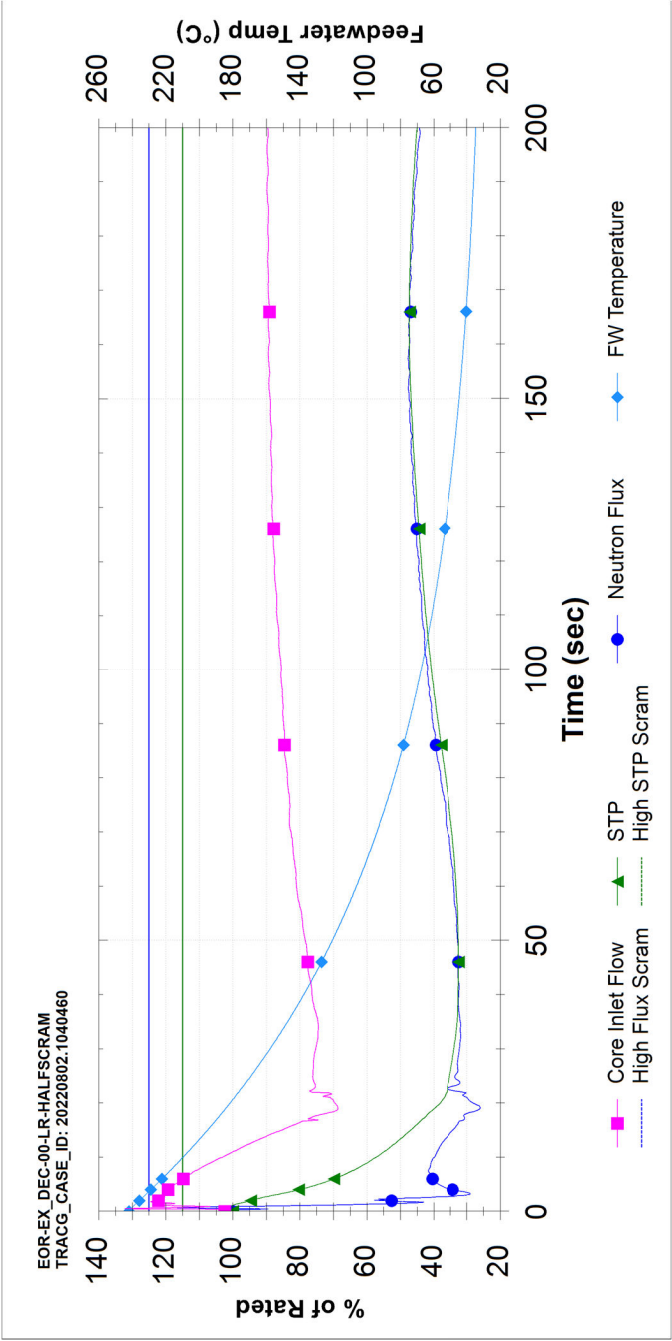


Figure 15.5-130: Complex Sequence Generator Load Rejection (DEC)

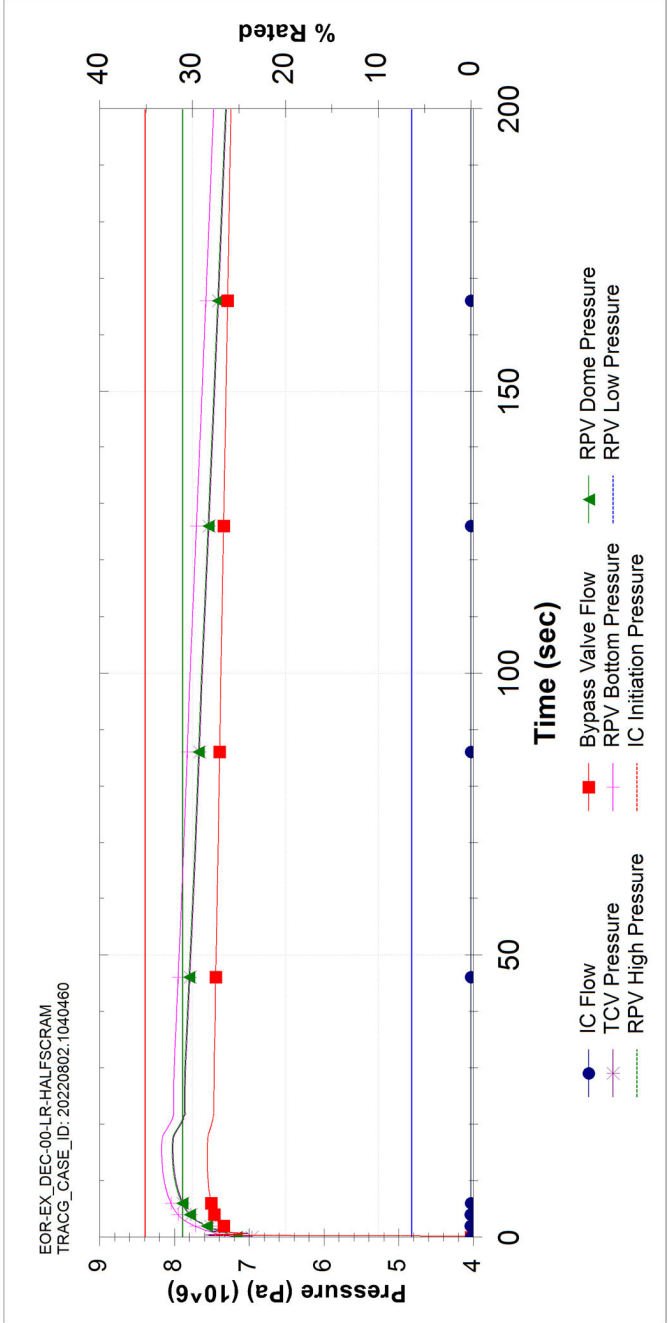


Figure 15.5-131: Complex Sequence Generator Load Rejection (DEC)

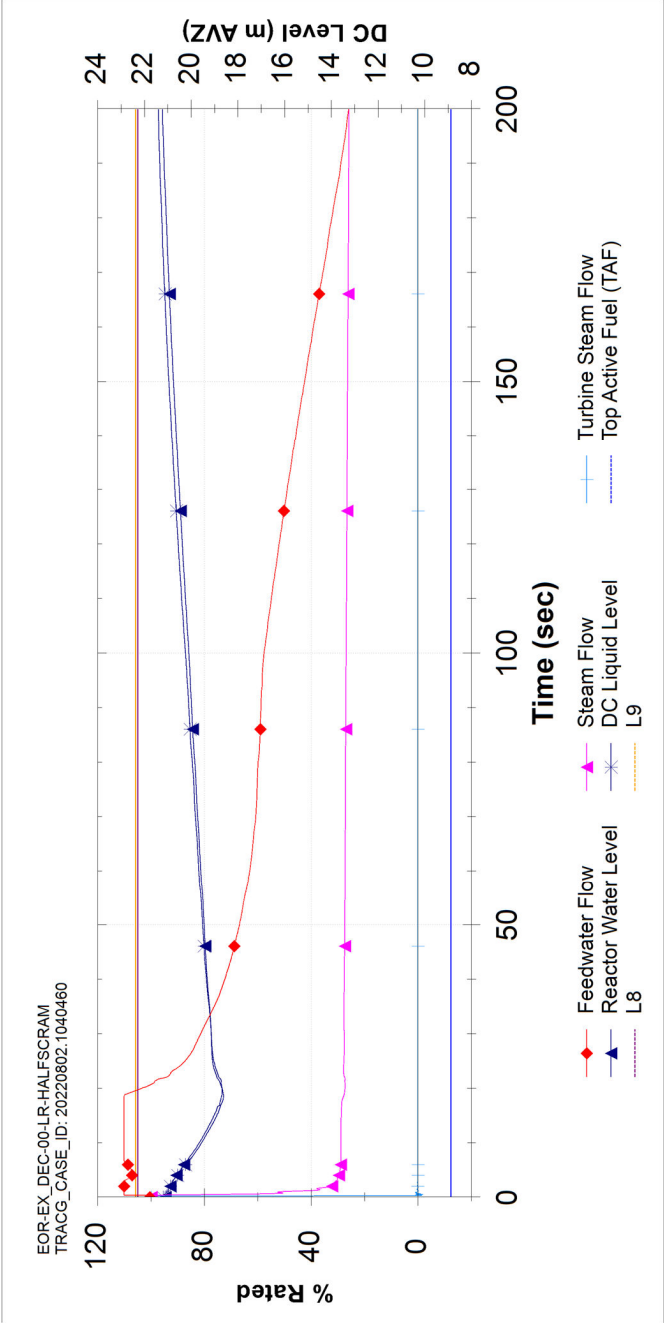


Figure 15.5-132: Complex Sequence Generator Load Rejection (DEC)

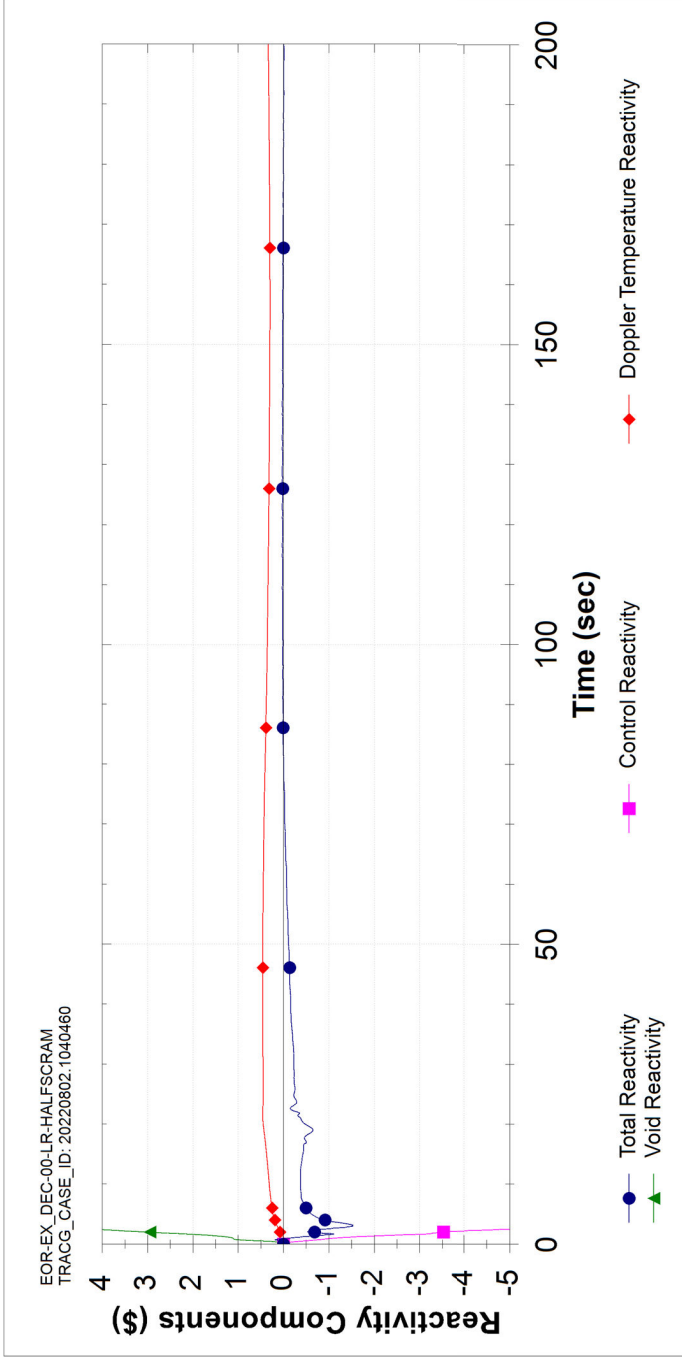


Figure 15.5-133: Complex Sequence Generator Load Rejection (DEC)

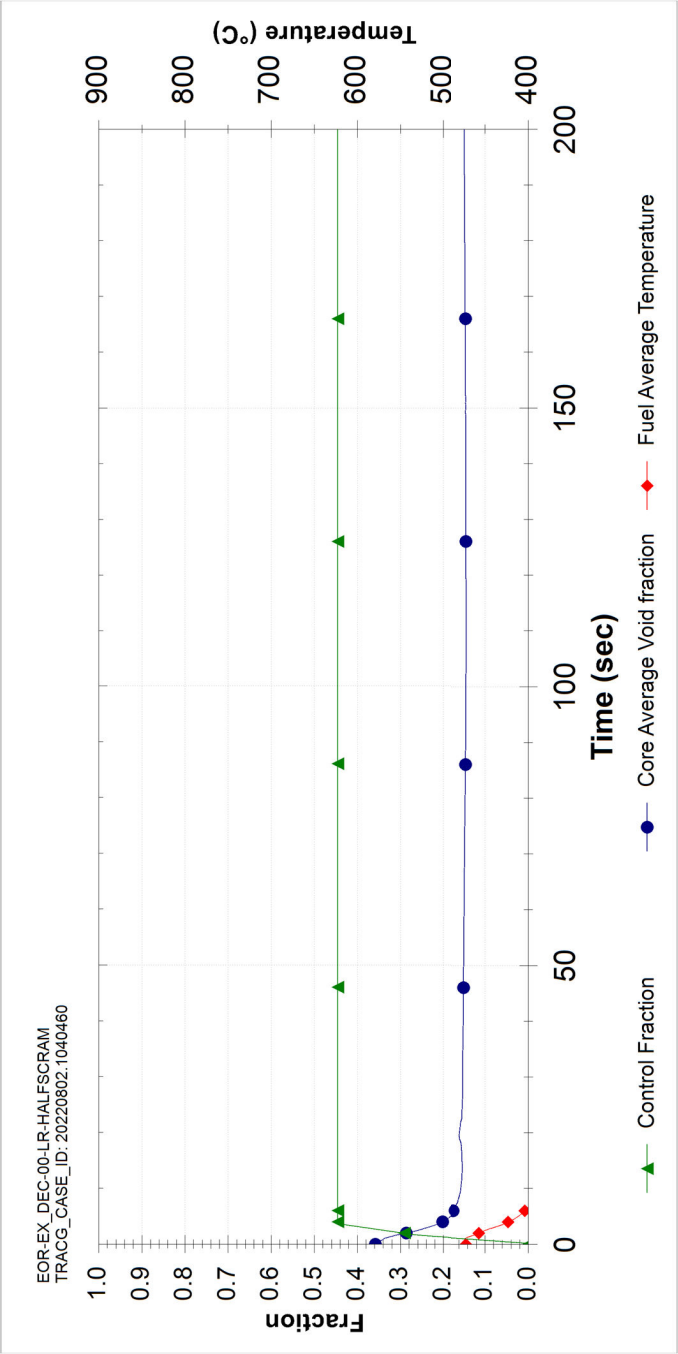


Figure 15.5-134: Complex Sequence Generator Load Rejection (DEC)

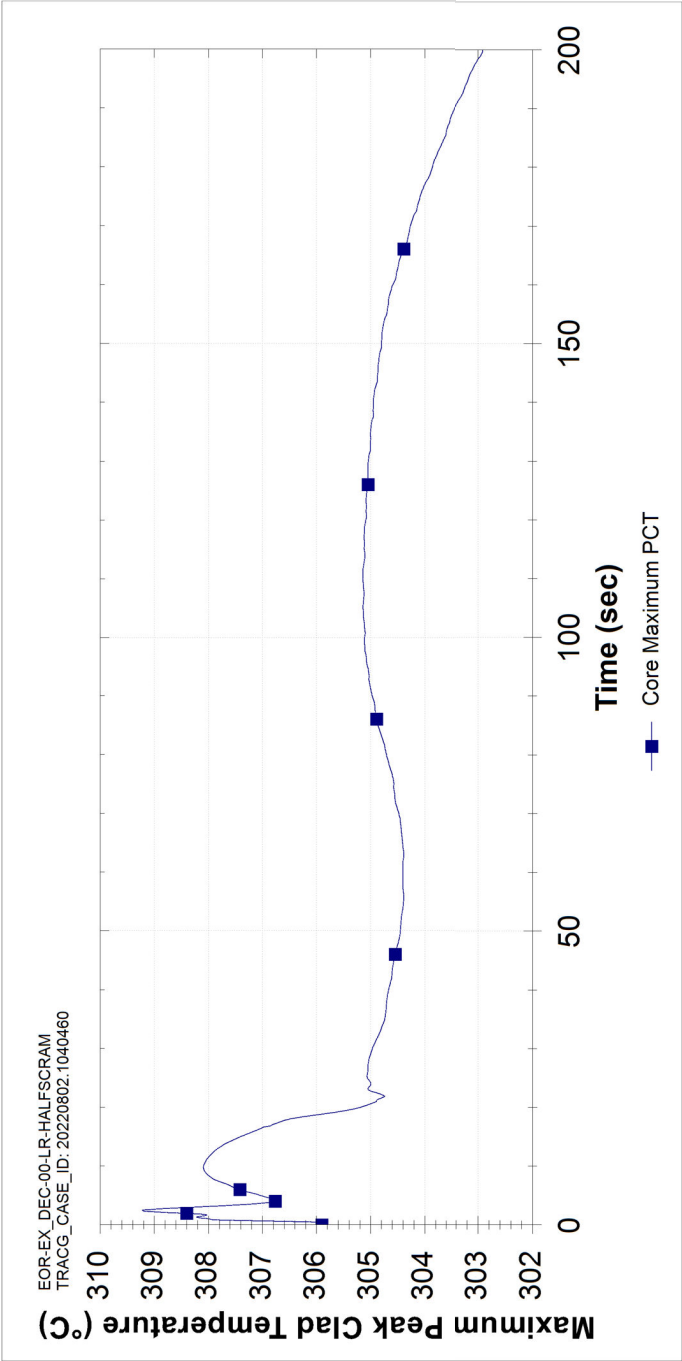


Figure 15.5-135: Complex Sequence Generator Load Rejection (DEC)

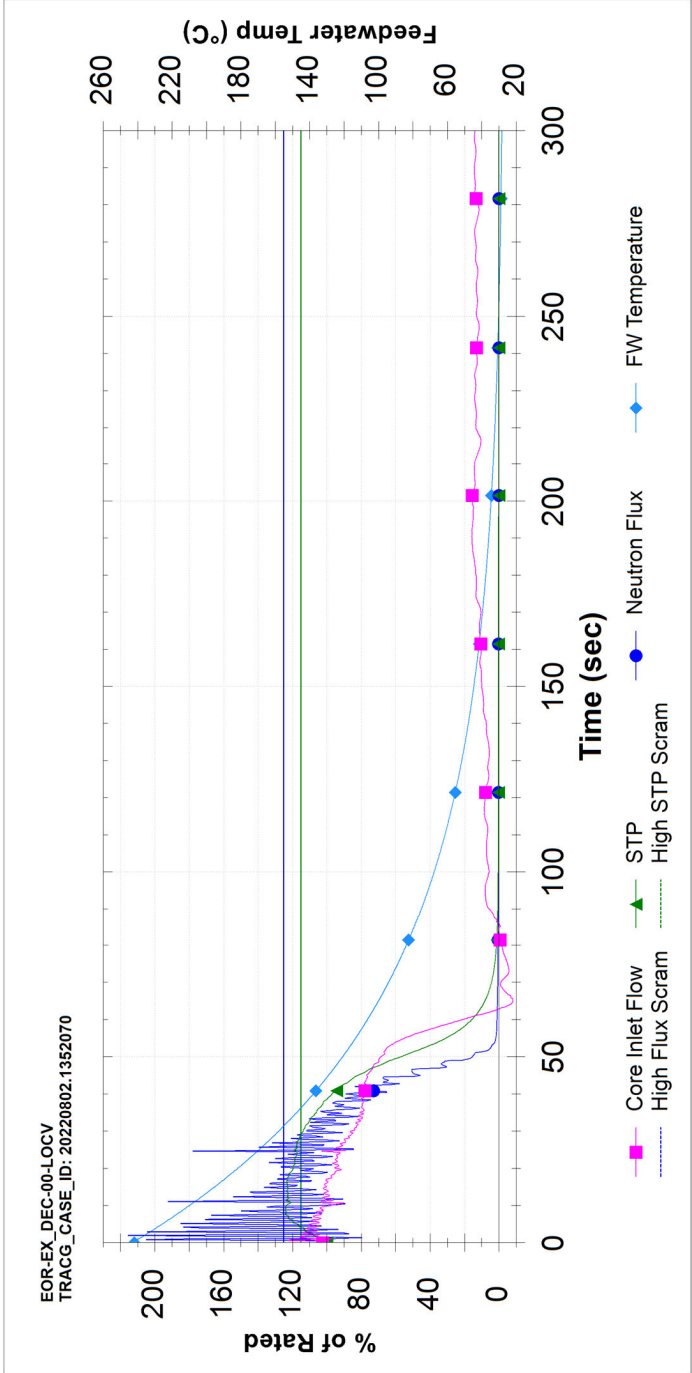


Figure 15.5-136: Loss of Condenser Vacuum (DEC)

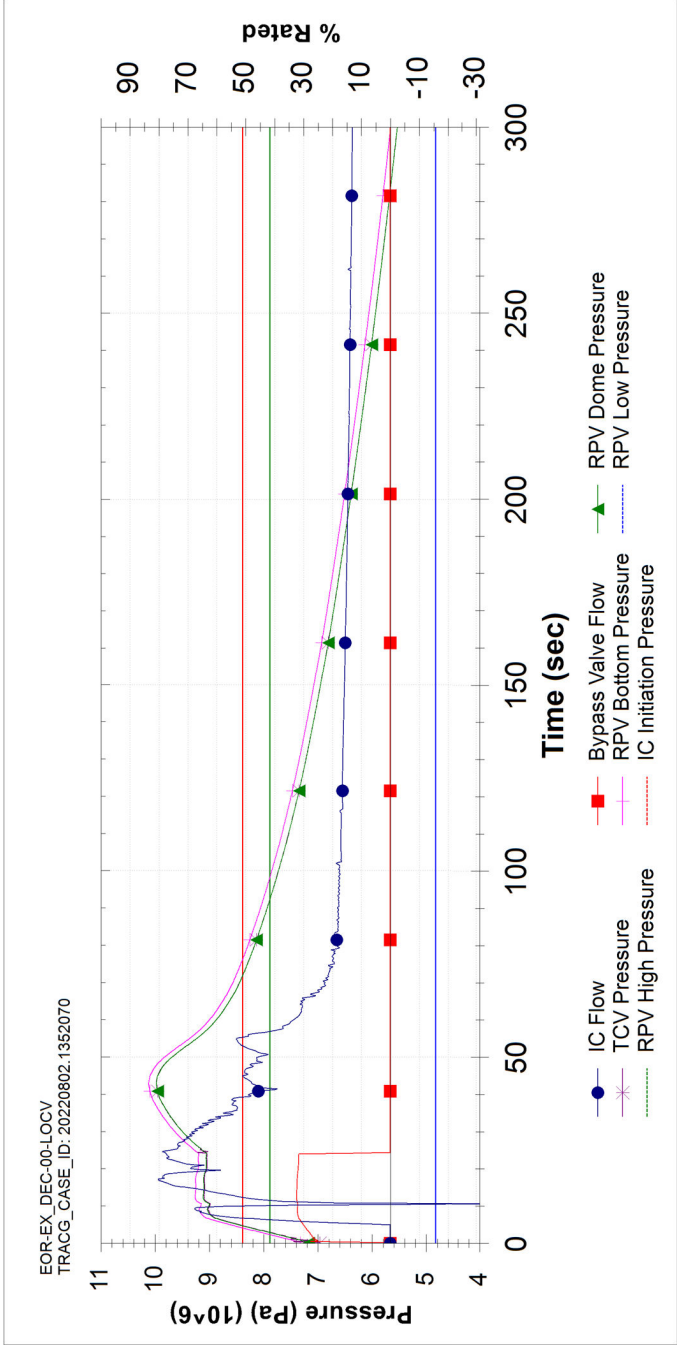


Figure 15.5-137: Loss of Condenser Vacuum (DEC)

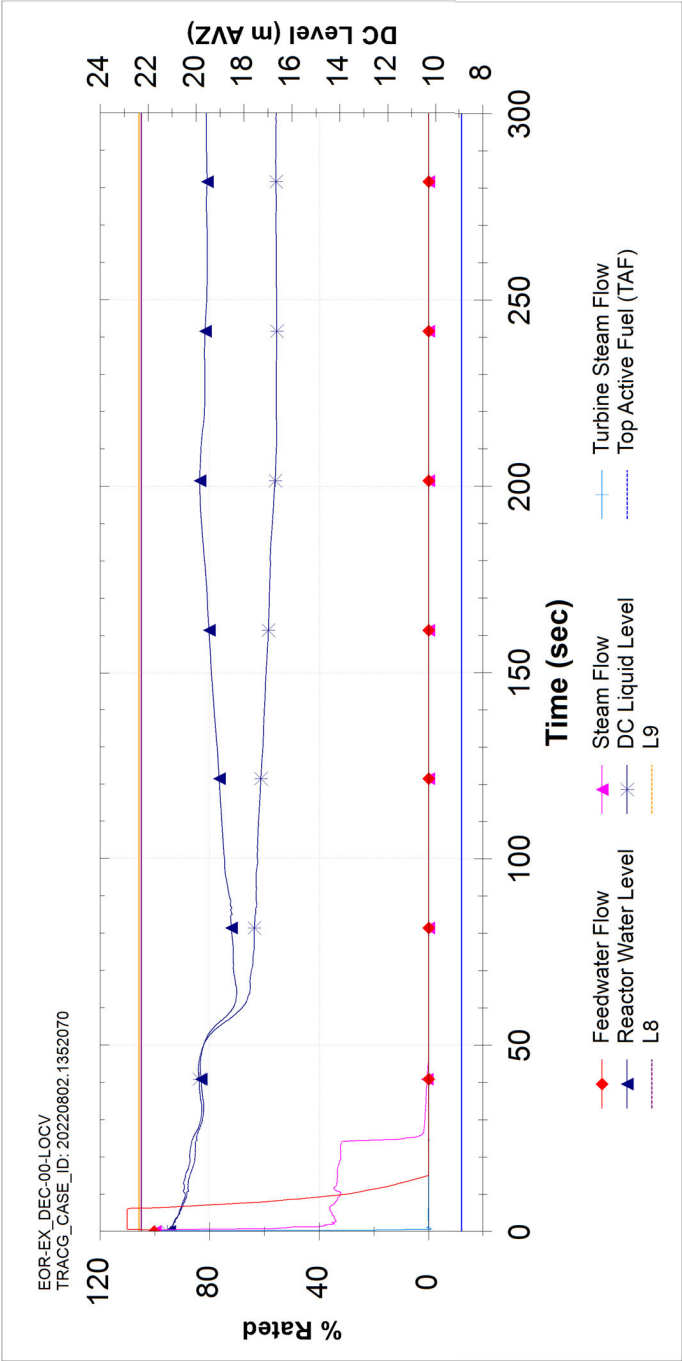


Figure 15.5-138: Loss of Condenser Vacuum (DEC)

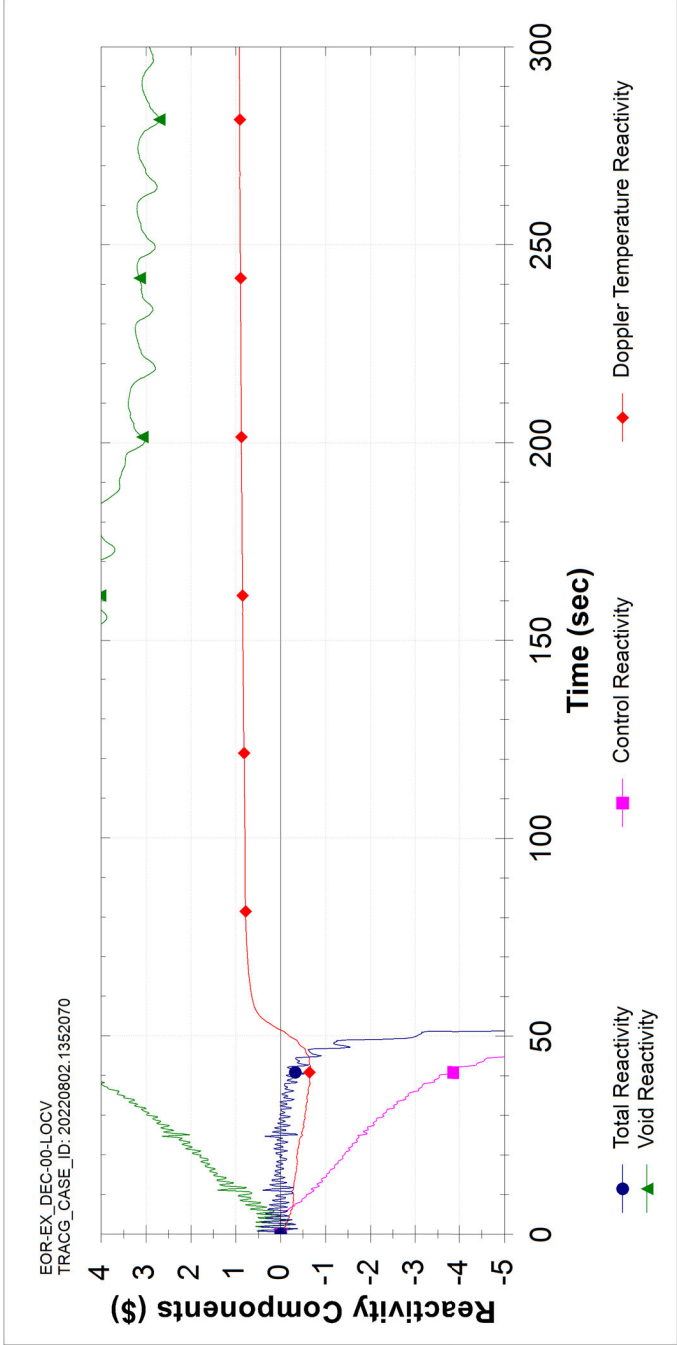


Figure 15.5-139: Loss of Condenser Vacuum (DEC)

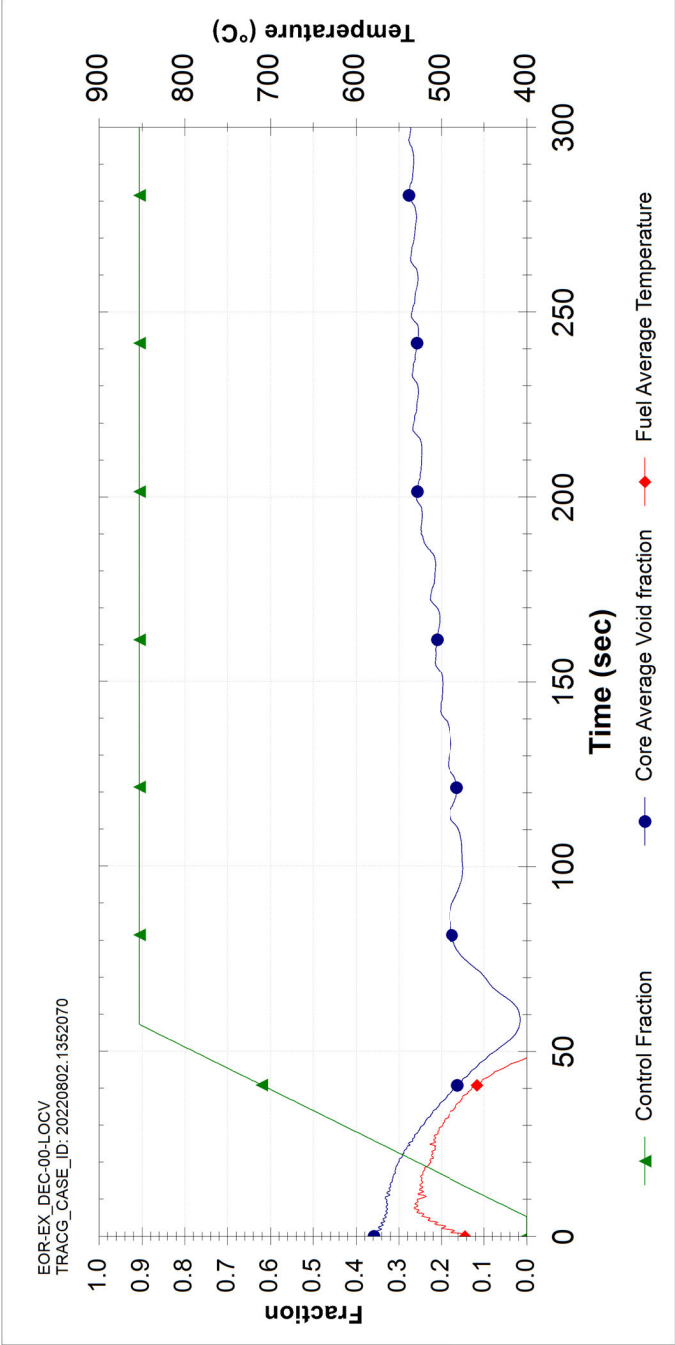


Figure 15.5-140: Loss of Condenser Vacuum (DEC)

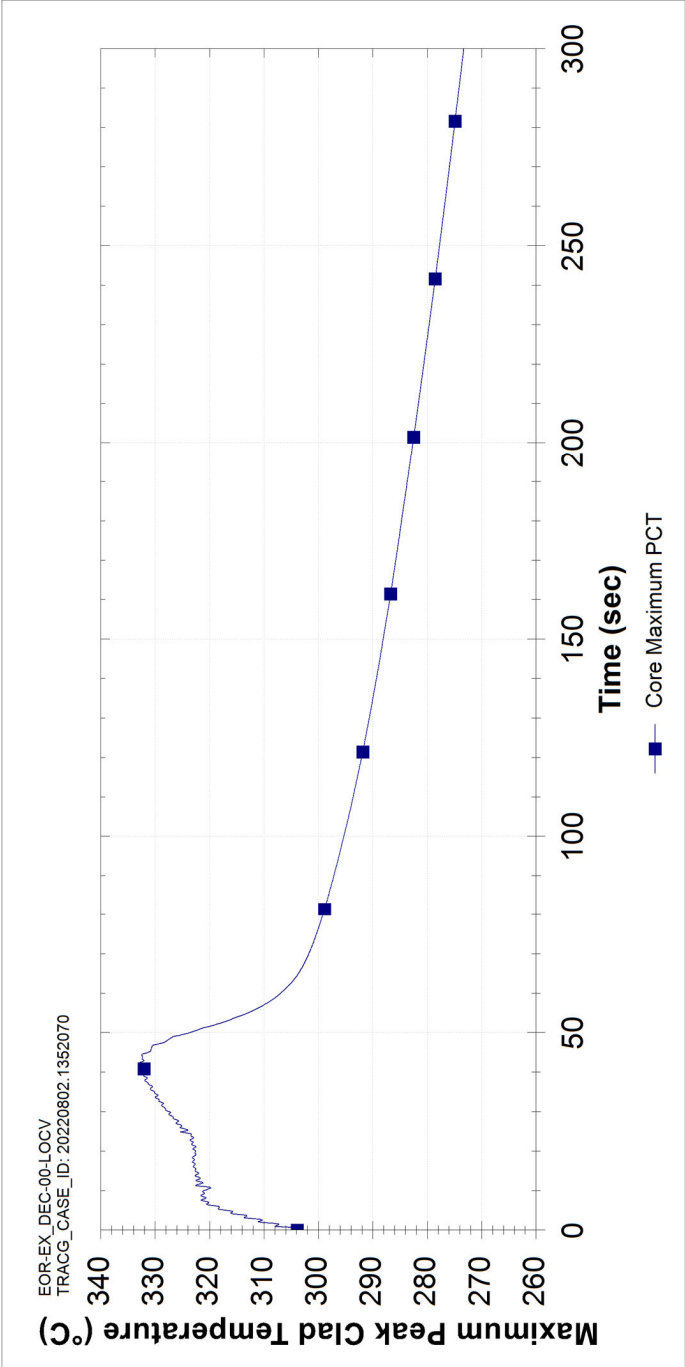


Figure 15.5-141: Loss of Condenser Vacuum (DEC)

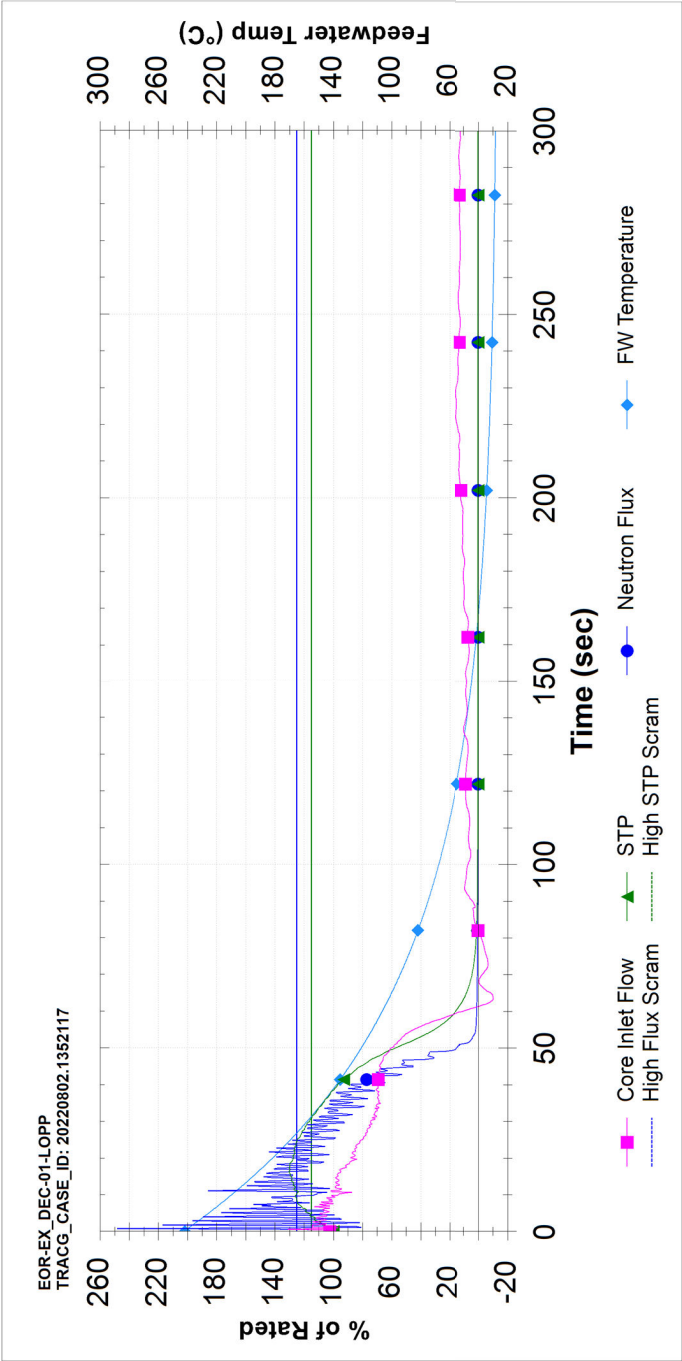


Figure 15.5-142: Loss of Preferred Power (DEC)

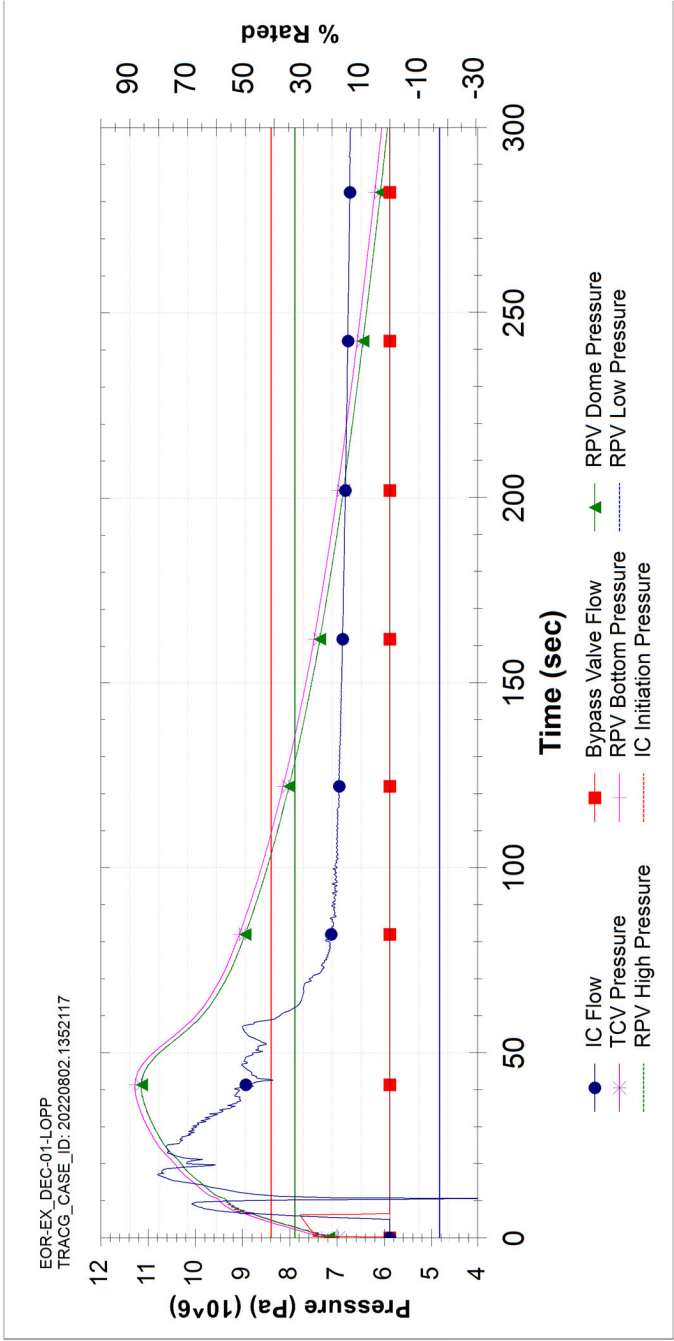


Figure 15.5-143: Loss of Preferred Power (DEC)

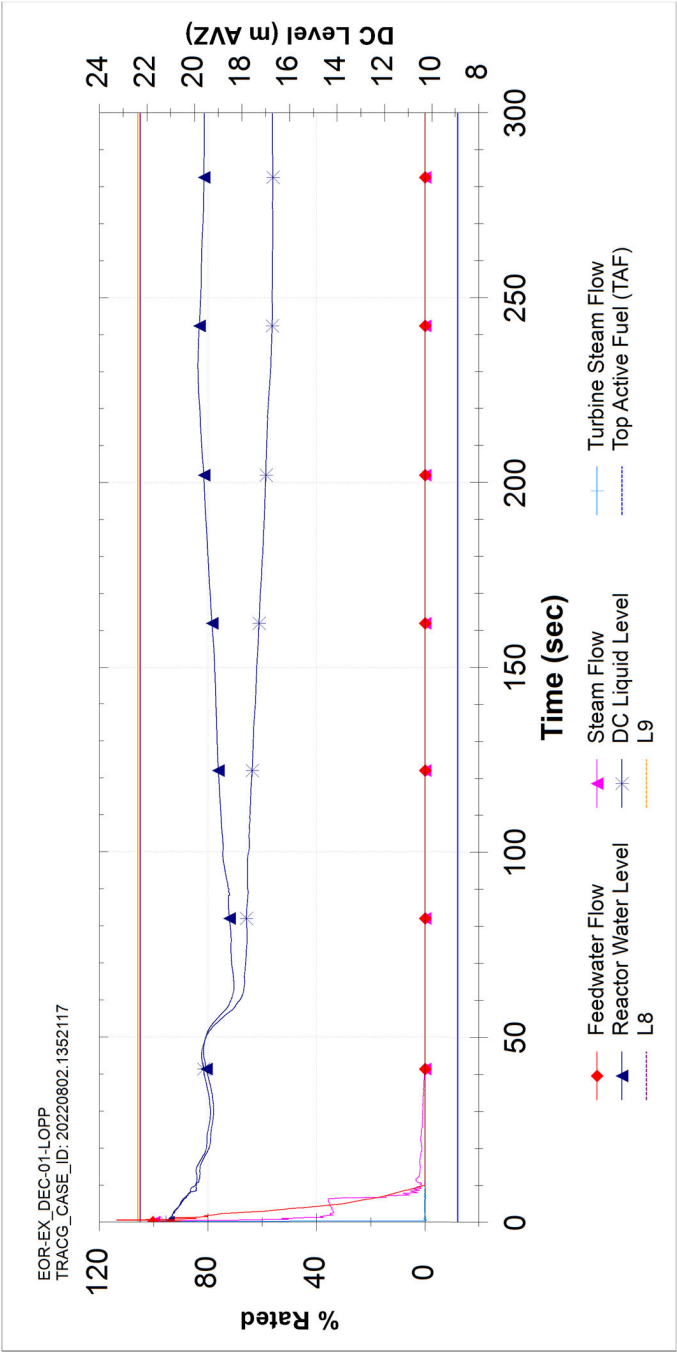


Figure 15.5-144: Loss of Preferred Power (DEC)

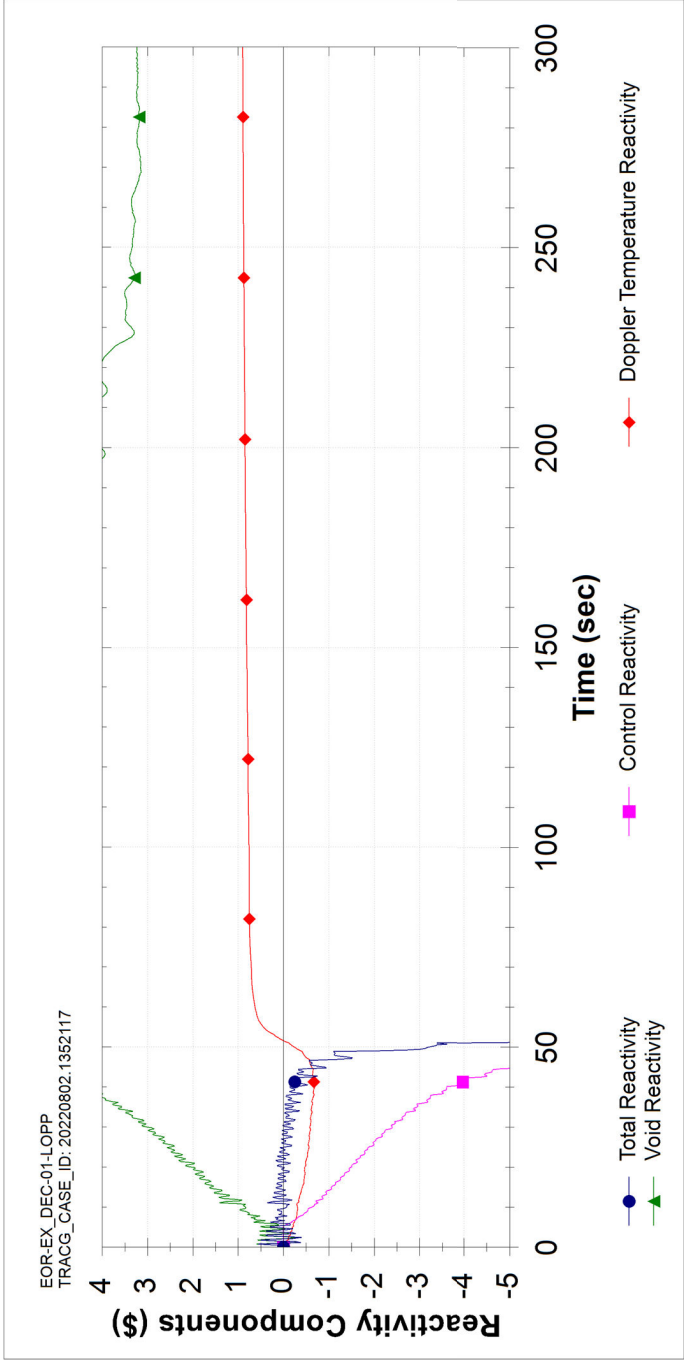


Figure 15.5-145: Loss of Preferred Power (DEC)

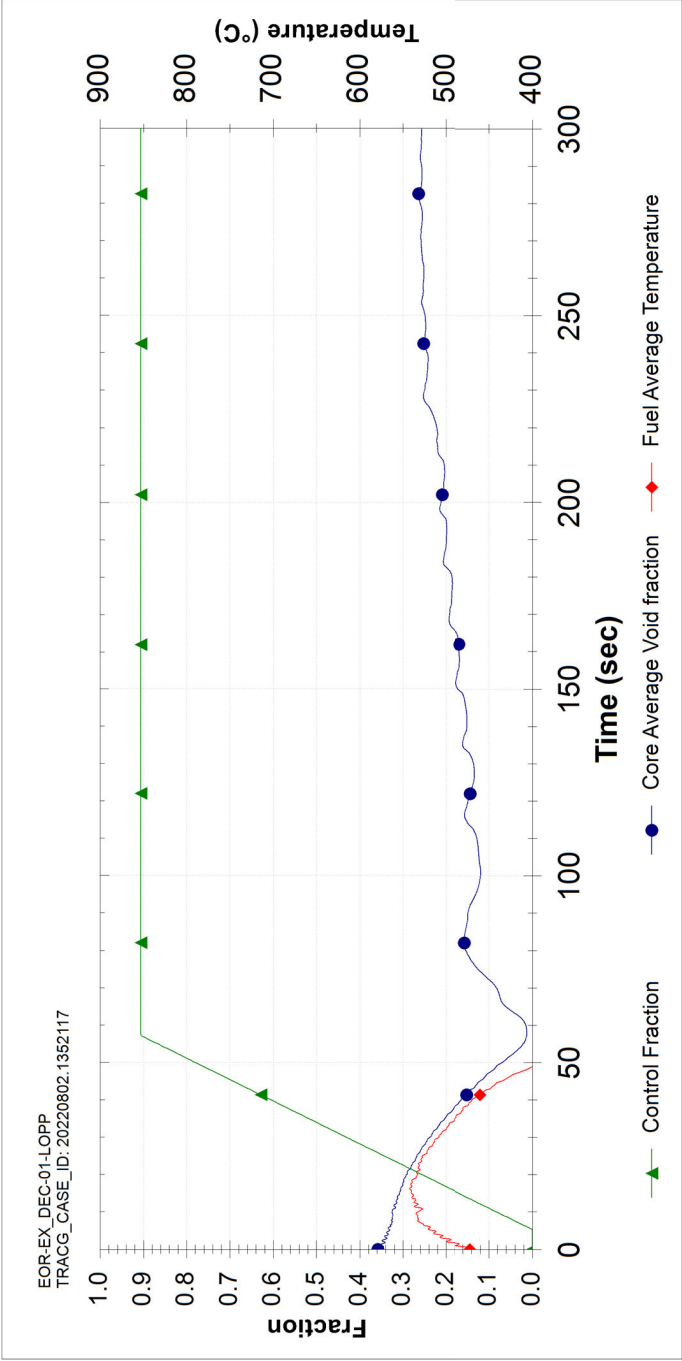


Figure 15.5-146: Loss of Preferred Power (DEC)

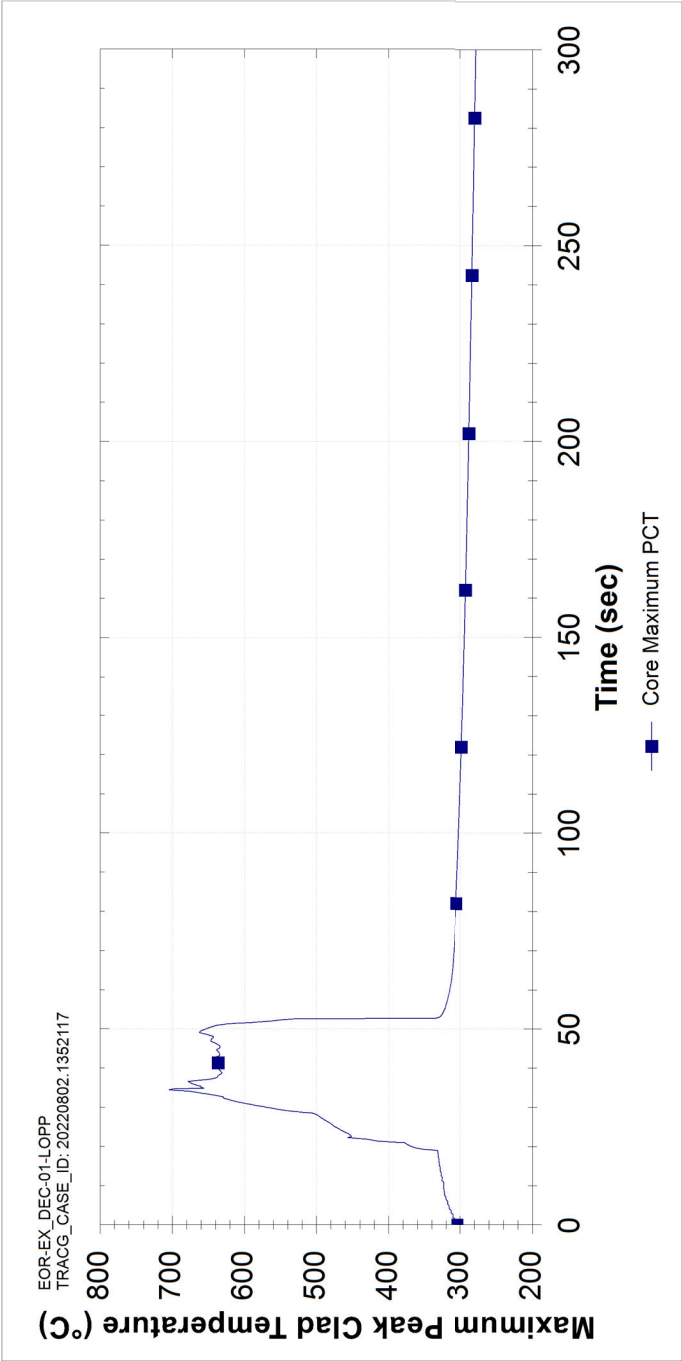


Figure 15.5-147: Loss of Preferred Power (DEC)

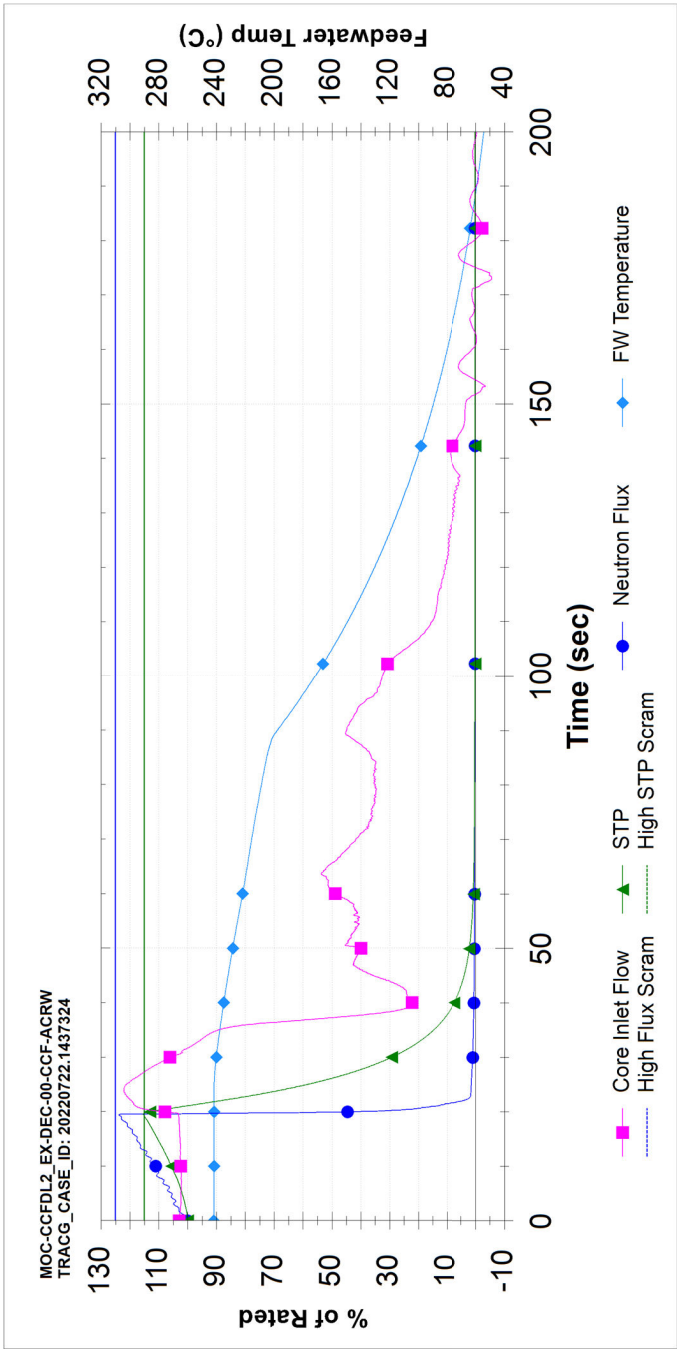


Figure 15.5-148: All Control Rod Withdrawal at Power (ACRW)

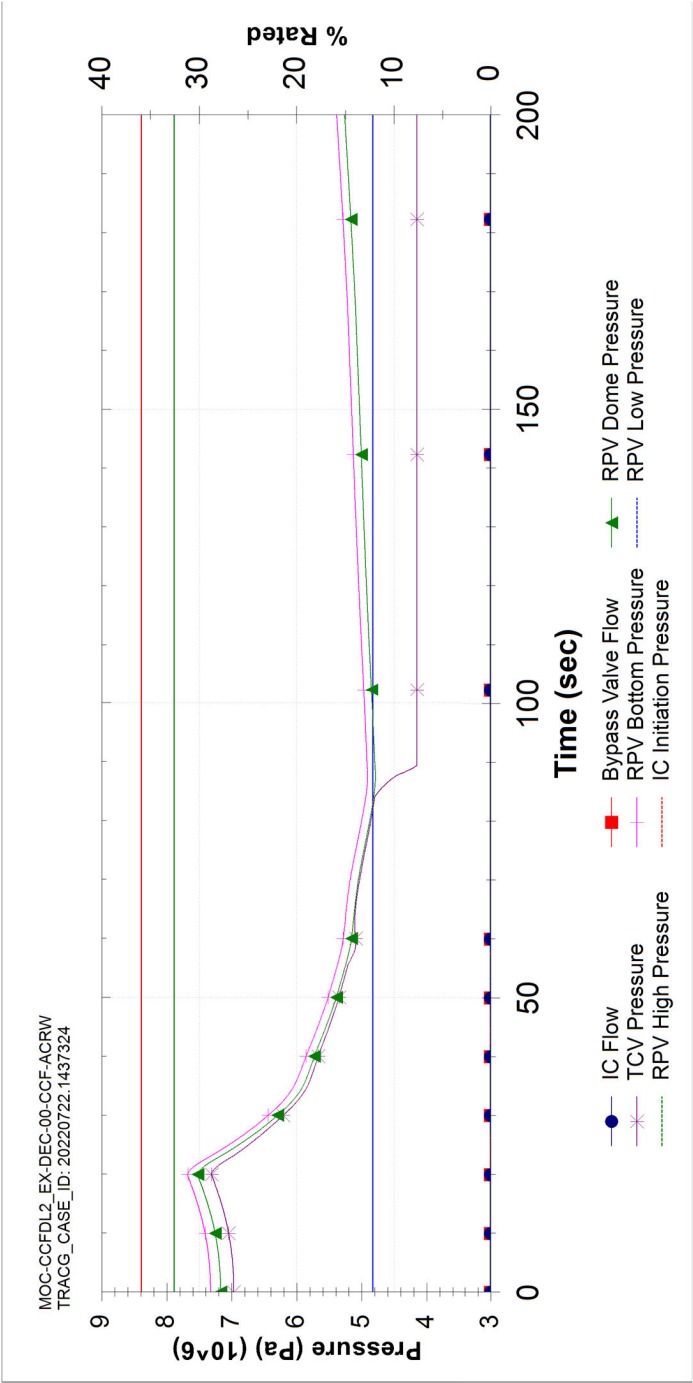


Figure 15.5-149: All Control Rod Withdrawal at Power (ACRW)

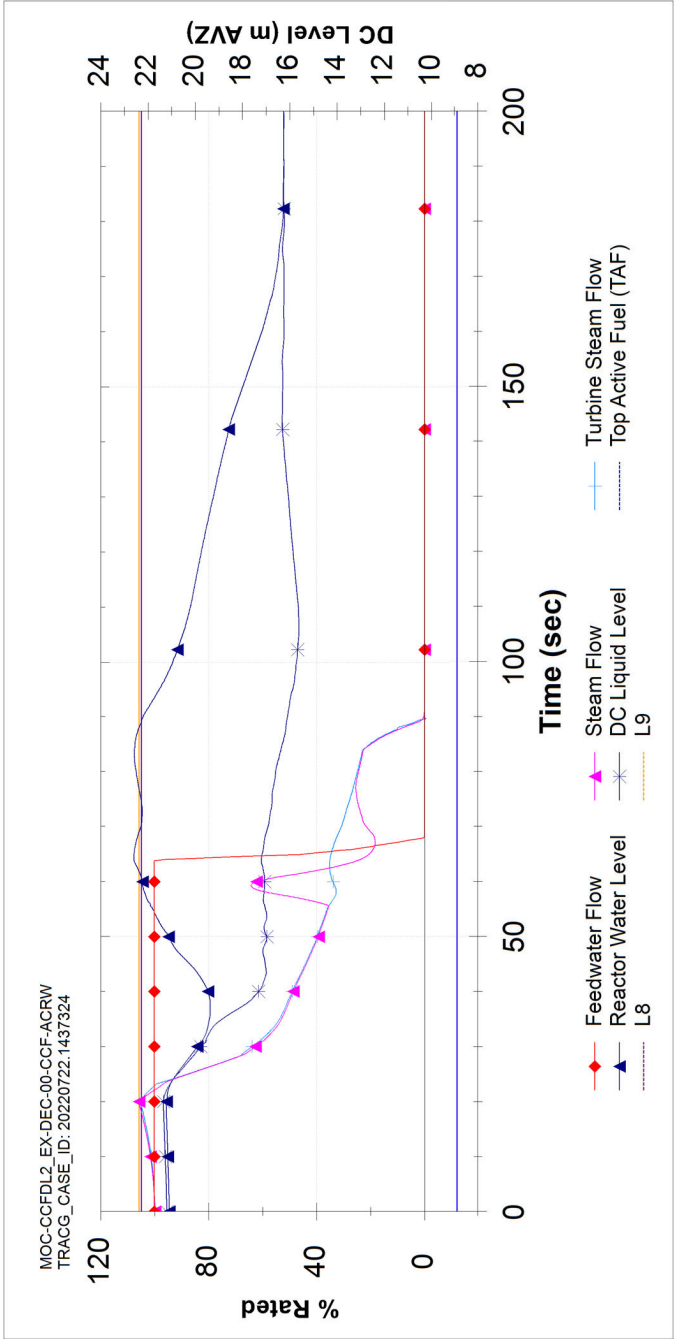


Figure 15.5-150: All Control Rod Withdrawal at Power (ACRW)

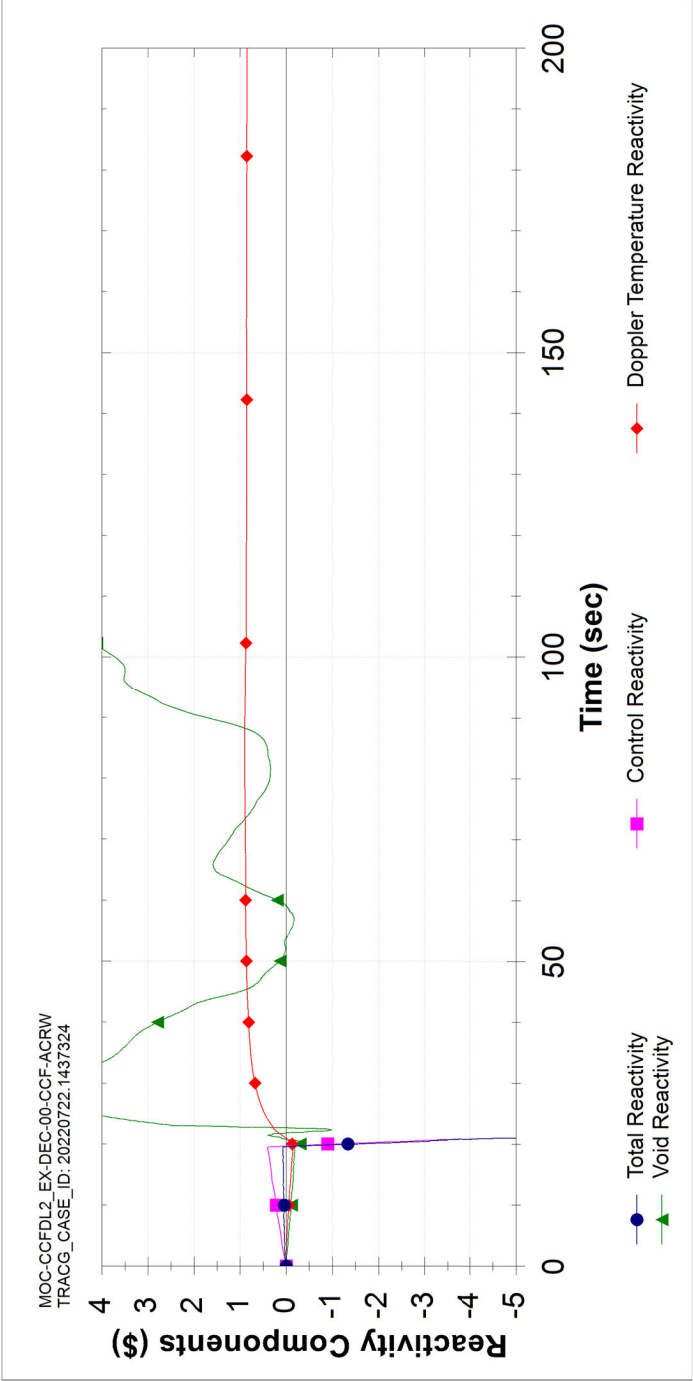


Figure 15.5-151: All Control Rod Withdrawal at Power (ACRW)

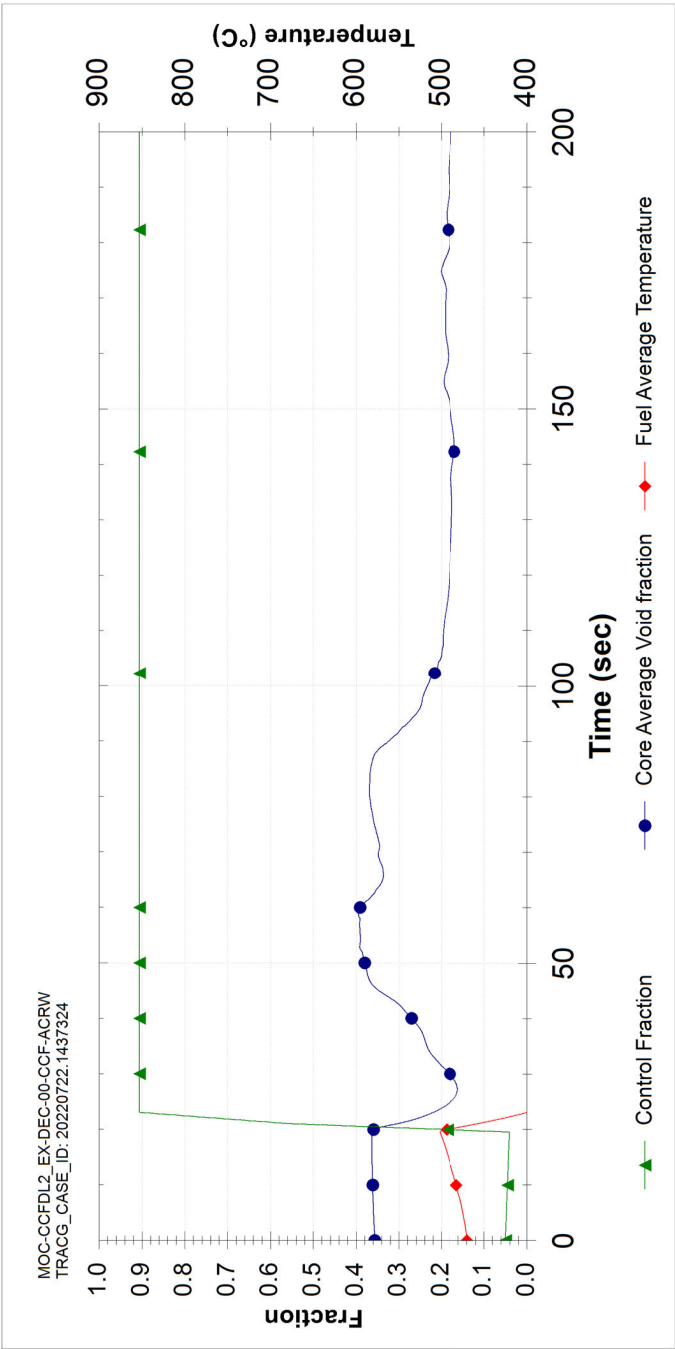


Figure 15.5-152: All Control Rod Withdrawal at Power (ACRW)

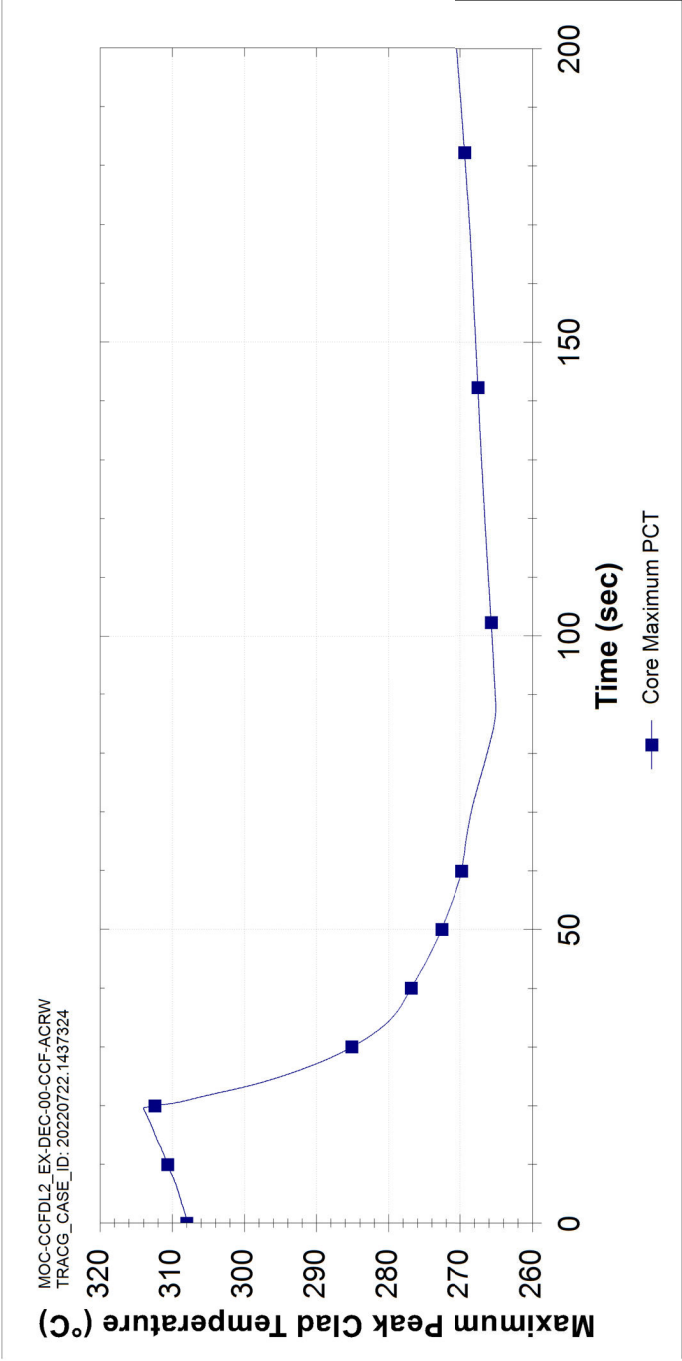


Figure 15.5-153: All Control Rod Withdrawal at Power (ACRW)

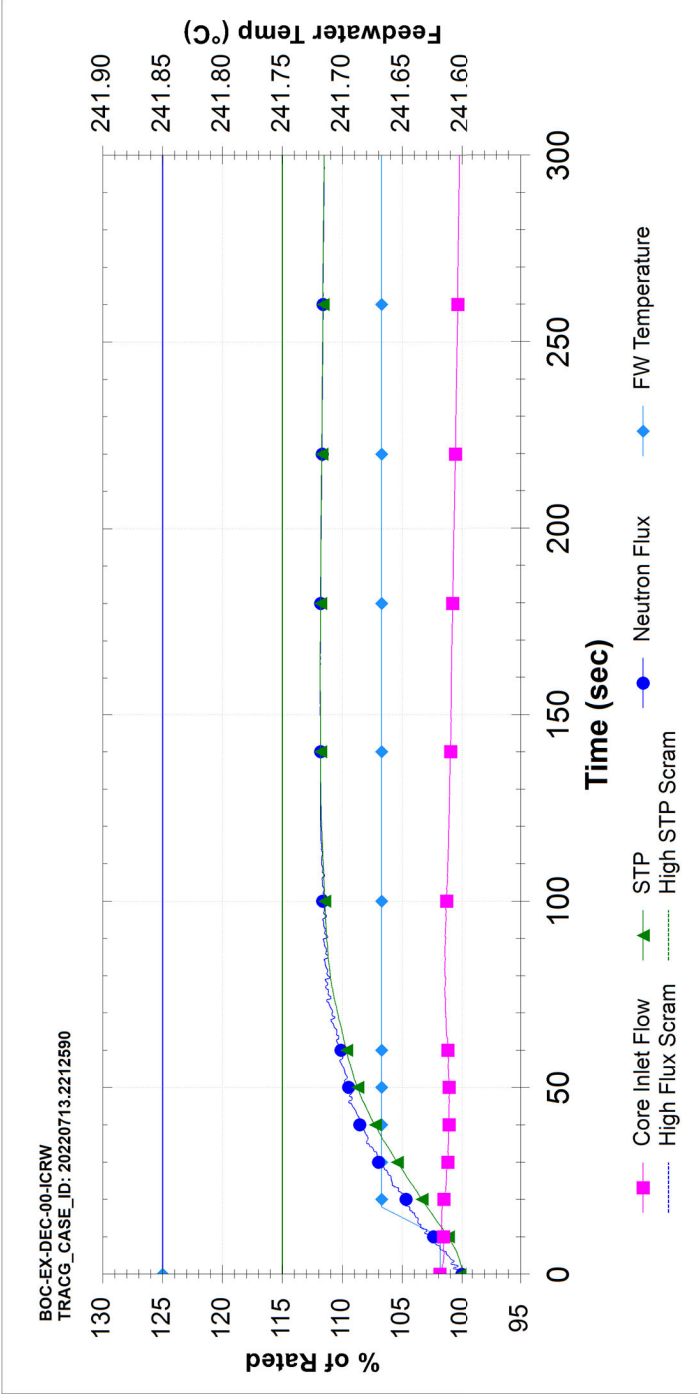


Figure 15.5-154: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

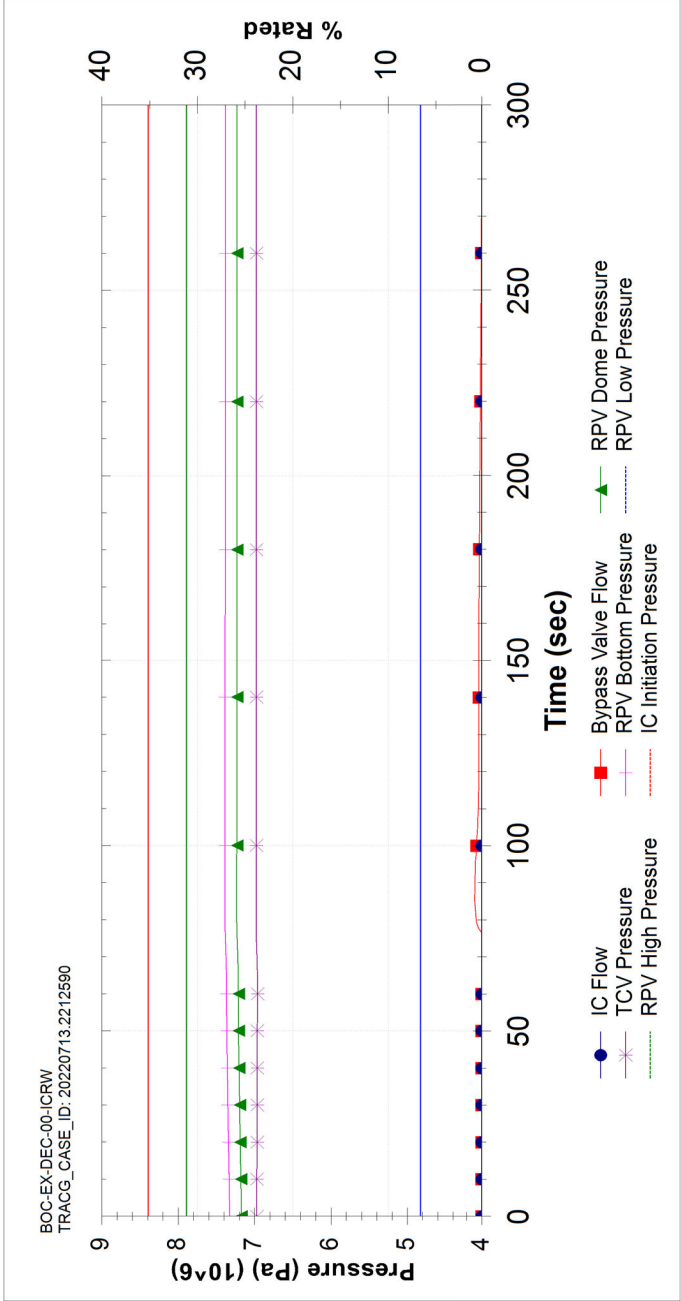


Figure 15.5-155: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

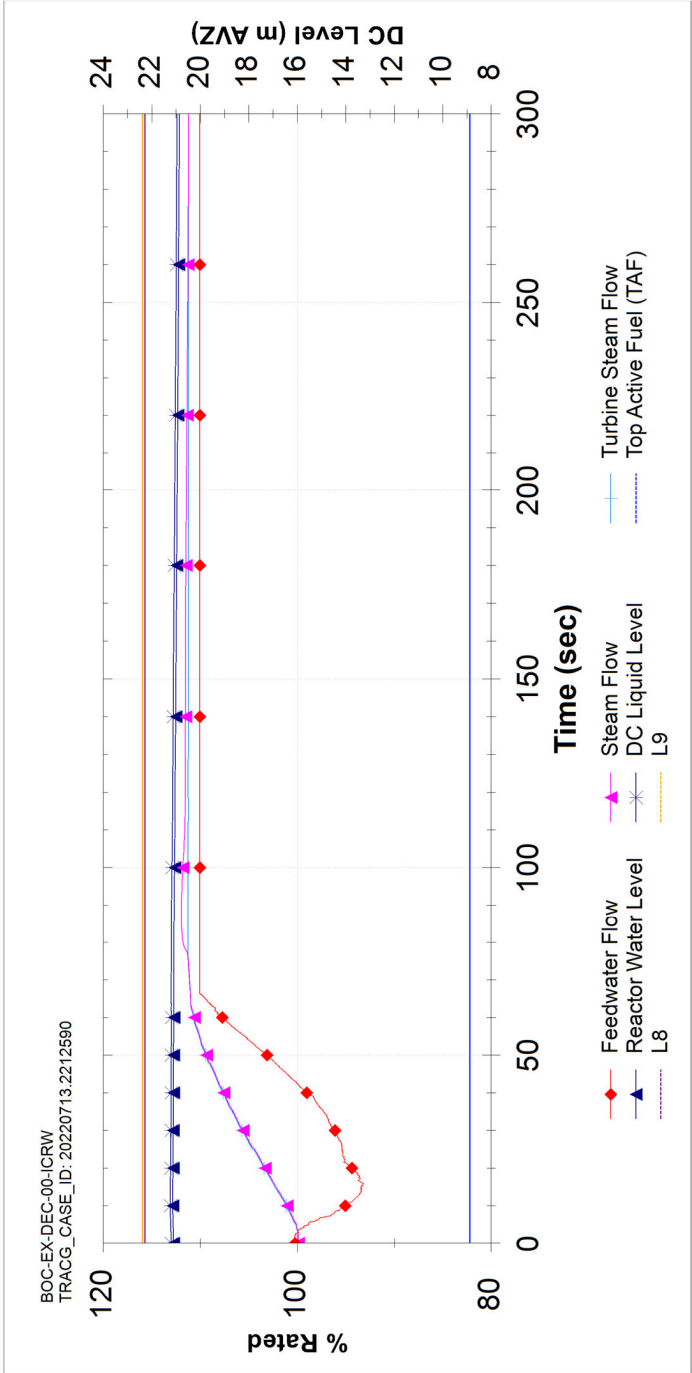


Figure 15.5-156: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

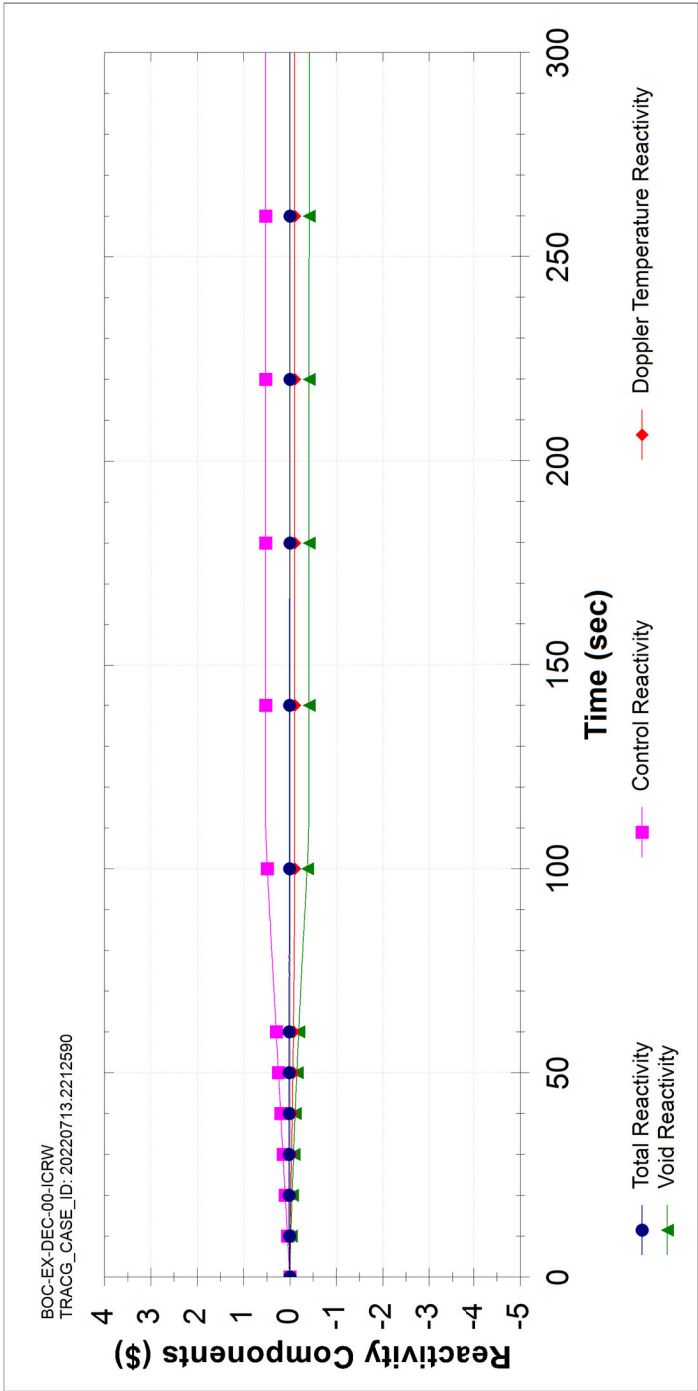


Figure 15.5-157: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

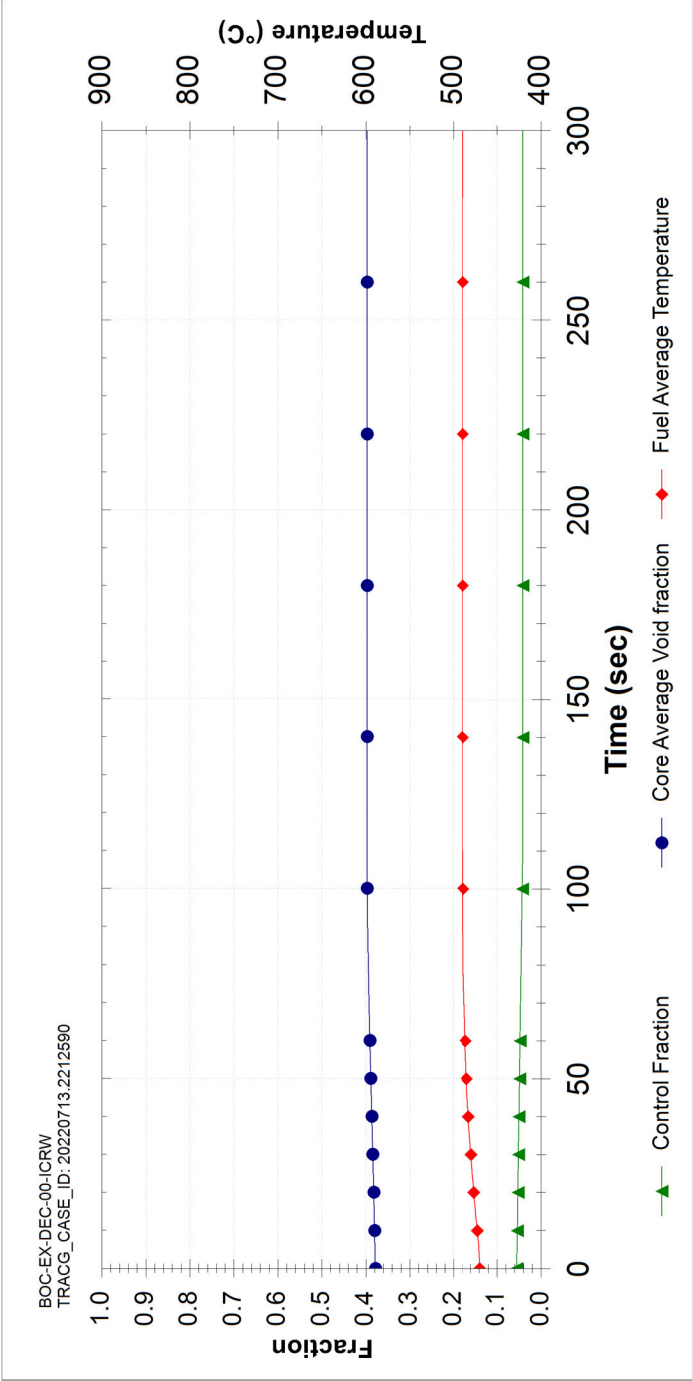


Figure 15.5-158: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

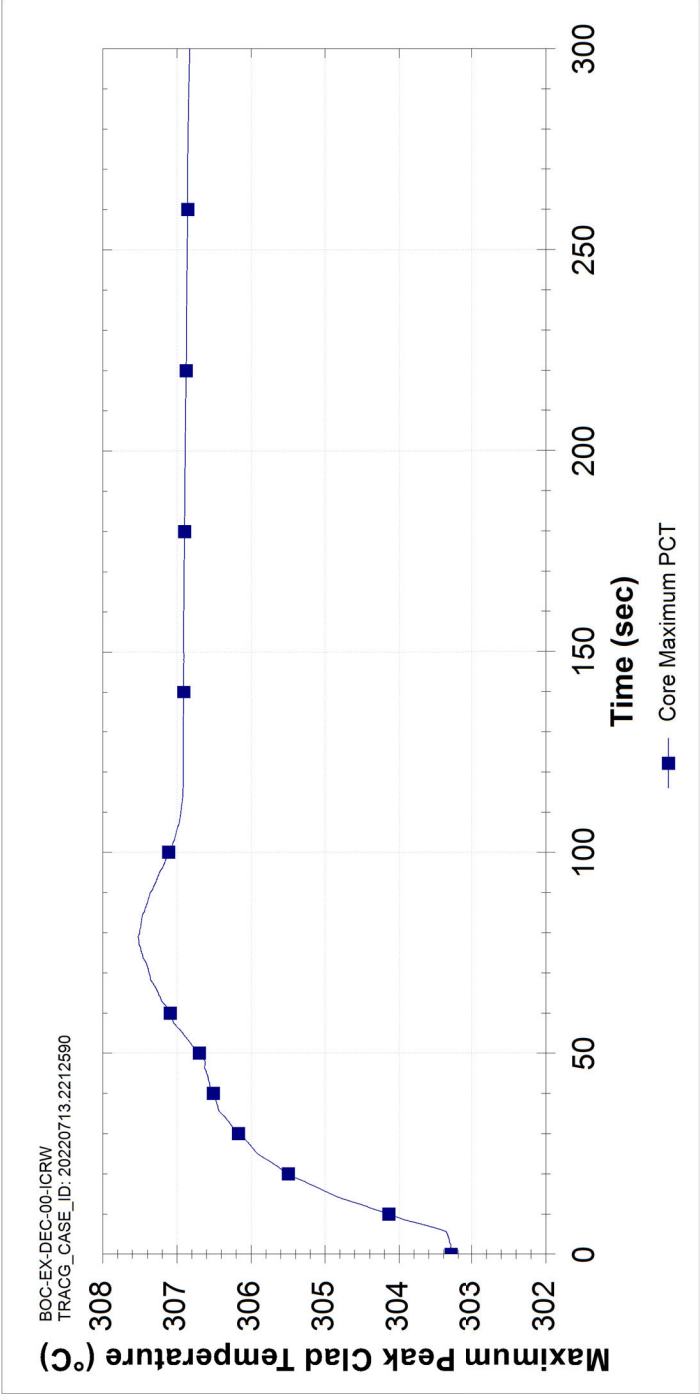


Figure 15.5-159: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

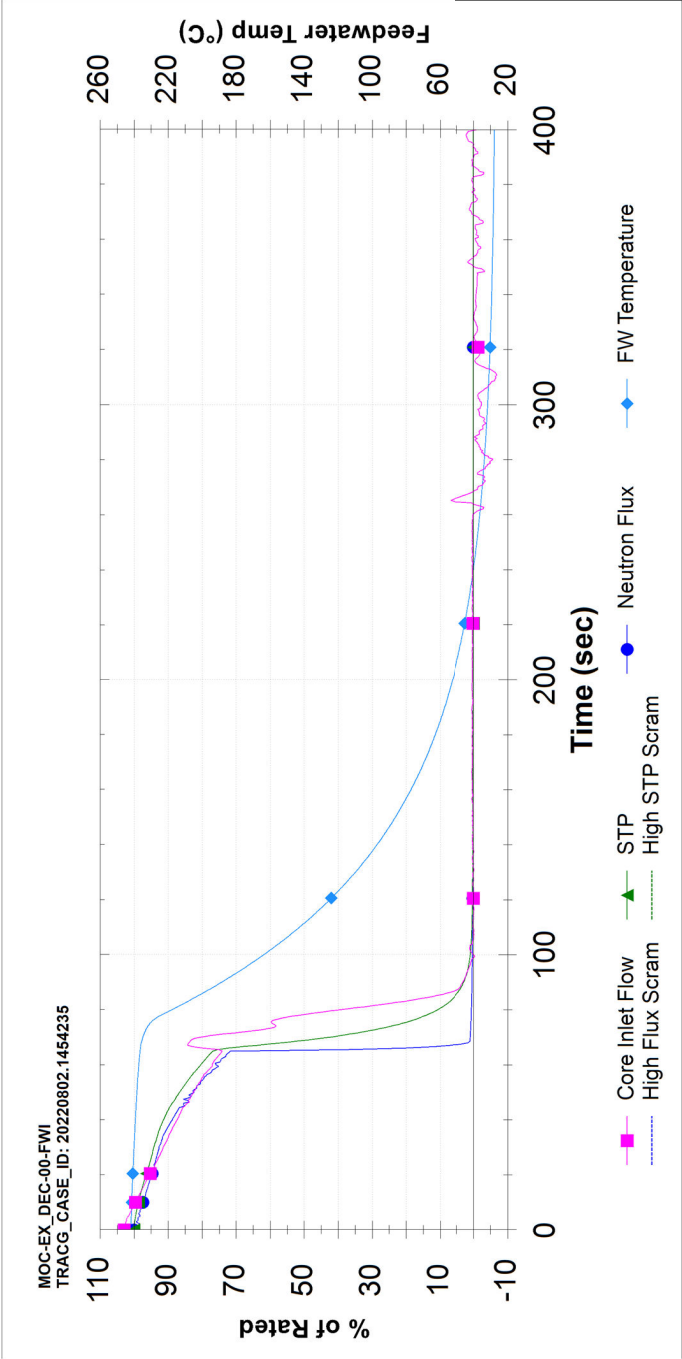


Figure 15.5-160: Feedwater Isolation (DEC)

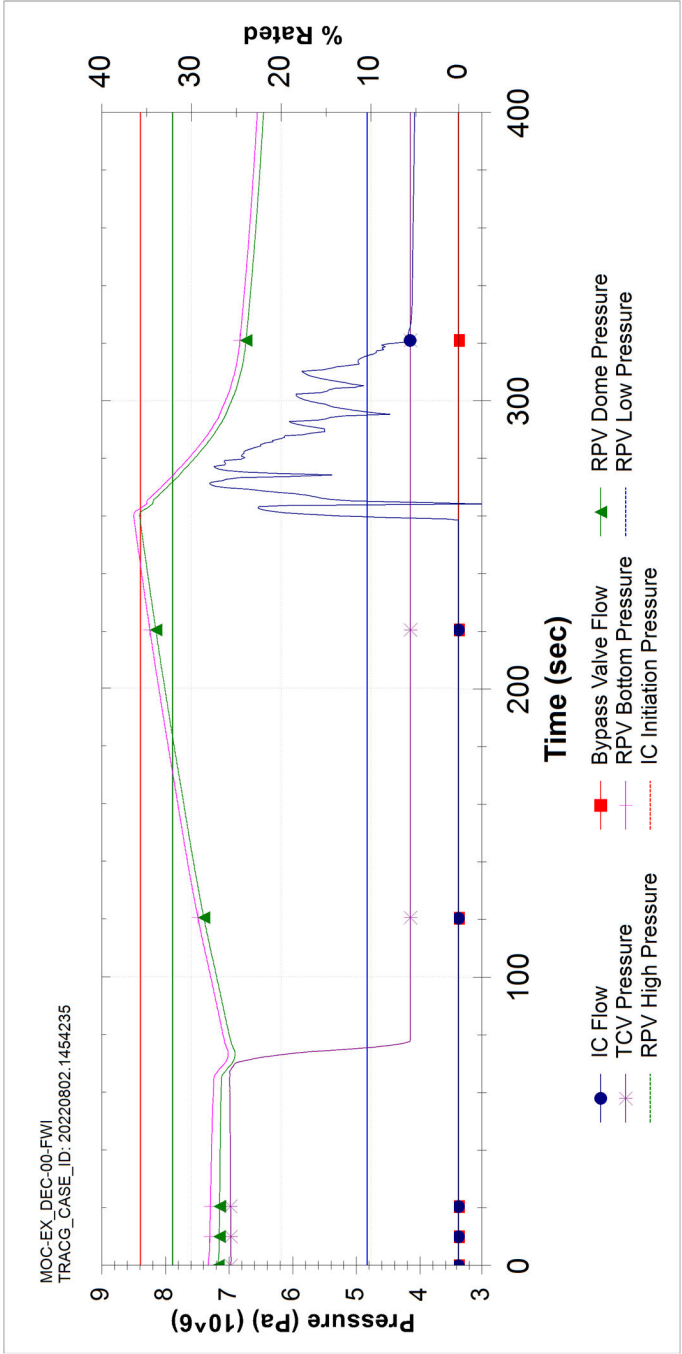


Figure 15.5-161: Feedwater Isolation (DEC)

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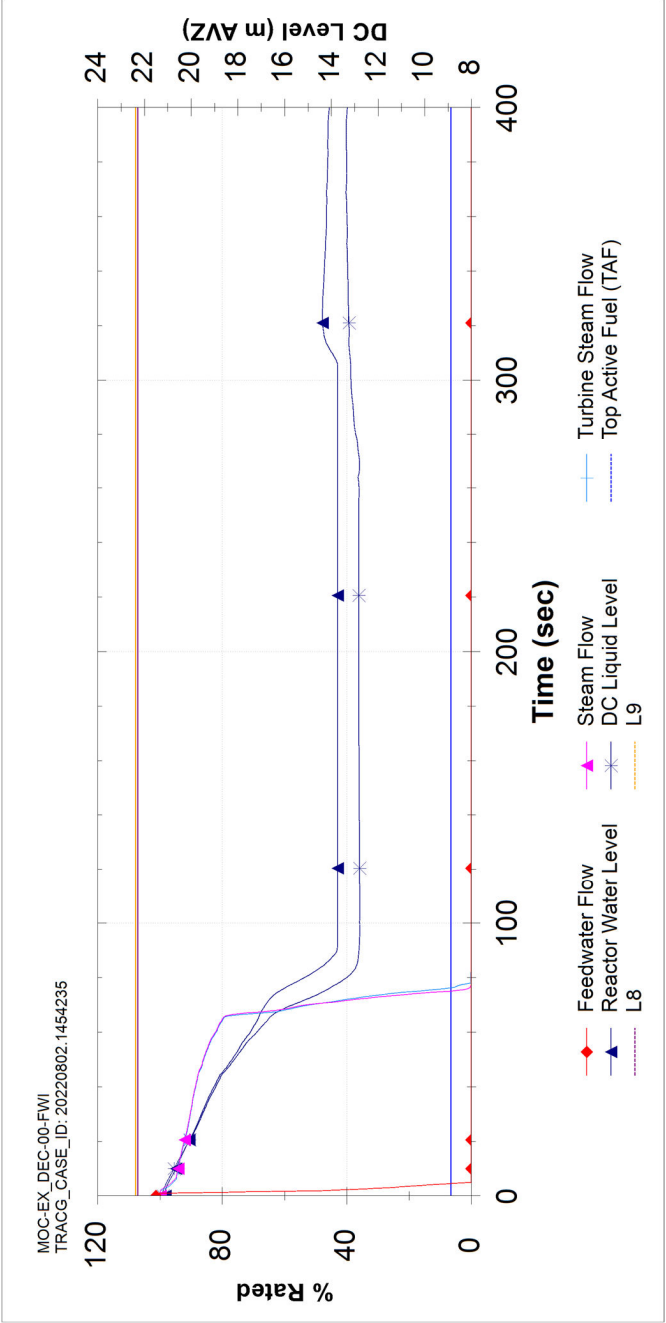


Figure 15.5-162: Feedwater Isolation (DEC)

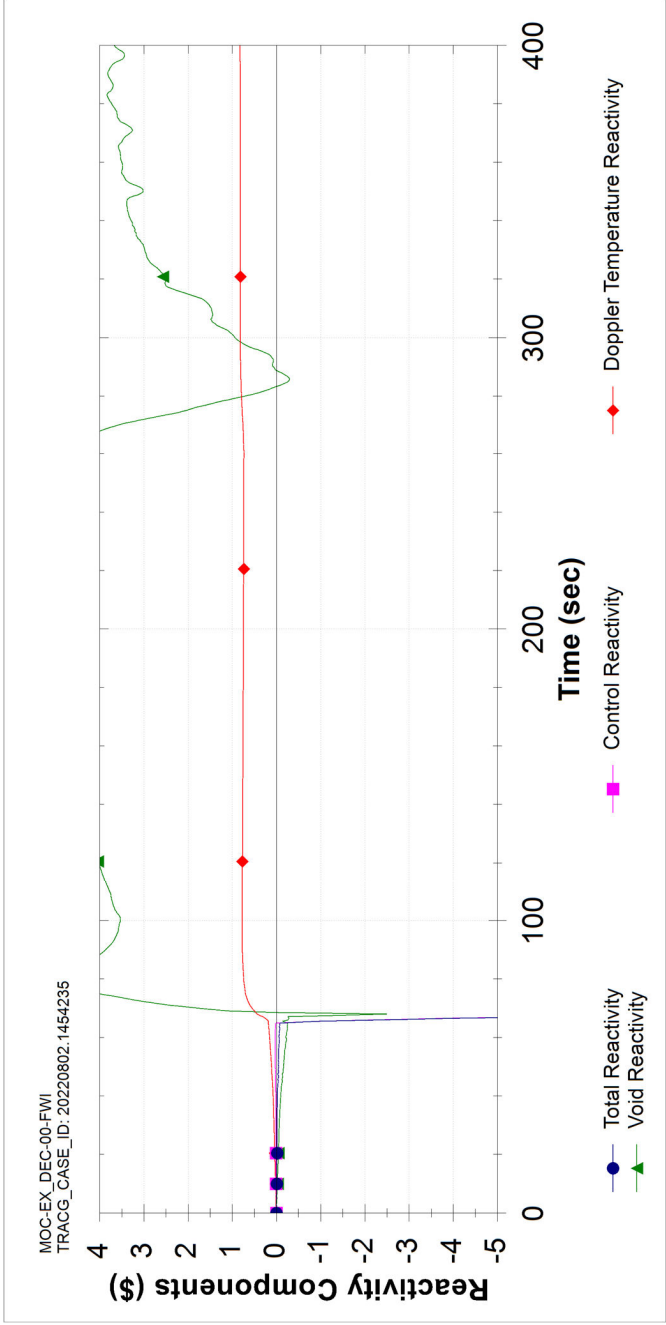


Figure 15.5-163: Feedwater Isolation (DEC)

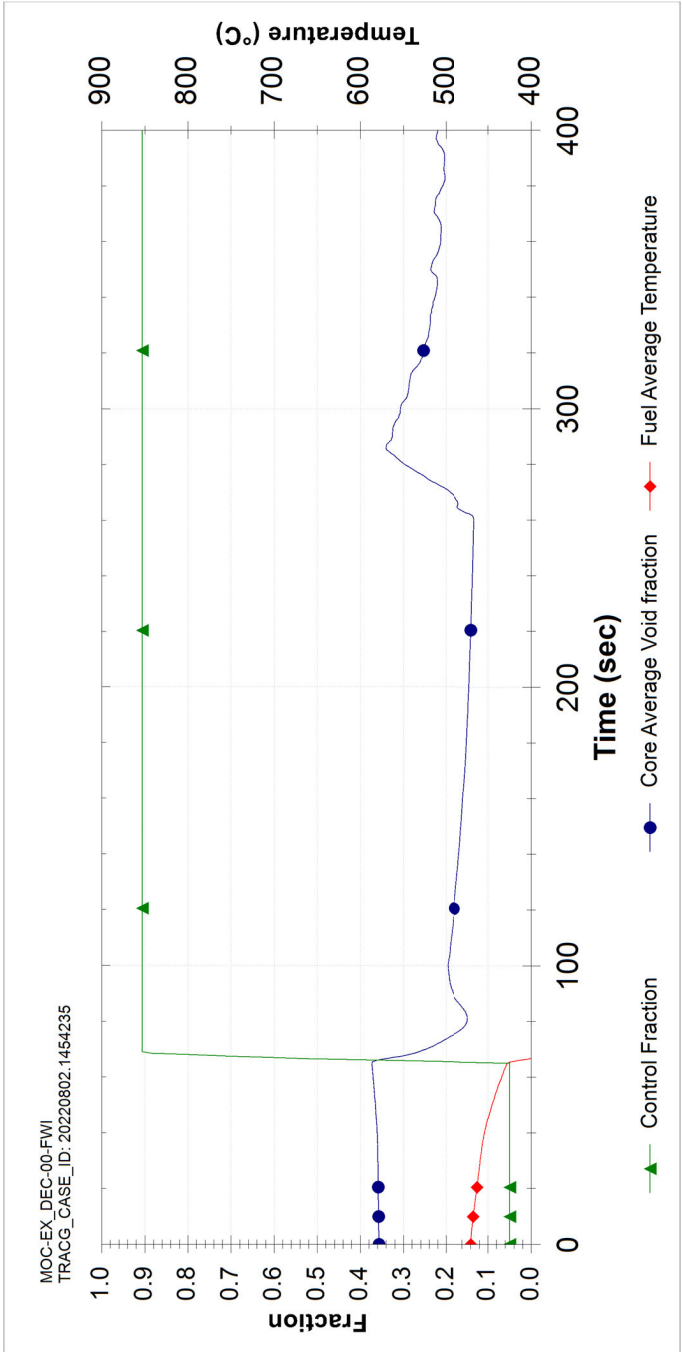


Figure 15.5-164: Feedwater Isolation (DEC)

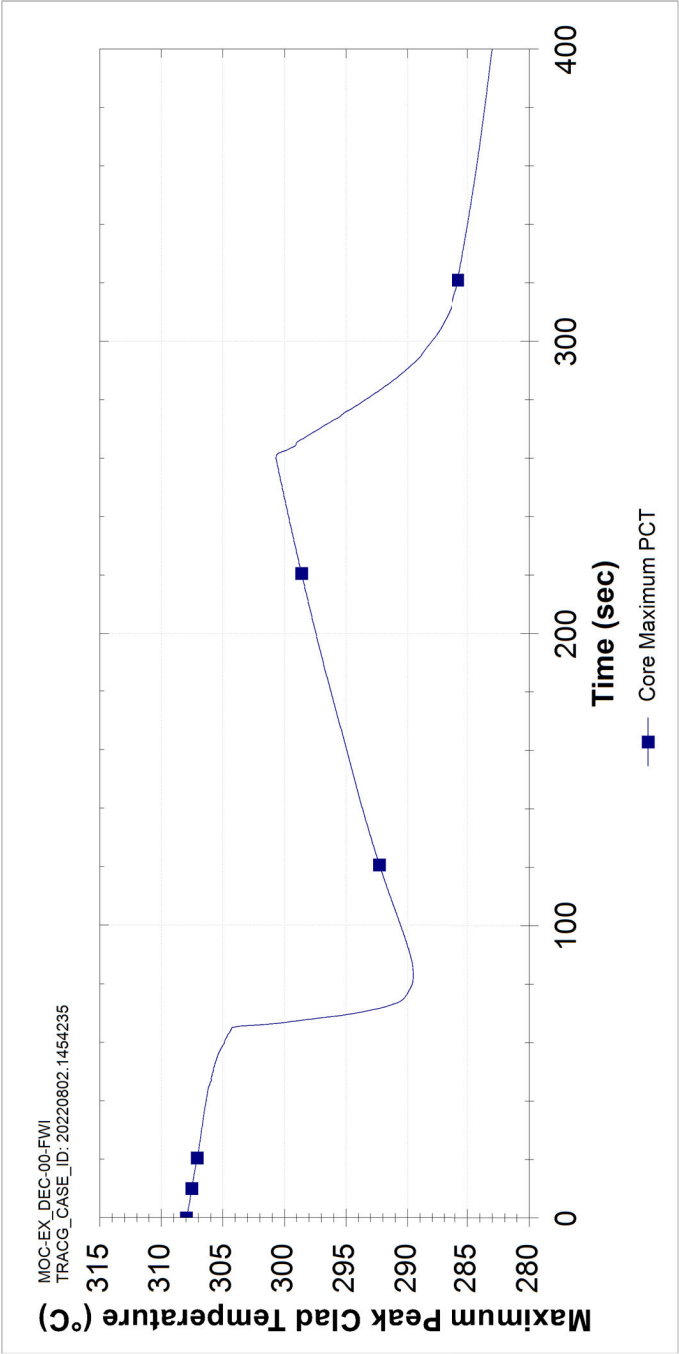


Figure 15.5-165: Feedwater Isolation (DEC)

Figure 15.5-166: Not Used

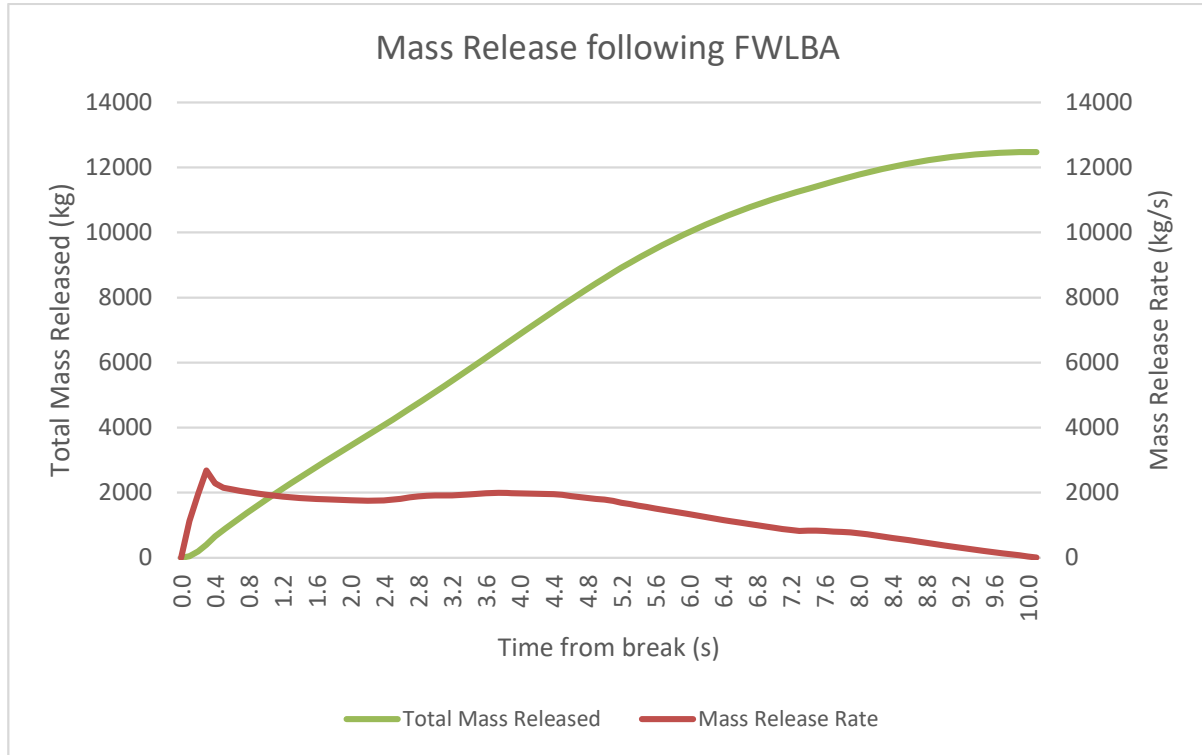


Figure 15.5-167: FWLB Dose Analysis Mass Release vs Time

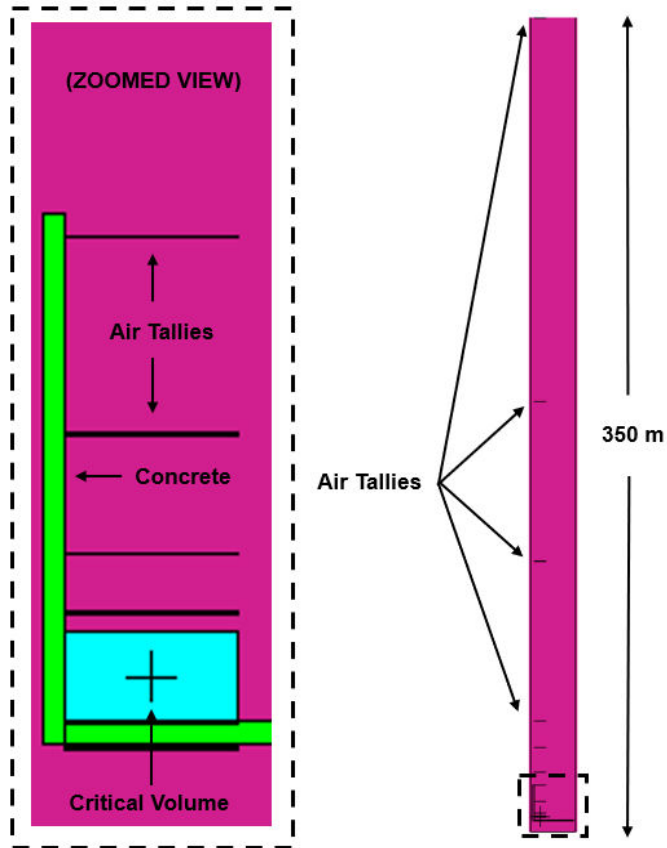


Figure 15.5-168: MCNP-06P Problem Geometry for Out of Core Criticality Model

Note: Dashed lines used to indicate left diagram is zoomed-in version of right diagram.

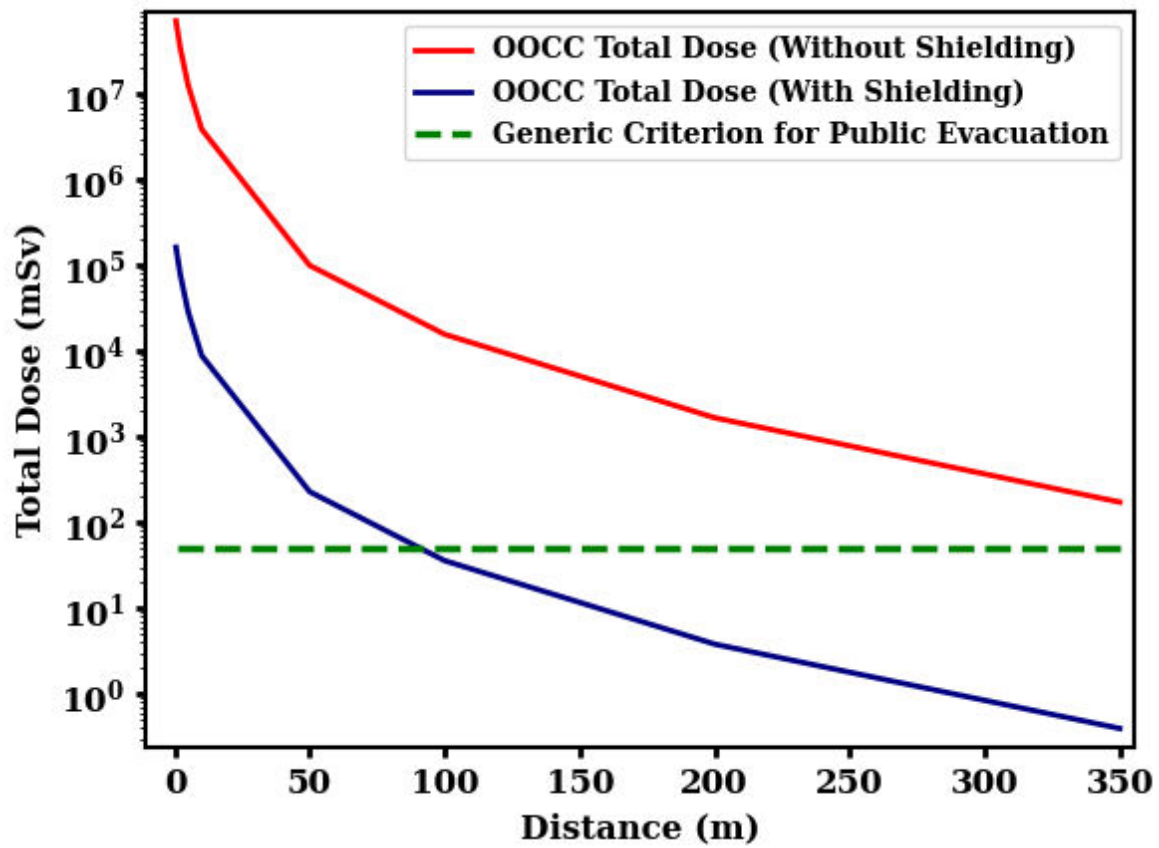


Figure 15.5-169: Plot of Out of Core Criticality Dose Consequence Versus Distance

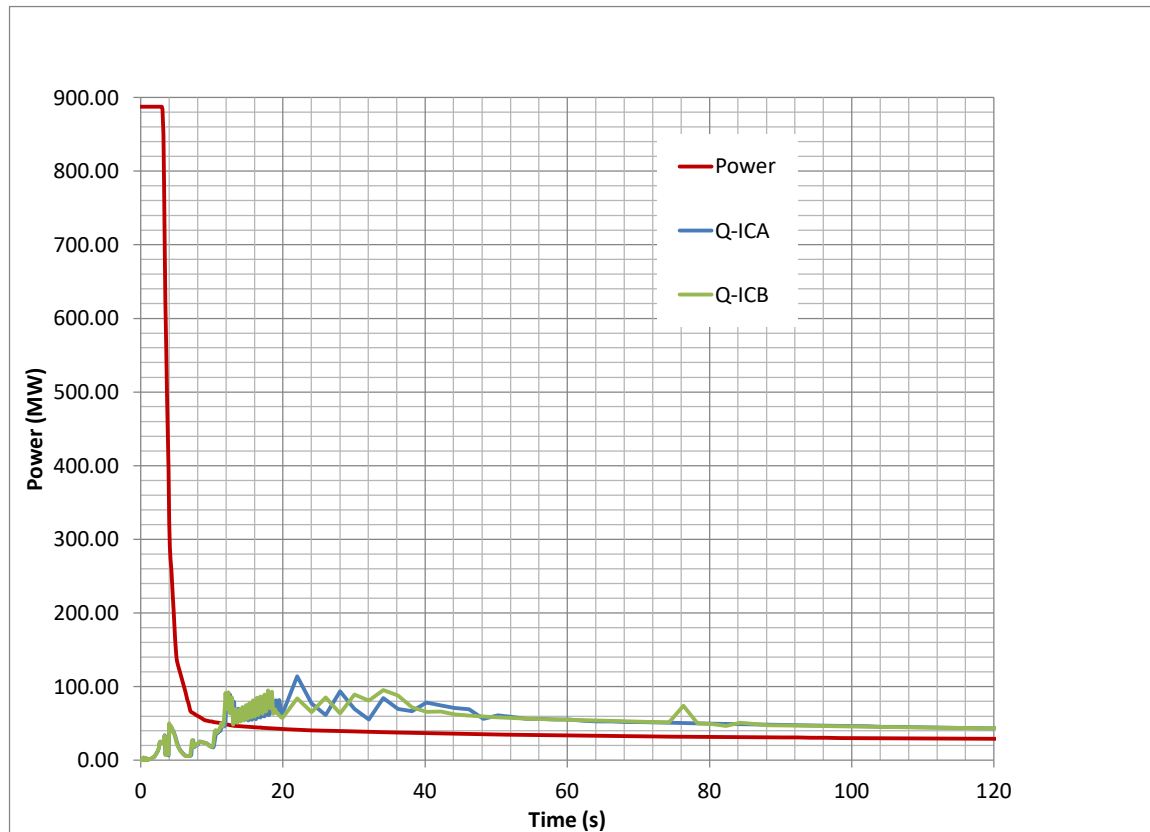


Figure 15.5-170: Reactor Power, Large FW Pipe Break, Conservative Case

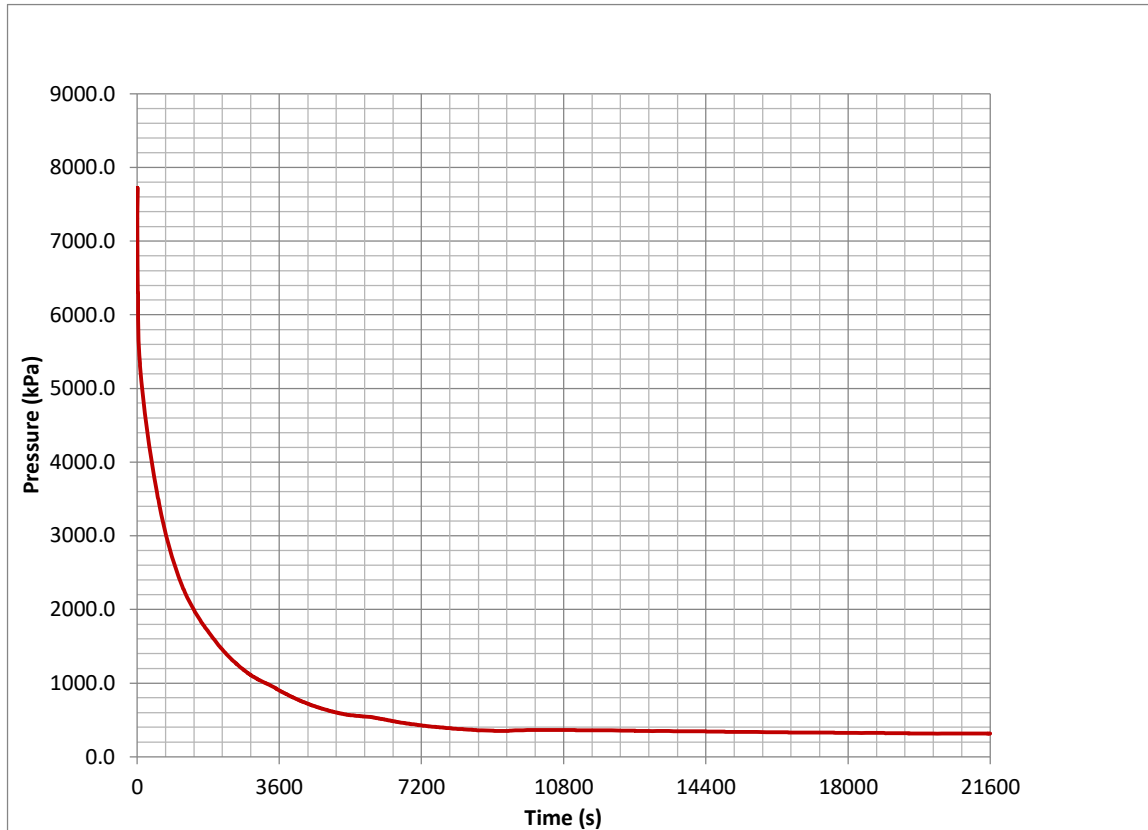


Figure 15.5-171: Reactor Pressure, Large FW Pipe Break, Conservative Case

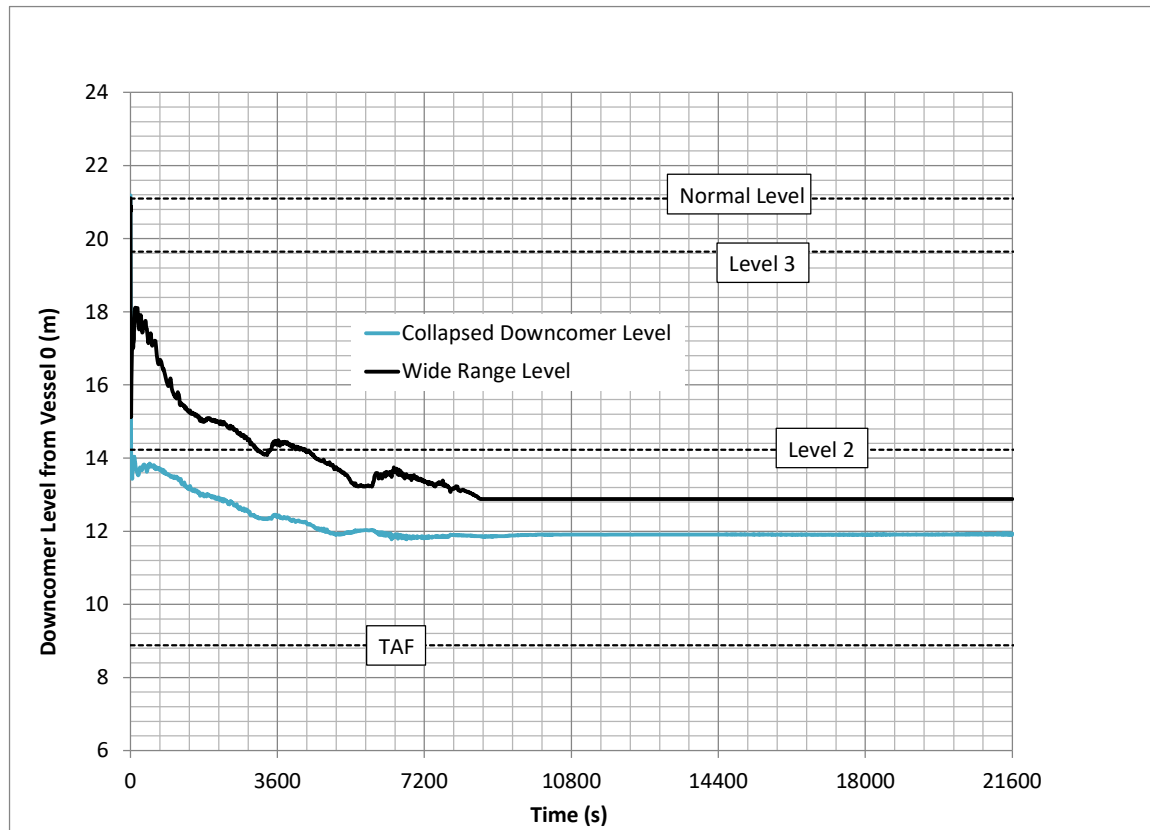


Figure 15.5-172: Reactor Water Level, Large FW Pipe Break, Conservative Case

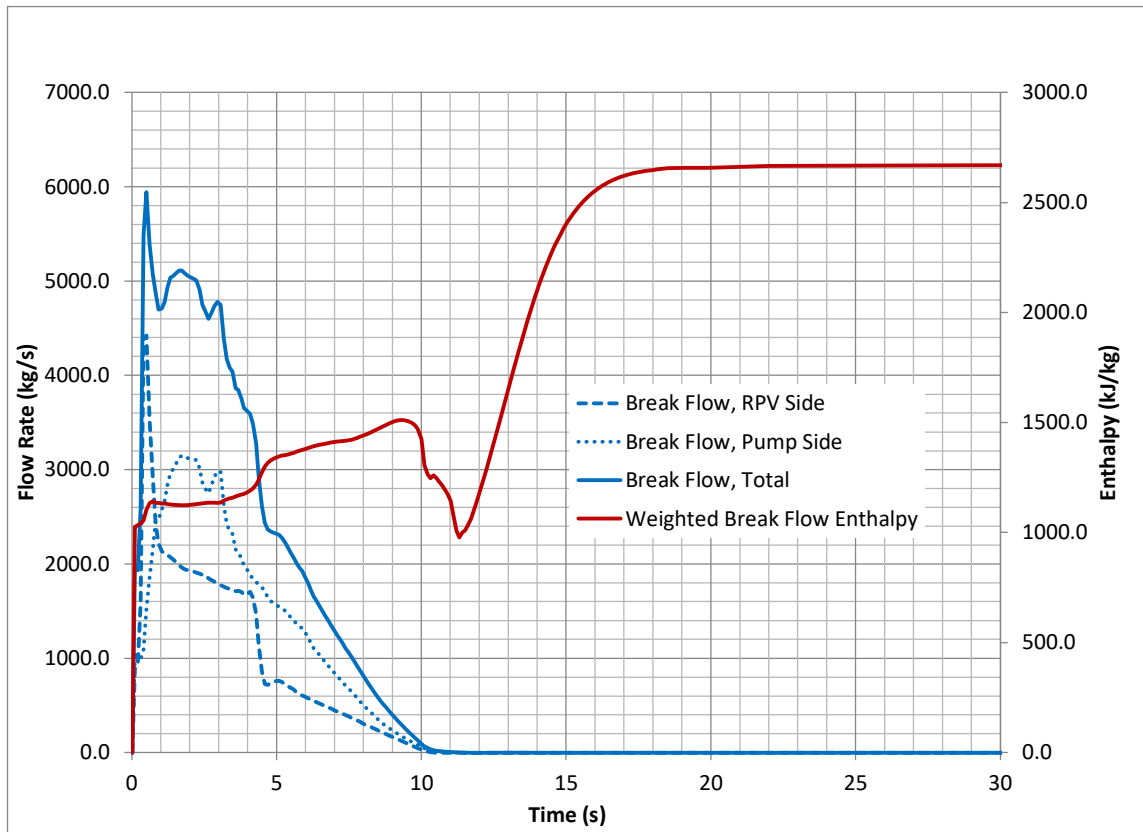


Figure 15.5-173: Break Flow Rate and Enthalpy, Large FW Pipe Break, Conservative Case

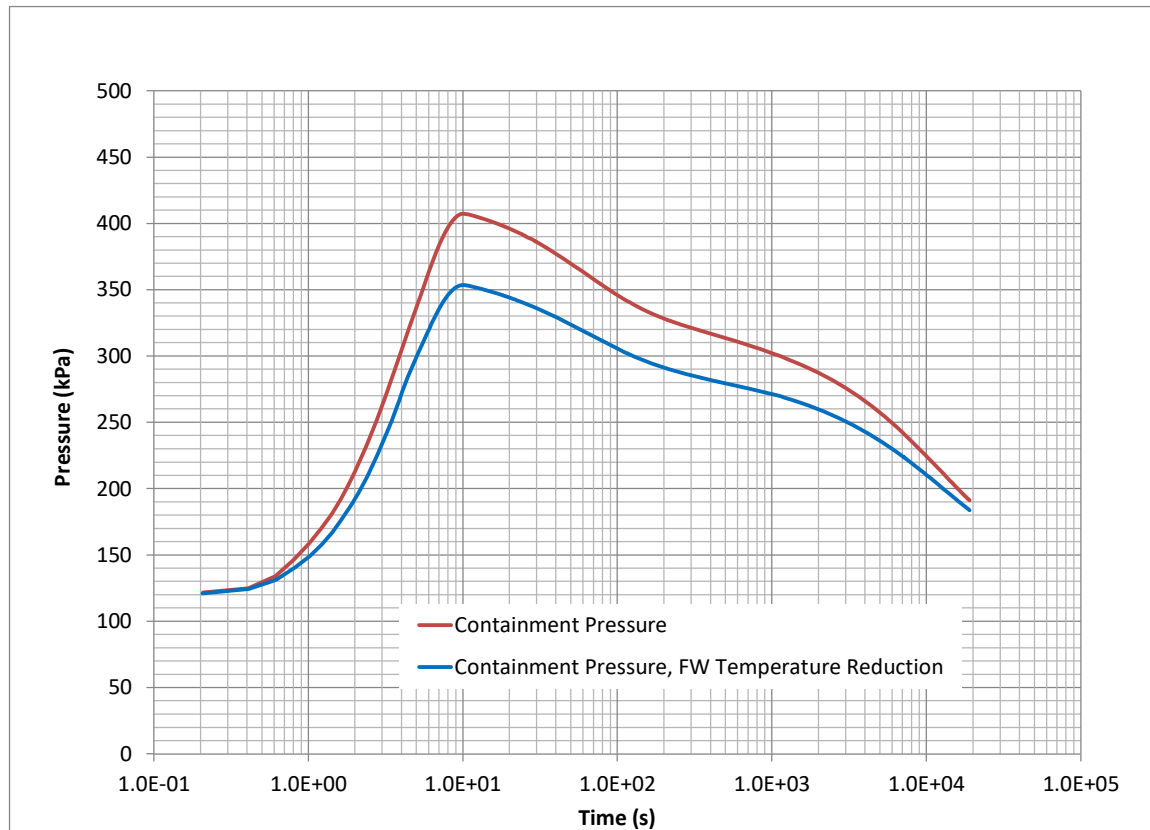


Figure 15.5-174: Containment Pressure, Large FW Pipe Break, Conservative Case

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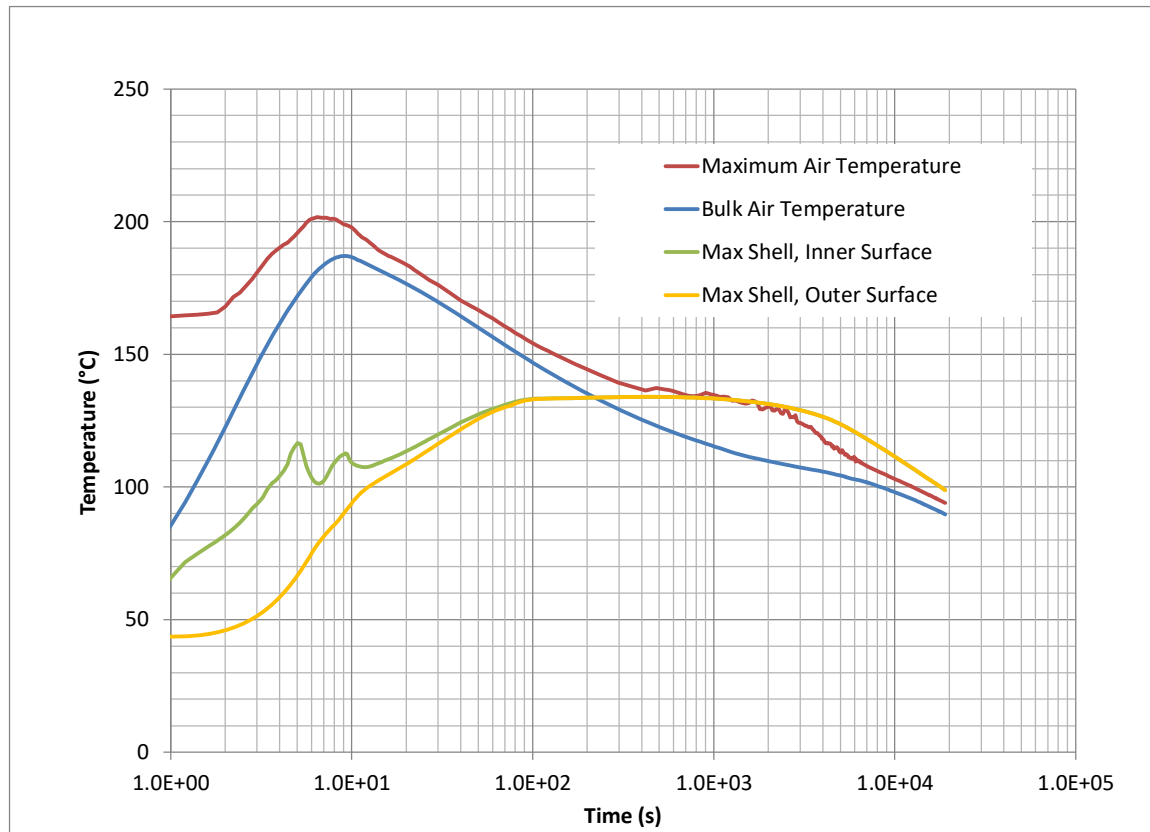


Figure 15.5-175: Containment Temperature, Large FW Pipe Break, Conservative Case



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Non-Proprietary Information

**Ontario Power Generation Inc.
Darlington New Nuclear Project
BWRX-300 Preliminary Safety Analysis Report:**

**Chapter 18
Human Factors Engineering**

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
0	All	Initial Release

ACRONYM LIST

Acronym	Explanation
AOF	Allocation of Functions
CBP	Computer-Based Procedures
CNSC	Canadian Nuclear Safety Commission
COO	Concept of Operations
COTS	Commercial-Off-The-Shelf
DCT	Data Connection Table
DNNP	Darlington New Nuclear Project
DSA	Deterministic Safety Analysis
EOP	Emergency Operating Procedure
FRA	Functional Requirements Analysis
GEH	GE-Hitachi Nuclear Energy
HED	Human Engineering Discrepancy
HF	Human Factors
HFE	Human Factors Engineering
HFEITS	Human Factors Engineering Issue Tracking System
HFEPP	Human Factors Engineering Program Plan
HPM	Human Performance Monitoring
HRA	Human Reliability Analysis
HSI	Human-System Interface
I&C	Instrumentation and Control
ISV	Integrated System Validation
MCR	Main Control Room
NUREG	Nuclear Regulatory Report
OE	Operating Experience
OER	Operating Experience Review
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
SAA	Severe Accident Analysis
SCR	Secondary Control Room
SPDS	Safety Parameter Display System
SSC	Structures, Systems, and Components

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Acronym	Explanation
T&E	Testing and Evaluation
TA	Task Analysis
TSV	Task Support Verification
UIS	User Interface Specification
V&V	Verification and Validation

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18.0 HUMAN FACTORS ENGINEERING

Preliminary Safety Analysis Report (PSAR) Chapter 18 presents the Human Factors Engineering (HFE) program for the BWRX-300 to demonstrate the adequacy of integration of HFE requirements and analysis results into the plant design. The program of HFE activities and analysis informing the design of the plant Structures, Systems, and Components (SSC) is based on clear definition of the full plant set of users and a clearly defined scope of application across the full plant design, operational modes, and lifecycle stages, with focus on important human actions. The HFE content for this PSAR chapter reflects the level of maturity of the HFE Program, plant design, and safety analyses at the time of submission. The pre-operational safety analysis report details further design and analyses development and summarizes HFE Program progression in support of the Licence to Operate submission.

Chapter 18 provides a summary of the BWRX-300 Human-System Interface (HSI) design goals and bases, analyses undertaken to understand the plant-specific HFE requirements related to task performance, the process for detailed HSI design, and activities supporting effective design implementation. The overall design and implementation process is described in detail in NEDC-33982P, "BWRX-300 Darlington New Nuclear Project (DNNP) Human Factor Engineering Program Plan" (Reference 18.1-1). The Human Factors Engineering Program Plan (HFEPP) presents the comprehensive, iterative design approach used for the development of human-centred interfaces and work environment for the plant.

Note that Section 18.1 provides an overview of the HFE Program and outlines its activities or "technical elements". The remainder of the chapter provides the details, including the scope and summary of methods, of the technical elements outlined in Section 18.1.

18.1 Management of the Human Factors Engineering Program

18.1.1 HFE Program Goals

The high-level goal of the BWRX-300 HFE Program is to conduct a proportionate, integrated, and effective set of HFE design activities that considers users and all phases of the plant lifecycle and that result in a design that reduces the risks and consequences related to human interactions with the plant to as low as reasonably achievable. The program was developed and conducted in line with multiple nuclear regulatory requirements, particularly those in Canadian Nuclear Safety Commission (CNSC) REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility" (Reference 18.1-2), CNSC REGDOC-2.5.1, "General Design Considerations: Human Factors" (Reference 18.1-3) and CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 18.1-4).

The HFEPP defines the program and outlines the way that the specified general human-centred HFE design goals are operationalized and verified during the design process. This is achieved, through the application of the HFE analyses, integrated design and safety analysis support, and provision of tools, technical requirements, and guidance to designers. The HFE Program ensures that the plant-level design goals are achieved, including:

1. Design of HSIs reduces the likelihood of error and provides for timely, clear error detection.
2. Tasks can be accomplished within time and performance criteria.
3. Allocation of Function (AOF) and proposed job design (staff complement and job roles) are such that a suitable level of human vigilance is ensured and acceptable workload levels that minimize periods of human underload and overload is provided.
4. Presentation of information supports a high degree of situational awareness of the state of the plant and actions required.

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5. HSI design supports the capability of personnel to recover from previous decisions and actions that did not achieve intended results.
6. Application of ergonomic principles to working areas and their environments ensure these areas are safe and designed to support performance of required tasks.

The above HFE goals are embedded into the design through specification of requirements derived from codes, standards, best-practice guidance, and plant-specific HFE analyses, and through integrated HFE team support to the design. Achievement of the goals is confirmed using the design tools, HFE Testing and Evaluation (T&E) throughout the design development, and HFE Verification and Validation (V&V) of the realized design.

18.1.2 Program Scope

The HFE Program scope applies to HSI components and SSC within or that form part of facilities, systems, equipment, and components throughout the plant. HSIs are defined as any region or point at which a person interacts with a system, equipment, or component. System interface means any digital and electronic Instrumentation and Control (I&C) user interfaces, as well as hardware-based user interfaces and design features on panels, equipment, and individual components. This includes HSIs within or forming part of:

- Control facilities for reactor operations
- Facilities for supporting response to accidents and emergencies
- Control room or stations for radwaste processing
- Control room or stations supporting refuelling and maintenance outage work
- Local control stations
- Equipment- and process line-mounted HSIs
- Auxiliary and support facilities and equipment located external to the main reactor and powerhouse buildings

This includes specification to and oversight of HSIs that form part of SSC supplied by external vendors, ensuring that supplied design or selection of standard equipment and components is consistent with the HFE requirements of the HFE Program.

The HFE Program also applies across the full scope of users and activities that support plant operation, testing, inspection, and maintenance, including functions such as fuel handling, chemistry, radioactive waste processing, and radiation protection.

The HFE Program described in this plan applies to design activities that consider Human Factors (HF) risks that might arise in all phases of the plant lifecycle, including:

- Construction
- Commissioning
- Operation
- Decommissioning

The HFE Program applies to all HSIs, including those at the following locations:

- Main Control Room (MCR)
- Secondary Control Room (SCR)

- Emergency Response and Support Facilities
- Radwaste Building Control Room or Control Stations
- Local Control Station interfaces
- Equipment- and process line-mounted interfaces (e.g., control actuators and gauges)
- HSIs related to auxiliary and support facilities located outside of the main buildings (e.g., hydrogen tanks or fuel oil supplies)

The HFE Program applies to all plant conditions in the design basis, including normal, outage (refuelling and maintenance outages, including extended refurbishments), abnormal, emergency, and accident conditions. The scope of the HFE Program is extensive; however, the application of HFE support and activities to the scope of each phase and task location is graded (or proportionate), as discussed in Subsection 18.1.4.2 to apply a higher level of emphasis and rigour for important human interactions that are safety-critical or hazardous.

Note that for some phases of the plant lifecycle, particularly Construction and Decommissioning, the HFE activities are focused at a high level. By nature of the single iteration of these plant stages outside of commercial operations, they generally do not involve analysis of recurring operationally related functions and tasks. HFE in design related to these non-commercial operational phases is centred around providing basic guidelines and ensuring design strategies that aid in the achievability of the overall goals of the phase. For example, for decommissioning, HFE design guidelines and requirements for maintainability may equally apply, especially clearance and access for removal of large components and equipment, and consideration of radiological safety through plant structures and equipment layout.

The same general HFE methodologies and tools described in this chapter are also applied to the HSIs related to security. A risk-based approach to HFE design requirements, task support requirements, and testing methodologies is also applied to security considerations. However, due to the sensitive nature of the specific details of security risk ratings, credited human actions, security success criteria, and testing scenarios, the HFE activities for security are found within the Security Annex.

The HFEPP at the time of this PSAR, and in support of the Licence to Construct application, is focused on the design of the plant. After plant turnover to the utility, the utility HFE Program is defined through suitable processes and procedures to address HFE licenced and operating plant activities such as Human Performance Monitoring (HPM), management of change for operational documentation, and design modification activities.

18.1.2.1 Overview of the Human Factors Engineering Program

The HFEPP describes the goals and scope of the HFE Program, along with items such as:

1. Assumptions and constraints in conducting the program
2. Coordination of the HFE Program with the overall plant design activities, including coordination with the plant safety analysis
3. Tools and facilities (e.g., mock-ups, computer simulations) used in support of the program
4. Composition, qualifications, and responsibilities of the HFE organization
5. Process and procedures followed including the process for identifying and managing technical and programmatic issues
6. Documentation developed

7. Summary of how the results of the HFE analysis are incorporated into the design, operational documentation, and safety analyses

The HFEPP defines each of the technical elements, the specific activities that comprise the full integrated program, as outlined in the next section.

18.1.2.2 Human Factors Engineering Program Technical Elements

The technical elements for the HFE Program are described briefly below. The full description of these elements, and how they constitute a comprehensive and robust program of HFE integration across the plant design, forms the remainder of this chapter (Sections 18.2 through 18.6).

1. *Operating Experience Review (OER)* – identification, review and incorporation of any recommendations and learning (positive and negative) from past events and user feedback related to HFE in design (Subsection 18.2.1)
2. *Functional Requirements Analysis (FRA)* – determination of functions required to achieve plant goals in all plant states (Subsection 18.2.2)
3. *Allocation of Function (AOF)* – assigning the identified functions to system (technology) or human, based on respective capabilities and limitations of each (Subsection 18.2.3)
4. *Task Analysis (TA)* – identification of the tasks required to achieve the allocated functions, and decomposition into task steps to allow the identification and characterization of HSIs, personnel, locations, and support equipment (e.g., communications, lighting, personnel protection) required to perform each task successfully (Subsection 18.2.4)
5. *Staffing Analysis* – determination of the numbers and roles of personal required to support optimal task performance in all plant conditions (Subsection 18.2.5)
6. *Treatment of Important Human Actions* – activities supporting and providing input to the BWRX-300 safety analyses to ensure clear identification of human actions important to safety, ensure claimed actions are achievable and identify HSIs requiring the highest level of HFE focus and effort (Subsection 18.2.6)
7. *Human-System Interface (HSI) Design* – identification and management of the set of HFE design requirements from standards, codes, and best-practice guidance, and implementation of those requirements plus results from HFE analyses into the design of HSIs, including integration of HFE team design support; also includes HFE T&E activities (Subsection 18.3.1 through 18.3.6)
8. *Procedures* – process and activities for the development of usable and validated operational documentation, for plant task types (Subsection 18.3.7)
9. *Training and Qualifications* – process and activities for the development of relevant and validated training content, optimized for the plant design, operational documentation, and baseline personnel qualifications and attributes (Subsection 18.3.8)
10. *Human Factors Verification and Validation (V&V)* – detailed, staged set of activities to provide assurance of the correct and sufficient implementation of HFE requirements in the design, and the appropriate design to support required tasks (Section 18.4)
11. *Design Implementation* – support and monitoring of the design from “on paper” to a realized constructed plant, including integration with configuration control to ensure no loss of integrity of the HFE Program goals throughout fabrication and construction (Section 18.5)

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12. *Human Performance Monitoring (HPM)* – continuous monitoring of user task performance throughout the lifetime of the plant to ensure optimum plant and organizational design and identify early trends and issues that require HFE design improvements (Section 18.6)

NOTE: This element is only relevant within the future operational plant HFE Program and therefore not currently included in the HFEPP.

The scope and nature of the HFE Program, the HFE organization undertaking the program and the technical elements that comprise it, align with regulatory requirements and international standards and guidance, providing assurance that HFE has been suitably and sufficiently integrated into the plant design.

Table 18.1-1 illustrates the alignment of Chapter 18 and the HFE Program technical elements, with the CNSC expectations for HFE, as per CNSC REGDOC-2.5.1 (Reference 18.1-3) and CSA N290.12-14, “Human factors in design for nuclear power plants” (Reference 18.1-5). The PSAR and HFE Program technical elements inform the subsequent pre-operational safety analysis report and ultimately the overall plant lifecycle of HFE activities.

Although arranged somewhat differently in the PSAR chapter and HFE Program, all required elements are included and addressed. The elements are performed in an iterative manner, with activities and outputs progressively evolving with the design and related safety analyses. These elements inform one another, inform, and are informed by plant design and safety analyses, and are aligned with design and engineering processes and requirements, international best-practice guidance, and regulations.

Table 18.1-1: Mapping of HFE Program Elements with CNSC and CSA Requirements

ELEMENT NO.	PSAR CONTENT	CNSC REGDOC-2.5.1	CSA N290.12-14
--	18.1 Management of the Human Factors Engineering Program	<p>HFEPP, including: Goals, Scope, Background, Criteria for Areas of Consideration</p> <p>Human Factors Input, including: HFE Organization Roles and Responsibilities, Training Needs and Related Groups</p> <p>Methods, including intended tools and technical guides</p> <p>HFE Processes and Procedures</p> <p>Timelines, including logical links to related project activities</p> <p>Documentation</p> <p>Disposition of Human Factors Issues</p>	<p><i>HF Planning:</i> Determine methods, analyses, evaluations, project interfaces, and tools</p> <p>Identify constraints and drivers</p> <p>Graded approach based on risk and complexity</p> <p>Organization and resources</p> <p>Communications</p> <p>Source documents</p> <p>Issue identification and resolution</p> <p>Documentation</p> <p>Scheduling</p> <p>HFE Interfaces with other groups</p>
--	18.2 Human Factors Engineering Analysis		
1	18.2.1 Review of Operating Experience	Operating Experience Review	<i>HF in Concept Design:</i> OER
2	18.2.2 Functional Requirements Analysis	Functional Analysis	<i>HF in Concept Design:</i> functional analysis
3	18.2.3 Allocation of Function	Allocation of Function	<i>HF in Concept Design:</i> functional analysis (definition includes AOF)
4	18.2.4 Task Analysis	Task Analysis, Job Design	<i>HF in Preliminary Design:</i> TAs including workload and communications analysis; link analysis
5	18.2.5 Staffing	Staffing & Minimum Shift Complement, Job Design, Shift-Work Systems	<i>HF Interfaces:</i> HF in design shall consider the interfaces with staffing; the information common to both HF in design and interfacing disciplines, such as staffing analyses and strategies, should be shared.

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ELEMENT NO.	PSAR CONTENT	CNSC REGDOC-2.5.1	CSA N290.12-14
6	18.2.6 Treatment of Important Human Actions	Human reliability, activities with potentially hazardous human interactions	<p><i>HF in Concept Design:</i> identification of scenarios to be analyzed</p> <p><i>HF in Preliminary Design:</i> participation in the assessment of human actions and error consequences; assessment of the feasibility of human actions in the deterministic safety analyses</p> <p><i>HF in Detailed Design:</i> confirmation of the feasibility of human actions important to safety in the probabilistic and deterministic safety analyses; analyses to confirm the ability of the human to perform necessary actions</p>
7	18.3 Design of the Human-System Interface 18.3.1 Design Goals and Design Bases	Design human-machine interface system; design physical working environment	
7	18.3.2 Human-System Interface: Design Inputs		<p><i>HF in Concept Design:</i> a statement of system operational purpose and operational requirements under all anticipated conditions; development or selection of HF in design source documents; identification of SSC requirements to support necessary human actions; HFE assessment of design concepts and options</p> <p><i>HF in Preliminary Design:</i> document high-level HF-related requirements; input to specifications and bid evaluations; requirements derived from HF analysis results</p>
7	18.3.3 Human-System Interface: Detailed Design and Integration		<i>HF in Detailed Design:</i> detailed HSI design; design integration of Commercial-Off-The-Shelf (COTS) products

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ELEMENT NO.	PSAR CONTENT	CNSC REGDOC-2.5.1	CSA N290.12-14
7	18.3.4 Human-System Interface: Tests and Evaluations		<i>HF in Preliminary Design:</i> modeling, mock-ups, or prototyping of user interfaces; evaluations <i>HF in Detailed Design:</i> Usability testing
7	18.3.5 Human-System Interface: Design of Main Control Room		Covered within all HF in Design activities
7	18.3.6 Human-System Interface: Design of Secondary Control Room		Covered within all HF in Design activities
8	18.3.7 Procedure Development	Procedure Development	<i>HF in Detailed Design Stage:</i> HF analyses output to development of training manuals, operating procedures, and commissioning procedures
9	18.3.8 Training and Qualification Program Development	Training Program Development	<i>HF in Detailed Design Stage:</i> HF analyses output to development of training manuals, operating procedures, and commissioning procedures
10	18.4 Human Factors Engineering Verification and Validation	Verification Validation	<i>HF in Detailed Design Stage:</i> Verification (carried out before the design is released for construction) Validation (validation activities split between detailed design and implementation)
11	18.5 Design Implementation (post-construction)	Design Implementation	<i>HF in Design Implementation Stage:</i> HFE during installation and commissioning
12	18.6 Human Performance Monitoring (at start of testing)	Human Performance Monitoring	N/A (scope of standard is design stages only)

18.1.3 Team and Organization

The HFE team consists of a core and extended team dedicated to integrating HFE requirements and principles into the design. The core HFE team sits within the organization as a separate engineering team, at an equal level to all other discipline teams. The HFE team holds the technical authority over HFE activities and requirements and has the equal authority and issues resolution mechanisms as any other engineering team. The extended HFE team includes members from other disciplines within the engineering organization. This ensures fully integrated and timely consideration of HFE in the daily engineering design decisions and activities.

The core HFE team is comprised of an HFE Technical Lead and two general roles: HF Engineer or HFE Specialist and HFE Operations/Maintenance. The qualifications for these roles may be met by individuals or collectively by the HFE team. The responsibilities and qualifications of these roles are defined in the HFEPP as summarized below:

1. Technical Lead: Provides technical and program oversight and review; responsible for ensuring that HFE activities, interfaces, and outputs meet HFE requirements and align with HFE Program objectives; point-of-contact for schedule development, integration, and management of the program. This role is expected to have the base qualifications of either the HFE Specialist or HFE Operations/Maintenance role, with additional HFE capability across a breadth of HFE competence areas suitable for the full scope of the HFE Program and experience in project management and managing HFE or other technical, cross-cutting programs.
2. HF Engineer/HFE Specialist: Provide specialized knowledge of human cognitive and physical capabilities and limitations, applicable HFE design and evaluation practices, and HFE principles, guidelines, and standards; develop and perform HFE analyses; identify and participate in the resolution of identified HFE issues and non-compliances. This role requires a bachelor's degree in HFE, Engineering Psychology, or related science with four years of cumulative experience related to the HFE aspects of HSIs (design, development, and T&E), particularly modern digital process control HSIs, and four years of cumulative experience related to the HFE aspects of workplace design.
3. HFE Operations/Maintenance: Provide knowledge of operations and maintenance activities, including task characteristics, HSI characteristics, environmental characteristics, and technical requirements related to operational activities, and apply those insight in support of activities such as development of HSIs, procedures, and training programs. Participate in the development of scenarios for Human Reliability Analysis (HRA) evaluations, task analyses, HSI T&E, validation, and other evaluations. This role requires a bachelor's degree in a technical field; experience as a senior authorized reactor operator, or as a qualified maintenance technician; five or more years of plant experience, preferably in Boiling Water Reactors exposure to plant procedure development, personnel training, and operational nuclear plant programs; and two or more years of experience in one or more areas of HFE analysis, design, T&E, and HFE V&V.

The responsibilities of the HFE team are to establish and perform the activities as defined in this PSAR chapter throughout the design lifecycle to ensure that the facilities, systems, equipment, and tools are designed to be compatible with the capabilities, limitations, and needs of the human. The specific duties of the HFE team are to guide, perform, and support the analysis and design activities, ensuring the execution and documentation of all steps in the activities are performed in accordance with the established program and procedures.

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The HFE team is responsible for:

1. Development of HFE plans and procedures including management of any identified HFE issues
2. Oversight, participation in, and review of HFE design, safety analyses, development, T&E activities, including identification of in-process HFE issues
3. Recommendations for, and support to implementation of design-based resolutions for issues identified during the implementation of the HFE requirements and analysis results
4. Verification of correct and robust implementation of HFE requirements, analysis results and issue resolution into the design
5. Assurance that HFE activities comply with HFE plans and procedures
6. Managing documentation of HFE activities and issues management
7. Plan and implement HSI design configuration control during design implementation

To ensure suitably qualified and experienced persons are performing the work, HFE Program activity assignments are allocated by the Technical Lead based on the team member's role and their specific experience. For example, not all the team HF Engineers/HFE Specialists are experienced in HFE safety analysis, identifying, and evaluating important human actions. Only those team members with adequate experience and training (if applicable) in a particular HFE activity will be assigned to those activities. To ensure there are no singleton specialisms within the team and associated vulnerability to knowledge loss, more than one team member will be required to be deemed competent for each activity. Team members that do not meet the full qualification of an HFE team role, or who are not deemed suitably qualified for a specific activity, will receive mentoring and technical oversight to support developing the skills required for the role or work assignment.

18.1.3.1 Cross-Discipline Support and Integration

Due to the cross-functional nature of a completely integrated HFE design process, HFE activities interface with many other disciplines. In addition, the other disciplines act as extended parts of the HFE team for some aspects of the HFE Program implementation. The integration of related groups with HFE is formally addressed through an integrated detailed schedule, as well as through the HFE technical project management role of the HFE Technical Lead.

Specifically, work activities that require integration of HFE and other disciplines are entered into the resource-loaded schedule by the HFE Technical Lead with all required resources, including those from other teams. Pre-job briefs are held for each group of activities, or workplan, to clearly define the relevant resource roles and responsibilities for the completion of the work. Further detail on GEH and the BWRX-300 project design processes, including inter-discipline communications and issues resolution, are described in Chapter 17, Sections 17.2 and 17.3.

The descriptions of the following disciplines and groups and their contributions to HFE are representative based on best-practice HFE integration principles. The actual engineering design team disciplines may vary, but the scopes described are covered.

1. Mechanical Engineering and Electrical Engineering
 - a. Provide knowledge of the purpose, operating characteristics, and technical specifications of major plant systems
 - b. Provide input to HFE analyses, especially function and task analyses

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- c. Allocate and implement HFE requirements and recommendations relevant to their scope of design, including management of those requirements provided to suppliers
 - d. Participate in developing scenarios for use in TA, validation, and other analyses
2. I&C Engineering
- a. Provide detailed knowledge of the HSI physical design, including control and display hardware selection, design specification, functionality, and installation
 - b. Support HFE design of information display design, content, and functionality, particularly connection to the underlying I&C platform
 - c. Participate in designing, developing, testing, and evaluating the HSIs
 - d. Provide knowledge of data processing associated with displays and controls
 - e. Allocate and implement HFE requirements and recommendations relevant to their scope of design, including management of those requirements provided to suppliers
 - f. Participate in designing and selecting HSI components, such as controls and displays
 - g. Participate in developing scenarios for HRA, validation, and other analyses involving failures of the HSI data processing systems
3. Civil/Structural Engineering
- a. Provide knowledge of the overall structure of the plant, including performance requirements, design constraints, and design characteristics of the following:
 - Containment structures (i.e., Steel-Plate Composite Containment Vessel)
 - Control Rooms (main and secondary)
 - Local control
 - b. Provide knowledge of the configuration of plant components
 - c. Allocate and implement HFE requirements and recommendations relevant to their scope of design, including management of those requirements provided to suppliers
 - d. Provide input to plant analyses, especially function analysis, TA, and development of scenarios for TA and validation
4. Plant Integration Engineering
- a. Prepare the Deterministic Safety Analysis (DSA) establishing the SSC and human actions that are credited for successful event mitigation
 - b. Provide knowledge of maintenance, inspection, and surveillance activities based on previous plant design and consideration of evolving new plant design, including:
 - i. Development of maintenance and outage strategy and plan documents
 - ii. Expected tasks
 - iii. Relevant SSC and HSIs
 - iv. Task performance requirements
 - v. Workspace environment characteristics
 - vi. Technical information related to the conduct of these activities

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- c. Allocate and implement HFE requirements and recommendations relevant to their scope of design, including management of those requirements provided to suppliers
 - d. Support Plant Architectural and HFE design, development, and evaluation of the control facilities and other HSIs throughout the plant to provide reasonable assurance that each can be inspected and maintained to the specified reliability
 - e. Provide input regarding maintainability and inspectability during the development of procedures and training
 - f. Participate in the development of scenarios for HSI evaluations including task analyses, HSI design tests and evaluations, and validation
5. Risk and Reliability Engineering
- a. Perform Probabilistic Safety Assessment (PSA) and HRA to quantify the human contribution to risk and inform HFE analyses
 - b. Provide knowledge of plant component and system reliability and availability and assessment methodologies to the HSI development activities
 - c. Participate in the development of scenarios for HSI evaluations, especially validation
 - d. Provide input to the design of HSIs to provide reasonable assurance it meets reliability goals during operation and maintenance and maintains the specified availability
6. Simulation Assisted Engineering
- a. Develop the simulators for HFE T&E and V&V activities

18.1.4 Process and Procedures

18.1.4.1 Coordination and Documentation of Activities

The HFE Program is planned and conducted in accordance and alignment with overarching design and quality program processes and procedures, within accredited quality management systems as described in Chapter 17. The work is performed in an integrated manner with HFE as an equal design discipline, whose cross-cutting requirements and support are embedded within all other design disciplines, with activities, inputs, outputs, and dependencies coordinated through a detailed schedule and associated schedule management processes. The schedule includes activities and deliverables for all disciplines and orders them with logical connections to ensure they are completed in the required sequence.

To help ensure cross-discipline communication and coordination, activities include scheduled periodic formal design reviews conducted by representatives of each discipline. Additionally, deliverables are completed in accordance with a deliverable standard, which specifies the required content from all related disciplines, dictates the format for consistency and quality and specifies the required discipline reviewers for each document. This includes other disciplines reviewing and incorporating outputs from HFE documentation and the HFE team reviewing and incorporating outputs from other disciplines, as appropriate and according to the plan.

The documentation for the HFE Program uses a standard design process that includes documenting internal design records to capture inputs and outputs, as well as providing the basis for formal deliverables. The information in the design records is incorporated into the design by HFE and other disciplines as appropriate and in accordance with the BWRX-300 project work breakdown structure. A full description of the management and integration of HFE activities within the project is described in the HFEPP.

In this way, using the project processes and HFE coordination measures described in the HFEPP, the design activities related to HFE are conducted and documented such that design basis, input maturity and rationale for design and analysis scope is provided for HFE design decisions and analysis results. HFE requirements and recommendations are addressed by the requirements management process (Subsection 18.1.4.3), and either incorporated into the design directly or via alternate solutions agreed as acceptable by HFE, or if not implemented, tracked as an HFE issue as described in Subsection 18.1.5.

18.1.4.2 Risk-Based Graded Approach

A graded (or proportionate) approach to HFE is applied to the conduct of activities within the HFE Program, to provide the appropriate focus for analysis and design. The graded approach provides basic HFE attention to human interactions within the system and provides emphasis and more detailed, rigorous HFE effort on aspects of the plant design related to HSIs used to perform human actions important to safety, or tasks that are novel, complex, or inherently hazardous. The approach uses a risk-based grading system to grade each of the tasks or human actions identified throughout the plant based on four key risk categories:

- Nuclear Safety
- Personnel Safety
- Asset Protection
- Generation Capability

Although Asset Protection and Generation Capability are typically discounted as relating to safety, the HFE Program recognizes that equipment damage creates requirements for forced outages and corrective maintenance, as well as impacting production goals, all of which has an indirect impact on personnel and nuclear safety. Consideration of aspects of equipment protection, reliability, and production risks, also ensures that tasks outside of reactor operations have a decreased likelihood of being classified as Low-Risk Level. Loss of power generation and production shortfalls equate to loss of income for the plant which is recognized to have a direct effect on plant condition, organizational health, and ultimately, nuclear safety culture.

The overall risk level for the human action is determined by the highest risk level assigned to each of the four categories. The base risk level is then used to assign a minimum HFE Application Level. The minimum HFE Application Level dictates the minimum degree of application when considering each HFE technical element. Further detail is provided in Subsection 18.2.3.

In addition to the formal task grading method for determining proportionate effort during design, the scope and level of effort of HFE activities is also proportionate to project lifecycle risk and change management considerations. This is done by applying greater scope, focus and degree of support on HFE activities that occur earlier within the design lifecycle, when changes are more effectively and easily managed. For example, while not all HSIs receive a full HFE V&V, all HSIs receive some degree of HFE support during the design phase.

18.1.4.3 Requirements Management

HFE requirements management is performed in accordance with a requirements management process that is standardized and controlled across the entire plant design, as described in Chapter 17, Subsection 17.3.1. Requirements management and traceability is developed and maintained to:

- Ensure HFE requirements relevant to each scope of work are clearly identified, allocated, communicated, and understood by all relevant project personnel

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- Outline how the requirements are met
- Reference evidence that demonstrates compliance has been achieved

HFE requirements are categorized and dispositioned as follows:

- A. Process Requirements – Requirements related to how the HFE Program is conducted and the interfaces among HFE and other disciplines. HFE Process Requirements are allocated to the HFEPP and are fulfilled via the construction and implementation of the architecture and processes defined within this document. As such, all HFE Process Requirements are allocated to and end with the HFEPP.
- B. Product Requirements – Requirements for the design of, or provision for plant workspace and environmental attributes, SSC, and HSIs. Product requirements are either derived from HFE design standards, codes, and guidance, or generated from HFE analyses as required to support successful task performance in the specific context of plant conditions. These requirements are implemented through design requirements specifications or design records communicated to and implemented by the relevant design teams. Requirement traceability and HFE support level is proportionately applied based on the assessed risk-based grading (Subsection 18.4.2).

Where HFE inputs are not in agreement with one another or where they conflict with other discipline design requirements, precedence is given as follows:

- Laws and Regulations
- Regulatory Requirements and Guides
- Requirements related to supporting, maintaining, or recovering the plant in a safe state
- Nuclear Standards
- Nuclear Industry Guidance Documents
- Non-Nuclear Codes, Standards, and Guides

Conflicts between HFE and other design requirements are resolved with the HFE team using the HFE issue resolution process described in Subsection 18.1.5. Decisions regarding trade-offs and design optimizations are conducted within the integrated design process in accordance with standard GEH design procedures.

18.1.5 Issue Tracking

Included in the HFE Program is the establishment and maintenance of an on-going HFE Issues Tracking System (HFEITS) for documenting HF issues that may be identified throughout the full scope of HFE activities, and the actions taken to resolve those issues.

The HFEITS is used to capture issues related to design and implementation HFE activities and specific Human Engineering Discrepancies (HEDs) identified through HFE V&V. The tracking of issues include:

1. Evaluation of each issue/HED to determine significance and whether it warrants correction when evaluated in the context of the integrated plant design
2. Identification of appropriate solutions to address issues/HEDs, including, as appropriate, changes to HSI design, procedures, staffing/qualifications, or training
3. Verification that the solutions implemented to address the issue/HED resolve the problem without generating additional issues/HEDs

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4. Documented traceability of the issue/HED resolution process and identification of residual risks associated with it, if necessary

The HFEPP details the HFEITS development and management including:

1. Responsibilities for HFE team members in identifying HFE issues and HEDs
2. The process and criteria for including HFE issues and HEDs within the HFEITS, as opposed to resolving non-compliant design through integrated design teamwork in normal workflow
3. The process for evaluating the priority and adequate resolution of the issue/HED
4. The means for confirming acceptable resolution of the issue/HED, based on the nature of the issue, its priority, and the plant lifecycle stage where the resolution occurs

18.1.6 References

- 18.1-1 NEDC-33982P, "BWRX-300 Darlington New Nuclear Project (DNNP) Human Factor Engineering Program Plan," GE-Hitachi Nuclear Energy Americas, LLC.
- 18.1-2 CNSC Regulatory Document REGDOC-1.1.2, "Licence Application Guide: Licence to Construct a Reactor Facility."
- 18.1-3 CNSC Regulatory Document REGDOC-2.5.1, "General Design Considerations: Human Factors."
- 18.1-4 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 18.1-5 CSA N290.12-14, Human factors in design for nuclear power plants, Canadian Standard Association.

18.2 Human Factors Engineering Analysis

In this section, six of the HFE Program technical elements are described in depth. The remaining elements are described in other sections. These six elements described in this section include:

- Review of Operating Experience (OE)
- Function Requirements Analysis
- AOF
- TA
- Staffing
- Treatment of Important Human Actions

These are addressed in Subsections 18.2.1 through 18.2.6.

Integrated HFE involvement in, and support of the plant safety analyses also informs these elements. As described in Subsection 18.2.6, the HFE safety analyses activities identify, characterize, and substantiate the human actions that are performed to maintain the plant within or bring it back to a safe state, as described in Chapter 15, Safety Analysis.

18.2.1 Review of Operating Experience

The HFE Program includes the early review of OE to identify applicable HFE issues related to process or personnel safety that can be resolved through design improvements. The issues and lessons learned from the OER provide a basis for improving the plant design in a timely way (i.e., at the beginning of the design process). In addition to the early HFE review of existing OE, the project as a whole has a formal OE identification and management process, which includes HFE team participation in both identifying any OE and implementing HFE-allocated OE items in the HFE Program and design.

18.2.1.1 Objectives and Scope

The objective of the OER is to obtain information and lessons learned from experience to support design of BWRX-300 SSC. OE related to the following areas are considered in the development of the plant design:

- Predecessor plant(s) and systems
- Experience in industries with applicable SSC
- Applicable Industry HSI design experience
- Risk-important human actions
- Specifically identified applicable industry issues
- Issues identified by predecessor or similar plant personnel
- Specifically identified positive features that support task performance

18.2.1.2 Methodology

The OER process includes the following:

- Identification of applicable OE sources, leveraging work previously performed for predecessor plant designs

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- Specific methods for gathering sources and performing reviews of sources since the predecessor plants and systems OER
- Establishing a systematic framework and performing systematic searches of OE sources
- Obtaining and incorporating personnel feedback from predecessor or similar types of reactors
- Conducting reviews of human actions from predecessor designs that are similar to human actions included in the plant safety analyses
- Analyzing and consolidating raw OE data into OE Item Summaries on the OER Capture Sheet
- Allocation of each OE item to the HFE activity or document in which the OE item is dispositioned
- Recipient OE item review and allocation acceptance
- Documentation of the findings within an OER report

Existing and new OE is reviewed by HFE, and relevant, applicable problems, issues, and positive insights are identified and addressed throughout the design process. The OER information is made available to design engineers to support development of design features that are expected to reduce human error. Likewise, positive features of previous designs are communicated so that they can be retained.

18.2.1.3 Results

The results of the OER are summarized in an OER report. The report provides the OER process description along with the review methods that were used. The results include:

1. Sources of OER information
2. Summaries of OER issues and improvements
3. List of issues from the OER requiring special attention in the design process based on the grading process
4. Information gathered from personnel interviews conducted at predecessor plants

Implementation of the OER results into the design is managed and tracked through the assignment of and reference to a unique OE identification number. Communication and allocation of the results to the appropriate design team and design requirements document is managed by the HFE team.

The HFEPP provides additional details of the OER activity.

18.2.2 Functional Requirements Analysis

18.2.2.1 Objectives and Scope

FRA is performed to define the necessary functions that enable the achievement of the plant safety and commercial goals. These principal design requirements are necessary to meet plant goals and objectives in all normal and postulated accident conditions. They include meeting regulatory and customer requirements that are documented in plant and product level design requirements specifications.

18.2.2.2 Methodology

FRA is conducted as an integral part of the overall engineering design process specified by standard design process documents, and in particular part of the BWRX-300 requirements management process. The requirements management process as described in Chapter 17, Subsection 17.3.1 consists of the following activities, which apply equally to all types of requirements including the functional ones: elicitation, analysis, documentation, allocation (to specific system(s)), tracing, and requirements V&V. This process is a multi-discipline activity jointly undertaken by all design teams, including the HFE team. Note that, because BWRX-300 is an evolutionary plant with some of the same or similar plant- and system-level safety and performance goals, FRA elicitation is not required to the extent it would be for a completely new design. Eliciting functional requirements from existing design is done through importing functions that exist in current plant designs and are expected to apply to BWRX-300.

18.2.2.3 Results

The result of the multi-disciplinary FRA activities is the definition of the full set of functions that support achievement of the plant goals and can be traced to the principal design requirements of the BWRX-300. Plant- and system-level requirements documents list the functional requirements associated with each system. Through the requirements management process described in Chapter 17, Subsection 17.3.1 the project has also captured system functional performance requirements which includes the full list of plant functions, with reference and traceability back to the source documents where the requirements were elicited.

These functions from the FRA are captured in a report that also includes the results from the AOF (see next section) and assigning HFE Application Levels to tasks (Subsection 18.1.4.2). The results are input into the AOF process as described in the next section. The output from the FRA and AOF process also contains the full list of functions with characterizations of relevance to HFE, particularly those required to perform AOF and feed to the TA and HSI design activities.

18.2.3 Allocation of Function

18.2.3.1 Objective and Scope

AOF establishes a plant control scheme that enhances plant safety and reliability by taking advantage of human and system strengths and avoiding human and system limitations. The overall allocation can also enhance plant performance and safety by specifying overlapping and redundant responsibilities to the human and system.

The AOF strives to provide personnel with groups of logical, coherent, and meaningful tasks within their capabilities, and ensures a design that maintains human vigilance and situational awareness for any functions allocated to the system. The goal of the AOF is to provide acceptable workload levels per job role that minimize periods of human underload and overload to the extent possible. This is done through review of the initial allocation as a whole and using expert judgement to determine if the assigned functions per job role are suitable and sufficient. Further analysis of workload and requirements for situational awareness are then undertaken through downstream activities such as TA, HFE T&E and HFE V&V.

The AOF also allows the risk-based task grading, which determines the HFE proportionate, graded approach to activities, as described in Subsection 18.1.4.2.

18.2.3.2 Methodology

The AOF process for BWRX-300 is based on the relevant best-practice methodology presented in IAEA-TECDOC-668, "The role of automation and humans in nuclear power plants" (Reference

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18.2-1). The methodology identifies functions that should not be assigned to humans due to criteria such as:

- Physical demands (forces, posture)
- Cognitive demands (multitasking, stress, situational awareness, and vigilance)
- Combination of physical and cognitive demands (accuracy, response time)
- Environmental conditions (temperature, radiation)

The AOF process also uses criteria from USNRC NUREG/CR-2623, "The Allocation of Functions in Man-Machine Systems: A Perspective and Literature Review," (Reference 18.2-2) that limit or preclude human participation in a function or, conversely, that make human participation mandatory. These combined criteria form the top-level, overriding criteria in the AOF process.

The FRA process provides the list of functions input to the AOF, as described in Subsection 18.2.2. The AOF process is composed of two stages: the first stage establishes an initial (hypothesized) allocation; the second stage evaluates the hypothesized AOF to determine its adequacy and validate that it is optimized within the larger integrated task performance and work environment.

The initial AOF is determined following a formal decision flow with key allocation criteria informing each decision point. The decisions are based on expert judgement formed by a panel that includes an HF Engineer/HFE Specialist, an HFE Operations/Maintenance representative, a Plant Integration Engineering representative, and a System Engineer for the respective system. The expert panel accounts for the mandatory criteria when hypothesizing the initial allocation. The panel also makes use of OE to determine how functions were allocated in previous or similar applications and evaluate how they have performed.

The second stage of the AOF process is the AOF evaluation. This stage is performed later in the design and HFE T&E process, following completion of system level and integrated TA. The AOF evaluation is a structured examination of function and task groupings that is used to assess allocations in a collective manner within an integrated work environment, instead of on a single function basis, where overload issues are less likely to be revealed. Functions and tasks allocated to humans are considered in combination using scenario development. The scenarios are then evaluated to determine acceptability based on expected concurrent task performance, workload (physical and cognitive), vigilance, and situation awareness.

As part of the AOF activity, functions are decomposed to tasks, and multiple tasks may be necessary to support each function. For example, to fulfil a core protection function, a task to perform a system readiness surveillance test supports the eventual task of safety system initiation. The function tasks require a "task allocation" that uses the same criteria applied at the overall AOF level. The safety analyses also provide input to the AOF, specifying when human actions are required to backup automatic (i.e., system) actions.

In addition to allocation, within the AOF activity, task grading is completed. As described in Subsection 18.1.4.2, a graded approach to HFE is applied to the BWRX-300 project. The human actions resulting from the AOF process described above are graded based on four key risk categories:

1. Nuclear Safety
2. Personnel Safety
3. Asset Protection
4. Generation Capability

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The characteristics of the human actions are assessed against criteria in each category that results in an initial High, Medium, or Low numerical risk rating. This gives the overall minimum HFE Application Level for activities within the HFE Program for each human action. The scope and level of detail for each HFE technical element is defined for each of these HFE Application Levels.

When the risk is assessed numerically using the initial ranking criteria, the result is a minimum HFE Application Level. The HF Engineer/HFE Specialist reviews the rating and the associated HFE scope and level of effort for each technical element and takes into consideration other HFE risk factors to determine if the HFE Application Level needs to be increased for that specific technical element. The additional risk factors include:

- Complexity of the action
- Anticipated complexity and constraints of HSI
- Complexity of the system
- Frequency of the task
- Physical environment
- Cognitive environment
- Novelty of the action, system, or HSI technology
- Time sensitivity of the action

The review of an individual HFE technical element for each human action may increase an HFE Application Level but not reduce it. For example, a particular human action may have a low minimum application level based on the broad key risk criteria, but because it is a complex or novel task, it is raised to a higher application level for the TA activity.

18.2.3.3 Results

Output from the first stage of AOF is the initial AOF to human, system, or shared (both human and system). For system or shared allocations, it may be necessary to establish backup actions when redundant functions with like allocations is not possible or reasonable. For these cases, the shared and backup allocation categories are used.

The initial allocation for each system AOF is documented in a workbook which is formally managed as an internal design basis record. The workbook is communicated to all relevant stakeholders (Mechanical and I&C engineers, as well as HFE analysts) to be used as the basis for the design and TA. The initial allocation is revised and refined as necessary as the plant design and safety analyses progresses.

The results from the AOF evaluation are a final refinement of the initial allocation, including design and safety analysis modifications where necessary to support changes to allocation outcome. The final optimal AOF is used for Integrated System Validation (ISV) testing to confirm that performance, workload, and situation awareness are suitable.

The results of the AOF development and refinement activities, and specification of the final AOF are provided in a summary report.

18.2.4 Task Analysis

TA is the identification of task requirements to accomplish the functions and tasks that have been allocated in whole, or in part, to humans. These are designated in the AOF results as Human, Shared, or System with Human Backup.

TA assigns tasks to the job positions specified by the staffing process. TA determines the steps needed to accomplish human actions and documents the task details and required task support (HSI controls, indications, and alarms). The TA process also assesses the graded HFE Application Level to determine if a change to is warranted based on task characteristics.

18.2.4.1 Objectives and Scope

The TA plan establishes:

1. Methods for conduct of the TA consistent with accepted HFE practices and principles
2. Scope of the TA including actions performed at the MCR, SCR, and at other control facilities, including those required to support response to accidents and emergencies
3. Range of plant operating conditions, including start-up, normal and abnormal operations, transients, refuelling, lower power, and shutdown conditions, and emergency or accident conditions
4. HSI operations during periods of maintenance, testing, and inspection of plant SSC
5. Links among task descriptions and safety importance, function achievement, human error potential, and impact of task failure
6. Descriptions of the personnel activities required for successful completion of tasks
7. Requirements for alarms, displays, data processing, and control

18.2.4.2 Methodology

The task inputs provided by the AOF and Task Grading process form the starting point for TA. These tasks are divided into levels of effort as defined through the task grading portion of the AOF process, as described in Subsection 18.2.3. All tasks regardless of HFE Application Level receive a TA. However, the level of detail within the TA varies based on HFE Application Level. For example, those at the lowest risk level may be performed by the responsible System Engineer and reviewed by HFE for acceptability.

There are two levels of TA: Basic TA and Detailed TA. Human actions ranked at the lowest HFE Application Level undergo a Basic TA; human actions at the medium and highest HFE Application Levels undergo a Detailed TA. The Detailed TA also includes preliminary workload analysis and assessment of requirements for situational awareness. In addition to the Detailed TA, the highest HFE Application Level requires additional link Analysis and timeline Analysis to be performed to evaluate and inform the layout of HSIs to optimize task performance.

Basic TA consists of:

- Task Selection
- Task Step Sequence Narrative, including:
 - Descriptive narrative of the task
 - Cue that determines the need for the task
 - Action to be taken
 - Prerequisites for the task
 - Time available versus time required to complete

Detailed TA consists of:

- Task Selection

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- Task Step Sequence Narrative, as per Basic TA
- Task-Level Support, Job Design, Workload, and Workplace Definition including:
 - Information needed
 - Controls needed
 - Alarms needed
 - Personnel involved
 - Communication needs
 - Location and access considerations
 - Workspace needed
 - Job aids, tools, or equipment needs
 - Environmental considerations and potential hazards
 - Special clothing or personal protective equipment needs
 - Time available versus time required to complete (if needed on a step basis)

The Detailed TA is coordinated with the qualitative human error analysis, which is informed by and used to substantiate the quantitative HRA, as described in Subsection 18.2.6. The TA also identifies critical task steps where incorrect or incomplete actions might lead to undesired or unsafe consequences.

The TA activity also includes Integrated TA. Integrated TA is conducted for those operations that require interaction with multiple systems and a coordinated response that may involve multiple plant personnel. The Integrated TA takes the system-level TAs and the results from the human error analyses (described in Subsection 18.2.6) and performs higher-level whole sequence analyses, including integrated workload analysis and timeline analysis. The Integrated TAs, at a minimum, include each event sequence that contains an HFE Application Level 1 human action and each unique scenario described in the plant safety analyses.

18.2.4.3 Results

The BWRX-300 TA procedure defines set templates for capturing the TA, which are combined with the AOF results workbook, documenting the full set of analysis activities. The activities are iterative and progressive, and the workbook method allows timely and effective update of the TA and distribution of the results. The TA workbook is functionally divided into subsets of information needed to process the output activities from TA, including input to:

- Staffing and qualifications
- HSI development, including I&C Data Connection Tables (DCTs)
- Training development
- Procedure development

The results from the additional link, timeline and preliminary workload analyses in the Detailed TAs are also used to further inform the HSI design requirements, confirm or identify issues with the AOF, and provide a baseline for HSI T&E activities. The Integrated TA results provide an input into group-use and aggregate HSI designs, control facility and plant location and workspace design considerations, development of plant-level procedures, and the scenarios selected for ISV.

The results of all the TA activities, when they are complete, are summarized in a TA summary report.

18.2.5 Staffing

The required and expected number of personnel available to achieve plant functions and goals is an important consideration throughout the design process and in HFE analyses. BWRX-300 staffing assumptions and analysis results are used to frame the future plant operating organization during TA, HRA, and HSI design.

Features of the BWRX-300, such as passive safety systems, increased automation, and simplified HSIs, information systems and content, decrease the assumed initial staffing requirements relative to previous Boiling Water Reactors. For example, TA done as part of the HFE safety analysis work, described in Subsection 18.2.6, may show that the extended time for safety actions may reduce the number of personnel needed for local actions. Safety analyses and identification of human actions may show that some actions that were important in previous boiling water reactor designs have been eliminated in the plant design.

18.2.5.1 Objectives and Scope

Staffing analysis is conducted to determine the minimum staff complement. The minimum staff complement is defined as the minimum number of workers with specific qualifications who are available to the site at all times. The minimum staff must be able to operate and maintain the plant within its defined safe operating envelope and to successfully respond to all postulated events in the safety analyses, in any plant state.

For staffing assumptions where minimum staff can be varied for different operational states, the most resource-intensive events for each plant mode are analyzed. The Staffing Analysis technical element also determines the maximum staffing in collaborative areas such as control rooms, workshops, and air locks. This informs the needed space, facilities, and other support features.

18.2.5.2 Methodology

The Staffing Analysis for BWRX-300 is conducted in accordance with a plan that was developed to meet the expectations and requirements of CNSC REGDOC-2.2.5, "Minimum Staff Complement" (Reference 18.2-3). The process takes place in three major steps: Expert Panel Staffing Assessment; Staffing Analysis in TA and HSI design; and Staffing Analysis in the HFE V&V Process.

The process starts with an assumed initial staff complement taken from predecessor and similar plants, and representative OE from the operating fleet. Using this information, the initial staffing is optimized, considering the modern design features and new systems of the BWRX-300.

The optimized initial staffing level is subject to an Expert Panel Staffing Assessment that evaluates the minimum staffing to cope with selected credible events. The goal is to perform early and iterative assessments as the design progresses such that the risk of less than adequate staffing is reduced in part with each evaluation. The evaluations are performed as a desktop (talk-through) exercise led by the expert panel. The expert panel is made up of personnel from the HFE team, supported by personnel from Plant Integration Engineering and Risk and Reliability Engineering.

The next steps take place in conjunction with the TA and HSI design process. During TA, task steps are defined, and personnel assignments are made. The TA forms the basis for job design and qualifications for each role. With the TA, timeline analysis is conducted for the most resource-intensive credible events. These events are selected by the expert panel, and the expert panel

also performs a review of the timeline analysis and input TA data to evaluate whether the minimum staff complement is adequate.

After this, scenarios for a wider range of events are created and analyzed in the T&E phase of the TA and HSI design process. This allows further evaluation of the minimum staff complement against the most challenging credible events. HSI tests evaluate, using platforms and mock-ups that replicate the interface design, the timing of activities, workload, and other factors such as situation awareness that can lead to changes in the minimum staff complement and job design.

In the final analysis stage, the minimum staff complement is evaluated through the HFE V&V activities. This occurs during early validation in accordance with the multiphase validation approach (Section 18.4), culminating with the ISV. ISV demonstrates the adequacy of the final staffing levels that resulted from the analysis.

18.2.5.3 Results

The staffing analysis results are recorded in a series of reports capturing each Expert Panel Assessment, and a further report capturing the Expert Panel Review of the related staffing analysis activities. In addition to these reports, outputs and reports that are created for the TA and HSI design process and HFE V&V in accordance with those associated technical element descriptions, include results related to or impacting the staffing analysis results.

The confirmed job role and complement determination is also used as an input to training and qualification program development, where base qualifications are established, and the training program is designed (Subsection 18.3.8).

18.2.6 Treatment of Important Human Actions

Consideration and integration of HFE within the safety analyses, and consideration of the results and assumptions of the safety analyses within the other HFE activities both comprise the technical element of Treatment of Important Human Actions. The set of activities supporting this HFE technical element was developed based on requirements for HFE expertise in safety analyses per IAEA SSG-51, "Human Factors Engineering in the Design of Nuclear Power Plants" (Reference 18.2-4), CNSC REGDOC-2.5.1, "General Design Considerations: Human Factors" (Reference 18.2-5), CSA N290.12-14, "Human factors in design for nuclear power plants" (Reference 18.2-6), and international best practice.

18.2.6.1 Objectives and Scope

HFE safety analysis activities, including review of safety analyses outcomes, provides assurance that the full set of human actions that are important to safety are explicitly identified, characterized, and substantiated as achievable within the task performance requirements.

The important human actions are determined using both deterministic and probabilistic means and include identification of the human actions that might cause or contribute to the cause of postulated initiating events. Inclusion in the DSA, PSA, or Severe Accident Analysis (SAA), or other identified contribution to risk determines the risk level in the Nuclear Safety category for determining the initial HFE Application Level to apply to the HFE activities (Subsection 18.1.4.2). Comprehensive, systematic identification and substantiation of human actions claimed within the safety analyses, coupled with the risk-based graded approach described in Subsection 18.1.4.2, ensures that HSIs and tasks associated with important human actions are analyzed and designed with a full detailed and robust HFE effort.

18.2.6.2 Methodology

The safety functions that are performed to maintain the plant within or return it to a safe operating envelope are identified through various means using DSA, PSA, and SAA, as described in Chapter 15, Safety Analysis.

HFE safety analysis activities use the various safety analyses, available system design information, particularly that related to HSIs, and any available and applicable TA results (Subsection 18.2.4) as inputs.

The activities that form part of this technical element include:

1. Perform a Human Operation Hazard Evaluation
2. Review the DSA for explicit and implied human actions, e.g., human actions related to maintenance that ensure safety-class SSC availability, and ensure claimed human actions are achievable through qualitative human error analysis
3. Review the PSA and HRA for:
 - a. Ensuring event sequences introduce a Human Performance Limiting Value
 - b. Perform qualitative human error analysis to substantiate or refine the human error probabilities used for all human actions claimed in the HRA
 - c. Provide HFE qualitative basis and substantiation for any dependency assumptions and analysis
 - d. Provide HFE qualitative basis and substantiation for any timeline assumptions and analysis
4. Review the SAA, including Level 2 and Level 3 PSA, to identify explicit and implied human actions, determine task achievability in required timescales, and capture input for procedure development (Subsection 18.3.7) and emergency planning and response (as described in Chapter 19)
5. Compile a database of all important human actions claimed in the safety analyses, including the source of the claim, their key characteristics, related assumptions and any associated HSIs

For each of the above activities, the task performance criteria are defined to enable the HFE analysis to determine the acceptable achievability of each identified human action. Task performance requirements depend on the plant conditions the task is performed in (normal versus abnormal versus emergency or accident conditions), and for events, are defined by the related safety analysis the HFE analysis is underpinning. For example, for precursor human actions, performance requirements are usually "performed correctly" or "performed in accordance with maintenance schedule timelines". For post-initiating event human actions, the requirements are defined by the event conditions. Tasks must be achievable, completed successfully and, where dictated by the related analysis, must be completed within the required time based on the event timeline.

18.2.6.3 Results

The outputs from the activities that form this technical element includes:

1. A Human Operation Hazard Evaluation report that documents the methods used and the evaluation results
2. Design records capturing results of safety analyses reviews

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3. Design records capturing HFE qualitative human error analysis and other substantiation of claimed human actions in the various safety analyses
4. A Human Action Claims database containing all important human actions and critical information for each, allowing for adequate and robust consideration in the design
5. A summary report, which summarizes the methods, results, and issues from each of the above outputs

The results are communicated directly to the appropriate design, safety analysis and other HFE team members performing related technical elements, to ensure timely consideration of the full set of human actions important to safety in their respective activities. The outputs from each activity and the Human Action Claims database capture the source of the identified human actions within the safety analysis to allow full traceability from the origin through HFE safety analysis and forward to any design requirements or safety analysis modifications.

In addition to specific HFE safety analysis activities described in this section, the full complement of safety analysis outcomes, including the PSA and HRA, informs and acts as input to the other HFE technical elements, as applicable.

18.2.7 References

- 18.2-1 IAEA-TECDOC-668, "The role of automation and humans in nuclear power plants," International Atomic Energy Association.
- 18.2-2 USNRC NUREG/CR-2623, "The Allocation of Functions in Man-Machine Systems: A Perspective and Literature Review," U.S. Nuclear Regulatory Commission.
- 18.2-3 CNSC Regulatory Document REGDOC-2.2.5, "Minimum Staff Complement."
- 18.2-4 IAEA Safety Standards Series No. SSG-51, "Human Factors Engineering in the Design of Nuclear Power Plants," International Atomic Energy Association.
- 18.2-5 CNSC Regulatory Document REGDOC-2.5.1, "General Design Considerations: Human Factors."
- 18.2-6 CSA N290.12, "Human factors in design for nuclear power plants," CSA Group.

18.3 Design of the Human-System Interface

This section describes the process by which HSI designs are established and evaluated. The HSI design process for BWRX-300 is governed by a process methodology report that outlines the required design inputs, design procedure to be followed, design outputs, and the process for conducting HFE T&E during design development. The HSI design process sits within and is fully integrated into the overall plant design process specified through standard engineering design process and procedure documents, as well as the relevant project-specific design process plans. General design principles and processes are described in Chapter 3, particularly Subsection 3.1.7.

18.3.1 Design Goals and Design Bases

The primary goal of the HSI design process is to facilitate safe, efficient, and reliable user task performance during plant normal operational states, abnormal events, and accident conditions. To achieve this goal, HSIs throughout the plant are designed and implemented consistent with HFE core principles and user-centred design practices. The following specific design bases are adopted for the plant:

1. HSI design promotes efficient and reliable operation through application of automated operation capabilities.
2. HSI design uses only proven technology.
3. The workstation and HSI layouts reflect I&C separation restrictions.
4. HSI design is highly reliable and provides functional redundancy such that sufficient displays and controls are available in the MCR and SCR and remote locations to conduct an orderly reactor shutdown and to cooldown the reactor to safe shutdown conditions, even during design basis equipment failures.
5. The principal functions of the Safety Parameter Display System (SPDS) as required by CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 18.3-1) are integrated into the HSI design.

NOTE: Historically, the SPDS has provided an overview display of important plant parameters during transients and accidents on a separate panel with separate safety categorization due to the safety analysis outcomes for full-scale plants. The SPDS terminology is used in this PSAR for consistency with regulatory documents. However, the BWRX-300 SPDS functionality does not require a different safety classification and is an integral part of, and included with the SC3 displays (see Subsection 18.3.5). There is no separate panel or system.

6. Accepted HFE principles and methods are used for ensuring HFE is integrated into the design, in alignment with international best practice and meeting the requirements of CNSC REGDOC-2.5.1, (Reference 18.3-2).
7. HFE design requirements are based on international standards and applicable CNSC regulatory requirements, as outlined in the HFEPP.
8. The design basis for accident and emergency control and monitoring facilities meets international standards as well as CNSC REGDOC-2.5.2 (Reference 18.3-1), and CNSC REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response" (Reference 18.3-3).

Task performance criteria developed through the HFE Program analysis activities, in conjunction with those described in Chapter 3, Section 3.1 and Chapter 15, Subsections 15.3.2 and 15.3.3

are used to govern and direct HSI design specifications. These detailed task performance criteria, along with requirements specified in HFE standards and codes, encompass the set of necessary and sufficient design requirements that maintains the implemented plant SSC and HSIs in compliance with accepted HFE principles.

18.3.2 Human-System Interface: Design Inputs

The inputs to the HSI design are derived from several sources. These include specific information relating to the performance of tasks, as well as design requirements and guidance specific to the plant design and a defined full set of plant user characteristics, but not related to task performance. In addition to these documented input sources, the integrated set of HFE Program activities includes HF Engineer/HFE Specialist support to designers for instances where the correct application of the set requirements is not clear or where design conflicts exist, and suitable alternative design solutions are required. Finally, HSI design updates are made based on results from HFE T&E and HFE V&V activities, as described in Subsection 18.3.4 and Section 18.4, respectively.

18.3.2.1 Task-Related Input

A primary input to HSI development is the user task information and control needs established during TA (see Subsection 18.2.4). TA provides the following information that forms the HSI Task Support Inventory:

- Information determining the need to initiate a task
- Control needs to accomplish the task steps
- Information feedback to confirm that task step control actions have been accomplished
- Information for determining that task steps are accomplishing their intended objectives
- Information for determining when tasks may be terminated
- System and component alarms
- Information on task performance requirements for group-use and aggregate HSIs
- Information regarding where manual tasks need to be performed (remote or local)

18.3.2.2 Design Requirements and Guidance Input

The second main input to HSI design is the full set of HFE design requirements derived from international codes, standards, regulations, and best-practice guidance, that are applicable to the defined user group characteristics and the types of HSIs used throughout the plant. These requirements are managed through the formal requirements management process for the plant design that allows traceability from source to implementation. The HFE Design Requirements Document provides an extract from the requirements management database, providing a single repository of these common requirements (i.e., applicable to all SSC).

For the design of HSIs, the requirements in the HFE Design Requirements Document do not provide the entire basis for developing display interfaces. For example, within “requirement-compliant” screen designs, there are any number of acceptable ways to layout and create screen artefacts, for example, variations in colour, size, font, and placement. Further inputs are required that ensure consistent and intuitive HSI designs across the plant. The requirements for this part of the HSI design process are defined in the project HSI Style Guide.

For digital software-driven HSIs, the style conventions are further developed into an HSI Element Library, which contains HSI display templates and HSI elements (e.g., symbols, numerical displays, graphs) that the display designer uses to assemble the display content. The HSI

Element Library contains both HSI elements for primary interfaces (those that represent direct interface to the system and plant HSI) as well as secondary interfaces (such as navigation, which do not directly relate to system equipment).

The HSI Style Guide is also used to maintain consistency for hardware-based controls and indicators, where suitable components are selected and, along with HSI panel templates, their specifications are included in the HSI Element Library.

18.3.2.3 Human Factors Engineering Concept of Operations

The HSI concept design scope includes development of the HFE Concept of Operations (COO) which defines the physical and cognitive characteristics of the standardized plant full user population. The HFE COO provides user population anthropometrics for the full range of 5th percentile female to 95th percentile male users, for the worldwide population specified in ISO 7250:3. Other user population characteristics are provided, including population stereotypes (i.e., expectations of interface functionality of the whole user population based on country or nuclear industry norms).

18.3.2.4 Human Factors Engineering Support

The final input is provided by the HFE team members on an as-required basis. This input is specific to each design challenge or designer technical query. The integration of the HFE team with the other disciplines provides the mechanism for designers to request HF Engineer/HFE Specialist support for instances where is not clear, for the HSI design aspect they are implementing, how the pre-specified requirements are correctly or effectively applied. Designers also request support when they identify conflicting design criteria that limit or prevent implementing the HFE design requirements as specified. In such cases, they need HF Engineer/HFE Specialist advice on the most suitable alternative design solutions.

18.3.2.5 Results from Testing, Evaluation, Verification and Validation

Throughout the design development, HFE T&E is performed (Subsection 18.3.4). Later in detailed design, early HFE V&V activities start. The results from these HFE T&E and V&V activities may be the identification of an HFE issue with the design or an HED. Recommended resolutions requiring HSI design improvement form the inputs to further design development.

18.3.3 Human-System Interface: Detailed Design and Integration

In accordance with the HFEPP and the HSI design methodology, HSI designs are created through the interaction and coordination of the HFE team and discipline engineers. Degree and type of interaction is based on the risk-based HFE Application Level as described in Subsection 18.1.4.2.

18.3.3.1 Objectives and Scope

The objectives of the HSI design process are to:

1. Translate codes and standards, as well as functional and task requirements, into HSI characteristics, displays, software, and hardware that enhance safety and reduce the risk of human error to as low as reasonably achievable through design
2. Support the principal objectives of the HSI design to provide the indications, controls, and status displays necessary for tasks allocated to each user, for all the required plant functions during all plant conditions, and to provide the user with accurate, complete, and timely information regarding the functional status of plant equipment and systems
3. Ensure design trade-offs are resolved during the HSI design activities through the systematic application of HFE principles and criteria, and with HF Engineer/HFE Specialist support

4. Maximize the plant capacity factor in the HSI design by:
 - a. Facilitating planned operations, maintenance, inspection, and testing
 - b. Minimizing the occurrence of any undesired power reduction or plant trip caused by erroneous decision-making and actions
 - c. Permitting plant commissioning to take place effectively and allowing timely modifications and maintenance of the HSIs

The scope of the HSI design process is to specify requirements for HSIs throughout the plant. For hardware-based HSIs, the HFE team supports other discipline designers in designing and selecting HSIs. In the case of software-based HSIs, developing the optimized displays are HFE team responsibility. As with all other HFE Program activities, the scope and methods used for HSI design are graded based on HFE Application Level (Subsection 18.1.4.2). The HFEPP details the level of effort and scope for HSI design per HFE Application Level.

18.3.3.2 Methodology

The HSI design products are created through the interaction and coordination of the HFE team and discipline engineers. Degree and type of interaction is based on the HFE Application Level.

The HFE team provides design and task support requirements (Subsection 18.3.2).

Depending on the HFE Application Level and the nature of the HSI, the HFE team:

1. Provides design requirement-compliant, application-specific wireframes, and templates for HSI displays, panel layouts, and HSI elements, housed within the HSI Element Library
2. Works with the System Engineer to implement the HFE requirements in the requirements management database (as compiled in the HFE Design Requirements Document) that apply to the discipline and system, allowing them to design or specify and select compliant SSC or HSIs and to develop compliant system and equipment layouts
3. Depending on HFE Application Level, reviews, tests, and verifies all HSI design work to ensure acceptable requirements are implemented and compliance is documented or performs proportionate design work audits using the Design and Task Support Evaluation Checklists to ensure acceptable requirements are implemented and compliance is documented

Depending on the HSI type and the HFE Application Level, the same process is followed but the primary and secondary designers/engineers may vary.

The HFE Program HSI Design technical element includes roll-out of the HFE requirements and support to their implementation for other disciplines. For hardware HSIs, including plant layout and physical environment, direct physical SSC interfaces and HSIs that form part of COTS or bespoke design systems and equipment, the responsible discipline engineer includes the applicable task-based and HFE Design Requirements Document requirements as part of their system and component level design requirements. The applicable HFE requirements are included in “lower” level system and component requirements specifications, including procurement specifications, ensuring consistency of application throughout the plant design. This is managed using the design requirements management tool, the standard content for system design specifications, and the integrated HFE design support activities and issues management process outlined in HFEPP.

The HSI design process for software-based HSI display designs, which are the responsibility of the HFE team, is to create each system User Interface Specification (UIS), which contains a DCT.

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The DCT lists the I/O points associated with a HSI element on a display or panel and provides a mapping of instrumented parameters and controlled components to individual HSIs as follows:

1. Assemble inputs and requirements following the UIS input gathering procedure
2. Complete the system UIS and the DCT. The UIS provides detailed renderings of the HSI display and panel layouts, including the components and parameters to be included on the HSI displays and panels. A UIS is created for each system, and a UIS is also created for the plant-level displays as a part of control room system design.
3. Integrate the UIS and DCT for each system

The complete UIS standard deliverable provides the data, templates, and formats necessary for the development of the software and I/O points to drive the user display interfaces. The UIS standard deliverable consists of:

- HSI Task Support Inventory
- HSI Task Support Inventory – Key Parameters
- DCT
- HSI display screenshot

The HSI display screenshot is a record of the actual HSI display, contained in a software file, delivered with the UIS to the I&C team for data connection with the logic modules.

The HSI design process also includes development of an Alarm Management Design Guide during the concept design stage. The guide provides detailed alarm system and alarm presentation guidance, including outlining the principles for alarm identification, prioritization, filtering, and suppression, in line with human cognitive capacity and required response.

For HSI design, applicable requirements are applied to the developing design, and compliance is documented and maintained by the relevant engineering discipline. Where exception to a requirement is needed, HFE provides design support to the other disciplines to develop and document an HFE-approved justification for an HFE design requirement exception.

The HFE design requirements apply equally to the HSIs of COTS equipment and components. Ability of COTS equipment and component HSIs to meet the HFE design requirements is one of the standard selection criteria. However, it is recognized that not all COTS items require the same level of rigour; standard items that do not include HFE as part of their specification require HFE evaluation for non-compliance with the HFE design requirements. When evaluating COTS products that do not comply with HFE design requirements, special considerations are applied by the HFE team to determine and document acceptability of the discrepancy; these include:

1. Trade-off of benefits of using a proven, standard solution compared to the benefits of a custom solution that more closely meets the HFE design requirements
2. Analysis of COTS vendor HFE design basis and documentation in relation to HFE codes, standards, and relevant good practice
3. Evaluation of COTS HSI design applicability to the defined user population, conventions, and stereotypes
4. Degree of design and task support integration and consistency between the COTS product and the rest of the HSIs
5. Identification of usability or human performance concerns with the proposed application of the COTS product

To accommodate these evaluations, where COTS products are considered, this is determined as early as possible in the design lifecycle, following completion of the HFE COO and other HSI design input documents as soon as the relevant System Engineer specifies that such products will be used.

The individual system and integrated plant-level UIs, hardware-based HSI designs and COTS HSIs, and the plant-level HFE requirements (e.g., those related to workspace layout and working environment), are also integrated into relevant control facility HFE design specifications. The design process for the MCR and SCR are described in Subsections 18.3.5 and 18.3.6; the design for other control facilities, including those for responding to accidents and emergencies follows a similar process, with graded level of effort based on complexity of the HSIs. The facilities for supporting emergency and accident response are in early concept design as described in Chapter 7, Section 7.7 and Chapter 19, Section 19.2.

18.3.3.3 Results

The process and the rationale for the HSI design are documented and managed under GEH Quality Assurance and BWRX-300 specific design plans, as described in Chapter 17. The HSI design process uses the following templates within the TA and HSI Design workbook to create the UIS:

- HSI Task Support Inventory
- HSI Task Support Inventory Key Parameters
- DCT

The specific controls, indications, displays, panels, and HSI elements designed to support the user tasks are documented in the HSI Task Support Inventory table. Data in the table documents that the designer confirmed that the HSI characteristics are appropriate for the specific use application. The table looks at display and panel locations to confirm that information that needs comparison is located on the same display or panel and that information used in related task actions are located on the same display or panel or are available for concurrent display on adjacent video display units or panels.

In addition to the templates, a screenshot of the resulting HSI display is included in the workbook. The HSI design for displays performed by the HFE team also results in the HSI display software file.

To support the standard plant alarm system logic implementation, a detailed alarm presentation specification is produced in line with the Alarm Management Design Guide. The specification includes rationalization and prioritization evaluation results, providing alarm filtering and presentation requirements as input to the I&C team alarm system design activities.

Other outputs include the design documentation (system design specifications, system and component requirements documents, purchase specifications, and drawings and models), as appropriate to the discipline and HSI type being designed. The results of any non-compliance evaluations and design trade-off decisions are recorded in design records and (where appropriate) the HFEITS (Subsection 18.1.5).

18.3.4 Human-System Interface: Tests and Evaluations

T&E is an integral part of the HFE design process, with the results of evaluation T&E efforts leading to early and effective modification to requirements and design improvements.

18.3.4.1 Objectives and Scope

The purpose of HFE T&E is to find and address issues early, rather than waiting for HFE V&V activities near the end of the project (Section 18.4). It is the means to test the feasibility of concepts and early prototypes and to facilitate reaching design decisions. Another difference from the V&V is that design and HFE engineers involved during the design stages are not excluded from being test participants.

The scope of the HFE T&E includes:

- Defining the HSI prototypes and simulation testbeds
- Defining the HFE T&E team and participants
- Establishing HFE T&E methods
- Performing HSI selection and prioritization
- Performing HSI evaluation and user-based testing
- Collecting and analyzing data
- Documenting results, and communicating them to the relevant stakeholders

HFE T&E scope ranges in complexity from simple user questionnaire responses and comments to empirical, performance-based techniques to assess how the user responds to the design under increasingly realistic conditions. The level and complexity of HFE T&E is based on design phase, task complexity, integration of the design feature to be assessed, and design and project risk (new HSI, new systems, high HFE risk grading).

18.3.4.2 Methodology

To maximize the effectiveness of HFE T&E, HSIs are selected based on prioritization criteria. Primarily selection and prioritization of the HSIs are based on the HFE Application Level (Subsection 18.1.4.2). Where HSIs support more than one task, which means they may have more than one associated HFE Application Level, the worst-case (highest risk) level is used.

Beyond this grading, additional HSI are selected for HFE T&E inclusion based on consideration of any HSI design assumptions that require T&E. Assumptions made during the design phase are identified and refined so that they are specific enough for testing. The design assumptions are weighted to determine test priority (similar to the grading of human actions based on risk). The T&E focuses on the assumptions that have the highest impact if incorrect and the shortest time to learning the HSI.

Some examples of candidate HSI design assumptions include the following:

- Colours and status coding (short time to learning; medium impact if false)
- Hardware HSIs basis ergonomic check (short time to learning; high impact if false)
- Safety HSIs (long time to learning; high impact if false)
- HSIs related to the highest risk-level graded tasks (long time to learning high impact if false)
- New system functionality (long time to learning; high impact if false)

The T&E program is comprised of multiple assessment methods, with the most dominant being performance-based testing. Performance-based testing consists of observing users, given a goal to achieve, interacting with a suitable representation of the HSI design. Members of the test team

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observe the user's actions without intervening, recording what transpires. Post-test analysis focuses on any difficulties encountered by the user, both qualitatively and quantitatively obtained, depending on test stage and testbed fidelity. The results are used to highlight differences between the design team assumptions in developing the HSI and actual user behaviour when using it, indicating potential human error traps in the design.

During design development, the performance-based testing may be formative in nature. This type of testing allows for quick low-fidelity prototyping and problem resolution. During formative testing the test administrator and user both participate in the test. The administrator may prompt for information on what and why a user is performing actions to understand the thought process as well as to understand better the prototype or conceptual design limitations.

At appropriate points in the detailed design, testing is done with a summative approach. During summative testing the test administrator does not participate to limit test bias. Several users are tested separately to allow assessment of error variances and statistical comparison of test results.

Each performance-based test cycle begins with the development of a test plan that outlines the purpose, equipment needed, design features being tested, test and data collection methods, performance measures and acceptance criteria, as well as any testing material where appropriate. Design features selected for user testing, the test fidelity, user representatives chosen, testbed used, and performance measure(s) and acceptance criteria all depend on the maturity of the design at the stage of the testing.

Considerations for test design include:

1. Availability of plant modeling software and integrated HSI design status
2. Availability and fidelity of a mock-up
3. Availability of control area and equipment 3D modeling
4. Availability of procedures, procedure types, and training material
5. Availability of a sufficiently diverse participant population pool that is representative of the user population to the level required for the testing stage
6. Develop an observation and evaluation plan. The observation plan includes written test plans, scripts for observers and evaluators, standardized training for participants, and the same observers or proctors for all runs of an evaluation (whenever possible).

Once the test plan is complete and the testbed selected or designed, the T&E team develop an observation and evaluation plan. The observation plan includes written test plans, scripts for observers and evaluators, standardized training for participants, and the same observers or proctors for all runs of an evaluation whenever possible.

The test is then conducted in accordance with the plans.

The HFE T&E program uses a variety of methods and tools for analyses, reviews, and evaluations of the HFE T&E performed throughout the design process.

Data collection methods are selected appropriate to the type of test or evaluation being conducted, as detailed in the T&E plan. Techniques appropriate for the evaluation of HSI include:

- Participant questionnaires and interviews
- Direct observation of user behaviours (e.g., task time, task errors, HSI interaction or navigation errors)
- Simulation instructor console data

The following criteria are used to select the data collection methods:

- Safety and risk significance
- Type of design (depending on the type of design, there are some methods that may not apply)
- Type of technology
- Relative time to perform the test or evaluation
- Relative complexity
- Relative cost and value

In addition to performance-based user testing, the HFE T&E team conducts formal trade-off evaluations to determine the relative benefits of potential design alternatives. Trade-off evaluations are conducted by a multi-discipline group of relevant stakeholders – HFE, other discipline engineering experts, HSI designers, and samples of end users. The trade-off evaluation is conducted using a standard trade-off tool, ranking the design alternatives against weighted key HFE criteria. The output of the trade-off tool is used as the basis to make the HSI design alternative trade-off decision. If there are several closely ranking alternatives, further HFE review or analysis is undertaken to determine. HFE issues resulting from this evaluation are recorded and tracked using the HFEITS. Those that are not resolved at the time include the necessary information to address them in future project stages.

18.3.4.3 Results

The results from any HFE T&E are documented in design records and on associated test forms design to support the T&E process. Any issues and recommendations resulting from the HFE T&E activities related to design improvement are communicated directly to the applicable HSI designer, used as input to further design development (as per Subsection 18.3.2.5). When the T&E for each design stage is complete, a T&E summary report is prepared that summarizes all T&E activities and their results).

18.3.5 Human-System Interface: Design of the Main Control Room

The current concept of the MCR is described in Chapter 7, Section 7.5. The MCR design features are based upon proven technologies and are demonstrated, through broad scope control room dynamic simulation during HFE T&E and V&V, to satisfy the HSI design goals and design bases. Validation of the implemented MCR design includes evaluation of the design features, the user job roles, staff complement, and procedures, performed as part of the HFE V&V process as defined by the test specification and performance measures specified for each validation activity (Section 18.4).

The HSI design implementation activities include support to the development of dynamic models for evaluating the overall plant response as well as individual control systems, including operator actions. These dynamic models are used to:

- Analyze both steady state and transient behaviours
- Confirm the design of the advanced alarm system concepts
- Confirm the adequacy of control schemes
- Confirm the allocation of control to a system or an operator
- Develop and validate system and plant-level operating procedures

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Using part-task simulation, system models are developed, and linked to the HFE-designed the HSI displays. The part-task simulator is used in preliminary plant design and includes design features specific to BWRX-300.

As the design progresses, the part-task simulator proceeds through a series of iterative evaluations resulting in the development of a full-scope simulator. As soon as available, simulators are the preferred testbed for T&E, allowing for progression from static to dynamic testing.

Safety margins used in the DSA account for uncertainty and provide an added margin to ensure that the various limits or criteria important to safety are not challenged. Suitable margin is also added to the HFE analysis of human actions (during TA and human error analysis, per Subsections 18.2.4 and 18.2.6) by ensuring suitable conservatism is included in things like the timeline analysis or the generation of human error probabilities.

Design goals and design bases for the design of HSIs in the MCR and SCR and in other applicable facilities are established in Subsection 18.3.1, based on the HFEPP.

18.3.5.1 Objectives and Scope

The primary goal of HSI design and HFE input to design of the MCR is to facilitate safe, efficient, and reliable user performance during all phases of normal plant operation, abnormal events, and accident conditions. To achieve this goal, information displays, controls and other interface devices in the control rooms and other plant areas are designed and implemented in a manner consistent with best HFE practices. Further, the following specific design bases are adopted:

1. HSI design promotes efficient and reliable operation through application of automated operation capabilities.
2. HSI design uses only proven technology.
3. Safety-related systems monitoring, and control capability is provided in full compliance with regulations regarding divisional separation, and independence.
4. HSI design is highly reliable and provides functional redundancy such that sufficient displays and controls are available in the MCR, or as a backup, in the SCR and remote locations to conduct a reactor shutdown and to ensure the reactor achieves and maintains safe shutdown conditions, even during postulated accidents.
5. The MCR remains habitable and protected for all events and accidents during which it is required to be used (habitability of the MCR is described in Chapter 6).
6. The principal functions of the SPDS as required by CNSC REGDOC-2.5.2 (Reference 18.3-1) are integrated into the HSI design.
7. Accepted HFE principles and methods are used for integrating HFE into the MCR design. in accordance with international best practice and meeting the requirements of CNSC REGDOC-2.5.1 (Reference 18.3-2).
8. HFE design requirements are based on international standards and applicable CNSC regulatory requirements, as outlined in the HFEPP.
9. The principal functions of the SPDS as required by CNSC REGDOC-2.5.2 (Reference 18.3-1) are integrated into the HSI design.
10. The design basis for accident and emergency control and monitoring facilities meets international standards as well as CNSC REGDOC-2.5.2 (Reference 18.3-1), and CNSC REGDOC-2.10.1 (Reference 18.3-3).

The evaluation of the integrated MCR design provides confirmation that the MCR HSIs and other design features are compliant with the HFE Design Requirements Document and any analysis-based design requirements. Refer to Chapter 7 for description of I&C system content and to Chapter 13 for the Conduct of Operations.

18.3.5.2 Methodology

The MCR concept outlined in Chapter 7 contains a group of workspaces and individual HSIs, which form the foundation for the detailed HSI design. The development of the MCR workspaces and HSI design features is accomplished through:

- Consideration of existing control room OE
- Review of trends in control room designs and existing control room data presentation methods
- Evaluation of modern HSI technologies, including alarm system design, particularly alarm reduction and presentation methods
- Application of relevant compiled requirements from the HFE Design Requirements Document
- Design or specification and selection of individual HSIs
- Specification of the integrated HFE design requirements for the MCR as a whole
- Testing of a dynamic MCR prototype (full-scope simulator)

Detailed task performance criteria are specified as part of the TA (Subsection 18.2.4) and qualitative human error analysis (Subsection 18.2.6). These criteria are used to govern and direct all plant control room designs. These detailed task performance criteria, along with requirements specified in HFE standards and codes, encompass the set of necessary and sufficient design requirements that maintain the implemented plant control room designs in compliance with accepted HFE principles. This includes ensuring that any HSIs required to provide manual backup control to safety systems are identified and provided in a location and using technology (e.g., hardware-based high-reliability controls and displays) that are available in the postulated task conditions.

The full-scope simulator is evaluated under normal and abnormal reactor operating conditions by participants suitably representative of the defined user population, as described in the HFE T&E process in Subsection 18.3.4. Following the completion of the HFE T&E and V&V on the full-scope simulator, the MCR workspace, and HSI design features are finalized.

18.3.5.3 Results

The results from design MCR design activities are the same as those for the overarching HSI design process as described in Subsection 18.3.3.3. This includes the outputs from the related HFE T&E and V&V activities, and identification and tracking of design-related HFE issues and HEDs.

18.3.6 Human-System Interface: Design of the Secondary Control Room

The SCR provides means to safely shut down the plant from outside the MCR in a location that is protected and not impacted by the same scenarios that makes evacuation of the MCR necessary. The SCR provides the HSIs for the plant systems needed to bring the plant to hot shutdown, with the subsequent capability to attain safe shutdown, if the MCR becomes uninhabitable. The SCR is in early concept at the time of issuing this PSAR. The current concept

is described in Chapter 7, Section 7.6. Habitability of the SCR and protection of the route between the MCR and SCR is described in Chapter 6, Section 6.4.

18.3.6.1 Objectives and Scope

The SCR provides means to safely shut down the plant from outside the MCR in a location that is protected and not impacted by the same scenarios that makes evacuation of the MCR necessary. The SCR provides the HSIs for the plant systems needed to bring the plant to hot shutdown, with the subsequent capability to attain safe shutdown, if the MCR becomes uninhabitable. The SCR is in early concept at the time of issuing this PSAR. The current concept is described in Chapter 7, Section 7.6. Habitability of the SCR and protection of the route between the MCR and SCR is described in Chapter 6.

18.3.6.2 Methodology

The methodology for design of the SCR is the same as that for the MCR (Subsection 18.3.5.2) and more generally for HSI design (Subsection 18.3.3.2). As with all HFE Program activities, a proportionate, graded approach is taken to the design of the SCR. Due to the nature and purpose of the SCR and the plant conditions expected when it needs to be used, the human actions are by default important to safety and if incorrectly performed, lead to significant consequences and as such receive the highest HFE Application Level.

18.3.6.3 Results

The results are captured in the same means as for the MCR design (Subsection 18.3.5.3).

18.3.7 Procedure Development

Procedure development for the BWRX-300 is performed by the HFE team in accordance with a plan that details the inputs, method, and scope of procedure development activities.

18.3.7.1 Objectives and Scope

The objective of procedure development is to apply HFE principles and guidance to the development of procedures such that they are technically accurate, comprehensive, explicit, easy to use, and validated. The process for procedure development follows applicable requirements from IAEA-TECDOC-1058, "Good Practices with Respect to the Development and Use of Nuclear Power Plant Procedures" (Reference 18.3-4).

The plant procedures are developed as an integral part of the HSI design development. The procedures are developed either as new or modified from predecessor plants. Existing procedures are modified to reflect the characteristics and functions of the plant task types, modes and conditions, and any applicable OE related to procedure design.

The HFE Program includes activities to verify that all functions and tasks assigned to the plant personnel are included in the procedures. The HFE V&V activities include validation of the procedures using the mock-ups, part-task, and full-scope simulators to confirm their usability and accuracy. Procedure development is iterative and progressive, in line with the developing design and results from progressive HFE analyses.

The scope of procedure development addresses all tasks required to meet functional goals in operations, maintenance, inspection, testing, and accident management of the plant. The scope includes development of:

- Procedures Writer's Guides
- Plant and System Operations Procedures (start-up, normal (at power operations), and shutdown)

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- Maintenance, Inspection, Testing and Surveillance Procedures (including Refuelling and Outage Planning Procedures)
- Alarm Response Procedures
- Abnormal Operating Procedures
- Emergency Procedure Guidelines for Emergency Operating Procedures (EOP) development
- (EOPs)
- Severe Accident Management Guidelines
- Emergency Mitigating Equipment Guidelines

The procedure development process does not include specific requirements for the development or procurement and implementation of a Computer-Based Procedures (CBPs) platform or tool. However, the Procedure Writer's Guide specifies basic HFE requirements for usability of the CBP interface (the same as any other user interface) and the development of the content and format for any CBP inputs follows the same requirements for each procedure type given in the above list.

18.3.7.2 Methodology

The procedures development methodology establishes the process for developing technical procedures that are complete, accurate, consistent, and easy to understand and follow.

For each procedure type, a Procedure Writer's Guide is established. The Procedure Writer's Guide establishes objective criteria so that the procedures developed with it are consistent in organization, style, and content. The Procedure Writer's Guide provides instructions for procedure content and format, writing of steps, and specifying lists of terms used.

Procedures are then written after the associated TAs are finished. TA is an iterative process due to the amount of information that is created at any point in the design process. TA is conducted in a prioritized manner and done on a per task, per system basis. System Design Descriptions are used along with the TA output to give a complete understanding of the system and its operation. These inputs provide understanding of each system, its associated tasks, and its interrelationship to other systems, which allows the development of system level and then integrated operations procedures.

As the TA progresses beyond operations, when the System Engineers have defined test, inspection, and maintenance requirements and the iterative safety analyses have sufficiently matured, procedures are developed for other than normal operations (e.g., test, maintenance, surveillance, alarm response, outages and any other conditions that are not included in the scope of normal plant operations). Procedures that support important human actions for the higher HFE Application Levels are developed by the HFE team based on the detailed TA and qualitative human error analysis. Procedures to support human actions for the lowest HFE Application Level are developed by the responsible System Engineer or the vendor, based on the Basic TA (Subsection 18.2.4).

Initial procedures are tested and evaluated for their usability and efficacy using physical mock-ups, simulation, and plant 3D models early in the design, through the HFE T&E activities. The HSI design and procedures are evaluated together to make a more cohesive model of the future operational plant. This provides contextual feedback for both the HSI design and the procedure, allowing optimization of the design of both simultaneously.

The developed procedures are verified and validated as part of the HFE V&V program (Section 18.4), culminating in ISV. Procedures that support important human actions for the higher HFE Application Levels are validated by individuals that are independent of the design. Individuals that participate in validation of these levels of procedures include representatives of the end users. Final procedure validation is done on the installed physical plant hardware as part of the HFE design implementation once the plant has been built. Procedures to support human actions for the lowest HFE Application Level are validated by the designer.

The final procedure validations are done as specified by the plant pre-operational testing and start-up testing programs. Following ISV, the procedures are used as the basis for pre-operational testing, start-up testing, and operation of the plant. Once a procedure is validated and declared complete, procedure maintenance and control of updates is governed by the engineering change management process ensuring that changes to individual procedures are reflected throughout the full suite of related procedures, and that changes are confirmed as accurate and supported by the design and TA. Validated procedures are provided to operations for training and use.

18.3.7.3 Results

The results of the procedure development process are the final set of procedures and any procedure support documentation developed using the procedure development methodology.

The output documents include the following:

- Procedure Writer's Guides
- Plant and system operations procedures
- Maintenance, Inspection, Testing and Surveillance Procedures (including Refuelling and Outage Planning Procedures)
- Alarm Response Procedures
- Abnormal Operating Procedures
- EPGs
- EOPs
- Severe Accident Management Guidelines
- Emergency Mitigating Equipment Guidelines

18.3.8 Training and Qualification Program Development

Training and qualification program development is coordinated with the other elements of the HFE Program, for example by using HFE TAs to conduct a systematic analysis of job and task requirements. The program of analysis and training material development is conducted by the Training team, in accordance with a plan that provides the methods and framework for ensuring the program meets its requirements and technical basis. The HFE team provide inputs to the training analysis activities and provide support to the Training team in conducting the training program development activities.

18.3.8.1 Objectives and Scope

The aim of the BWRX-300 training and qualification program development is to systematically incorporate information from the other HFE design tasks to support development of accurate and applicable training content and implementation of effective personnel training. The training program development process is intended to produce a program that:

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- Identifies all performance requirements of a job or duty area relating to licenced activities
- Defines and documents the training based on a TA that provides the information to establish the knowledge, skills, and abilities to perform each task including identifying the safety-related attributes if any
- Ensures that the training is designed, developed, and implemented to meet the qualification requirements
- Ensures instructors meet and maintain documented qualification requirements, particularly in areas of subject matter expertise and instructional skills
- Ensures that formal evaluation methods are used to confirm and document workers qualifications
- Implements a change management control system that systematically identifies changes to tasks and task lists for revisions of training
- Ensures continuing training is provided as deemed necessary through training needs analysis
- Evaluates training regularly and incorporate the results of the evaluation into a training improvement process
- Ensures that workers training and qualifications records are established and maintained
- Ensures that workers have a level of training related to nuclear safety corresponding to their position including but not limited to radiation safety, fire safety, onsite emergency training, and conventional health and safety

The training and qualification program development includes the following stages:

- Analysis
- Design
- Development
- Implementation
- Evaluation

The overall scope of the resulting training and qualification program includes the following:

1. All categories of personnel conducting tasks within the plant, including the full range of job roles whose actions may affect plant safety
2. The full range of plant conditions (normal operational, outage, abnormal, accident, and emergency)
3. All activities conducted throughout the plant (e.g., operations, radwaste processing, outage refuelling, online and offline maintenance, testing, and inspection)
4. The full range of plant functions and systems
5. The full range of relevant HSIs

The scope of the training and qualification program development plan does not include the specific requirements for certification of plant personnel specified in CNSC REGDOC-2.2.3, "Personnel Certification, Volume III: Certification of Reactor Facility Workers" (Reference 18.3-5). The requirements for certification are incorporated as part of the overall program; however, they

do not need to be derived through the defined development process, since they are already specifically defined. The final training and qualification program specific to the operational goals of the plant and developed from the HFE and design inputs, as described in this section, are augmented by the general and specific training and certification requirements specified in CNSC REGDOC-2.2.3, Volume III (Reference 18.3-5).

18.3.8.2 Methodology

The training and qualification program development follows the fundamentals of the systematic approach to training method. The development process complies with the requirements of CNSC REGDOC-2.2.2, "Personnel Training" (Reference 18.3-6).

The Analysis Stage provides the identification of training needs, tasks, or competencies required for training and the knowledge, skills, and abilities required to perform a specified job position based on assigned tasks. Tasks that support plant functions are identified as part of Detailed and Basic TA, described in Subsection 18.2.4. Tasks are selected for training based on difficulty, importance, and frequency analysis. Depending on the difficulty, importance, and frequency ranking, a decision is made to determine if initial training and periodic retraining is needed. This evaluation of training tasks is the training equivalent to grading human actions (Subsection 18.1.4.2). The results of the TA, including identification of critical steps, inform the difficulty, importance, and frequency analysis and resulting rankings.

Depending on the difficulty, importance, and frequency ranking, a determination is made if initial and periodic retraining is required. Tasks that are selected for training are then analyzed to determine the required knowledge, skills, and attributes. The knowledge, skills, and abilities necessary for each job position, including entry-level education, training, and experience, is established to support training design. Any changes to the iterative HFE or system design inputs to the analysis phase are required to be assessed for impact on the training analysis.

During the Design Stage, learning objectives are developed and a description of the plan for training, including purposed methods and settings, is established. Specifically, the Design Stage includes the following activities:

- Determine the scope, purpose, and timeframe of the training
- Determine the ideal training environment
- Select training methods and instructional strategies in accordance with the environment
- Determine and group the job role knowledge, skill, and attributes addressed by each training module
- Determine the final and partial learning objectives for each training module, including defining performance statements, conditions statements and performance standards
- Prepare the table of contents and scope for each training module; scope includes number and type of documents developed in the next phase
- Prepare master training procedures and formats to ensure consistency across the course materials
- Prepare the training plans for each job position; plans comprise learning objectives, contents, learning activities, training equipment, and a list of materials needed for training, including guidance for their use

The completion of the Design Stage establishes the input that is needed for the Development Stage.

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In the Development Stage, detailed lesson plans and instructional materials are created, including any on-the-job training documents, and knowledge and performance assessment tests are established. The materials developed must incorporate the required features specified in the Design Stage. The materials are developed such that they have the following attributes:

- Course material content supports mastery of the subject learning objectives.
- Course materials are structured to provide consistent presentation.
- Course material presentation sequence supports effective learning.
- Course materials support successful presentation in the specified venue(s) the course is to be provided.
- Instructor certifications and training required to present training is specified for each course and supports successful presentation in the required venue. Instructors are trained during this phase.
- Exam question banks and examination structure and content are developed to adequately evaluate, and document trainee mastery of the course and job performance objectives associated with the training.

At the end of the Development Stage, the training package is reviewed, piloted on trainees, and revised if necessary.

In the Implementation Stage, instructors prepare for and deliver the training. Trainees are tested to determine if they have mastered the objectives. The results of trainee tests are examined during the Evaluation Stage. The Evaluation Stage examines the effectiveness of the training as delivered. This appraisal is done through the review of training results, training feedback, and continual monitoring of work performance (Section 18.6).

The training and qualification program, developed as discussed above, provides assurance that plant personnel have the capability and competence needed to perform their roles and responsibilities. Participants used for ISV, as described in Section 18.5, are trained using this program and provide validation of the integrated design.

18.3.8.3 Results

The specific program outputs include documentation defining the overall program goals and course structure, as well as the specific job role qualification and training requirements and developed course materials. In addition to the training program content itself, the results of the training and qualification program development are summarized in a report which documents the process and activities used in development, including any inputs used, issues identified, and recommendations made.

18.3.9 References

- 18.3-1 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 18.3-2 CNSC Regulatory Document REGDOC-2.5.1, "General Design Considerations: Human Factors."
- 18.3-3 CNSC Regulatory Document REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response."
- 18.3-4 IAEA-TECDOC-1058, "Good Practices with Respect to the Development and Use of Nuclear Power Plant Procedures," International Atomic Energy Association.

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- 18.3-5 CNSC Regulatory Document REGDOC-2.2.3 "Personnel Certification, Volume III: Certification of Reactor Facility Workers."
- 18.3-6 CNSC Regulatory Document REGDOC-2.2.2, "Personnel Training."

18.4 Human Factors Verification and Validation

HFE V&V is a critical HFE design assurance activity applied to the realized design of the plant HSIs and the working environment where those HSIs are used. The HFE V&V program evaluates the plant design (in parts and as an integrated whole) against HFE design principles and requirements, user task requirements, job design and staff complement, procedural accuracy and usability, and effectiveness of training.

18.4.1 Objectives and Scope

The HFE Verification is conducted through two activities with the following objectives:

1. Task Support Verification (TSV) verifies that the HSIs, as defined and baselined in the HSI inventory and characterization, include the necessary features (e.g., controls, information displays, and alarms) required to support tasks and that there are no unnecessary features.
2. HFE Design Verification verifies that the HSIs and plant SSC, are compliant with the applicable HFE design requirements contained in the HFE Design Requirements Document and design-to-analysis requirements input as a result of HFE analysis activities. Verification activities include identifying changes to the design that impact HSIs and other features due to competing design constraints, and checking for due consideration of OE items, user stakeholder input and HFE T&E results.

HFE Validation is conducted through staged activities, as follows:

1. Early and Partial System Validation activities are performed in advance of the full-scope simulator and fully constructed plant and are generally performed only on partial systems. Although they require a sufficient maturity of the design, HFE participants from the V&V team, and end users with enough level of familiarity with the system, they do not require the full integrated system. The purpose of these validation activities is to identify and solve HFE issues in advance of a fixed design.
2. ISV is the performance-based evaluation of the fully integrated system design. Simulations and virtual reality models are used to validate the ability of personnel, trained using the training and qualification program material, to use the integrated HSIs and finalized procedures in accordance with the task and scenario performance requirements. ISV is intended to evaluate those integrated aspects that were verified and validated singly through earlier, partial means.

HFE Validation ensures that the design, particularly the HFE-specified aspects, accomplishes its intended goals for usability and reducing the risk of human error to as low as reasonably achievable. Validation is an integrated, dynamic, performance-based test activity in which participants are subjected to a set of simulated scenarios that represent a realistic, challenging, and generalizable set of conditions to ensure that the integrated HSI supports safe operation of the plant.

The scope of the HFE V&V activities applies to user interactions with the plant when performing operations, maintenance, testing, and inspection activities. The HFE V&V activities are applied to HSIs within scope of the HFE Program. As with the other HFE Program activities, the application of HFE V&V is graded to focus on the HSIs, tasks, and plant conditions that involve important human actions, are complex or novel, or are inherently hazardous. The same risk-based approach described in Subsection 18.1.4.2 is applied to the HFE V&V activities to determine the appropriate scope, rigour, and level of detail for each activity.

18.4.2 Methodology

The HFE V&V program is conducted in accordance with a structured, systematic plan. The program was developed to meet the requirements and best-practice guidance specified in CNSC REGDOC-2.5.1, "General Design Considerations: Human Factors" (Reference 18.4-1), CNSC REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants" (Reference 18.4-2), CSA N290.12-14, "Human factors in design for nuclear power plants" (Reference 18.4-3), IEC 61771, "Nuclear power plants – Main Control Room – Verification and validation of design" (Reference 18.4-4), and IAEA SSG-51, "Human Factors Engineering in the Design of Nuclear Power Plants" (Reference 18.4-5). The program adopts a risk-based graded and multi-staged approach to V&V.

The V&V program plan specifies overall process used for HFE V&V, and the scope, inputs, methods, and outputs to be used for each V&V activity. The overall process is shown in Figure 18.4-1.

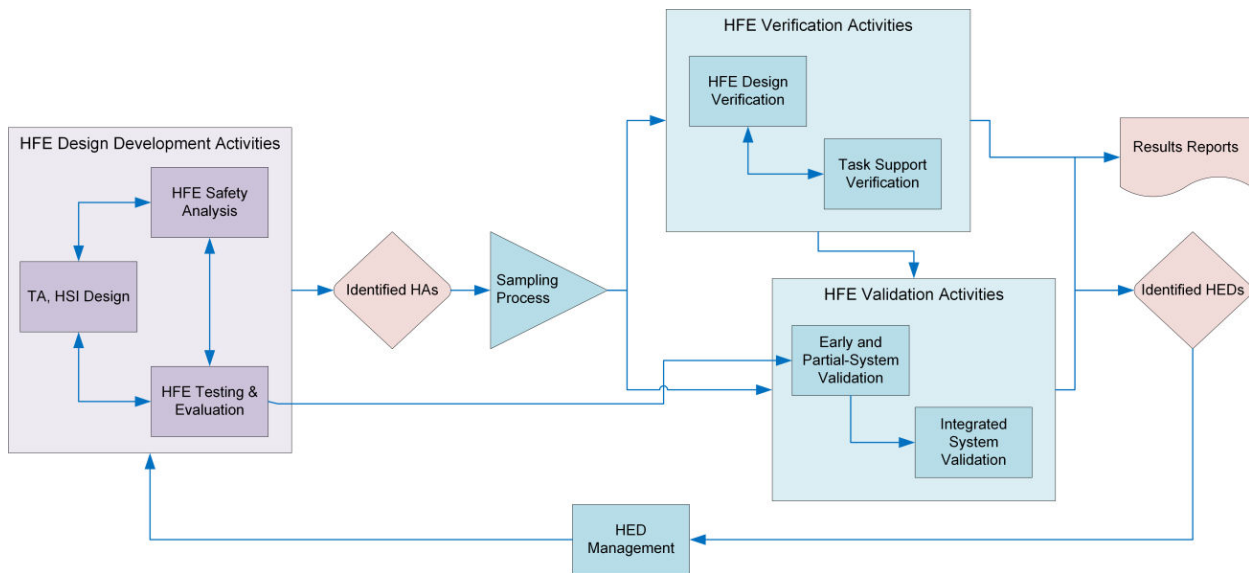


Figure 18.4-1: BWRX-300 HFE V&V Process Overview

The Sampling Process is a support activity that establishes the scope of the HFE V&V activities. In a new plant design, the number of scenarios and HSIs is too large to effectively perform HFE V&V to the same degree on all of them. The purpose of the Sampling Process is to focus on the significant, novel, and complex HSIs and tasks, ensuring a full breadth of HFE V&V scope but removing any duplication, thus improving the efficacy of the HFE V&V activities.

The Sampling Process selects the inputs that bound the scope of the HFE V&V activities. The verification activities target a selection of HSIs (e.g., displays, panel layouts, equipment-mounted controls, and indications) and the validation activities target a selection of scenarios. The goal of sampling is to maximize sample relevance and significance while ensuring that the sample is sufficiently broad and diverse, so that the HFE V&V results are generalizable to the overall population of HSIs and scenarios.

TSV compares the HSI elements (alarm, control, information and equivalent) identified during the detailed analysis of a task to the designed HSIs to ensure that all components needed to safely and efficiently complete the tasks present in the final design. The task support inventory and verification criteria are identified during TA (Subsection 18.2.4).

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In HFE Design Verification, various aspects of HSI and plant SSC design are compared to the relevant design requirements specified during the design development (Subsections 18.3.1 through 18.3.6). The aspects verified include:

- Static and dynamic HSI features, including HSI-specific and standardized features
- Interface management features such as navigation and data retrieval
- Workstations and workspace anthropometrics
- Global workspace features (i.e., layout, workplace environment, lighting, noise)
- Effects of degraded HSI and plant workplace conditions

During HFE Design Verification, the HFE verifier documents each HSI, or plant SSC element being evaluated (including document and page numbers, screenshots, or photographs as applicable), which subset of HFE Design Requirements Document requirements were applied, and whether the HSI or SSC element passed or failed each requirement.

Early Validations form an essential part of the HFE Validation activities; they are performed to identify and solve HFE issues in advance of a fixed design. They require a sufficient maturity of the design, HFE participants from the V&V team, and end users with enough level of familiarity with the system. However, they do not require the full integrated system, including trained users and final procedures, that the ISV requires. Early Validations are expected to progress HFE T&E activities and results (Subsection 18.3.4), using higher fidelity testbeds and more cohesive scenario-based sets of tasks. The general method for conducting the Early Validations is the same as that for ISV, without the requirements for a complete integrated system and complex high-fidelity testing environments.

ISV is the performance-based evaluation of the fully integrated system design. Simulations and virtual reality models are used to validate the ability of personnel, trained using the training and qualification program material (Subsection 18.3.8), to use the integrated HSIs and finalized procedures such that they support safe plant functionality. ISV is intended to evaluate those integrated aspects that were verified separately through earlier, partial means (i.e., through HFE T&E, TSV, Early Validation). ISV is performed using high-fidelity simulators, task trainers or virtual reality labs (i.e., for scenarios outside of control rooms and control stations). The ISV is the final activity that ensures the integrated design is fulfilling its intended function and demonstrates that claims made in the safety analyses are achievable to the performance requirements specified.

The general method for conducting either early validations or ISV is:

1. Perform preliminary activities
 - a. Scenario identification and development
 - b. Testbed verification
2. Perform testing
 - a. Participant selection
 - b. Scenario definition and documentation
 - c. Performance measures
 - d. Test design
 - e. Pilot testing
3. Perform data analysis and document results

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Any issues or non-compliances identified during the HFE V&V activities are identified as a HED. HEDs are processed using the HFEITS process as described in Subsection 18.1.5.

18.4.3 Results

The results of all the HFE V&V activities are captured in the following indicative documents:

1. HFE Verification Results Report
2. HFE Early and Partial Validation Result Report(s)
3. HFE ISV Test Specification
4. HFE ISV Summary Report

Another output of the HFE V&V activities is identified HEDs, which are captured in the HFEITS and managed through the process described in Subsection 18.1.5.

18.4.4 References

- 18.4-1 CNSC Regulatory Document REGDOC-2.5.1, "General Design Considerations: Human Factors."
- 18.4-2 CNSC Regulatory Document REGDOC-2.5.2, "Design of Reactor Facilities: Nuclear Power Plants."
- 18.4-3 CSA N290.12, "Human factors in design for nuclear power plants," CSA Group.
- 18.4-4 IEC 61771, "Nuclear power plants – Main control room – Verification and Validation of Design," International Electrotechnical Commission.
- 18.4-5 IAEA Safety Standards Series No. SSG-51, "Human Factors Engineering in the Design of Nuclear Power Plants," International Atomic Energy Association.

18.5 Design Implementation

Design Implementation addresses the implementation of the HFE design requirements in the final realized design.

18.5.1 Objectives and Scope

The HFE Design Implementation activities have the following objectives:

1. Confirm that the final realized HSIs, plant SSC, procedures, and training conform to the design requirements and design documents resulting from the HFE Program activities
2. Identify any deviations from the design during implementation and assess their impact on the HFE aspects of the design
3. Perform final procedure validation on the physical plant hardware
4. Verify aspects of the design that may not have been evaluated previously in the V&V process, i.e., any HSIs that were absent or modified from the simulator-based ISV, or any plant physical or work environment (e.g., noise, lighting, thermal) characteristics; verification of items not previously identified as needing evaluation uses the same grading process as the original verification during the design stage
5. Verify and document the resolution of all remaining HFE issues and HEDs (Subsection 18.1.5)
6. Verify HFE Application Levels (Subsection 18.1.4.2) are correct based on the final version of documents and data used as input to determine the levels

The scope of the design implementation is the full set of HFE aspects of the plant including design of the HSIs and plant SSC, plant procedures, and finalized training documentation.

18.5.2 Methodology

Unlike the other HFE technical elements, Design Implementation is performed after the design is complete, immediately prior to commencement of commercial operations. Despite the plant being built and undergoing start-up testing and commissioning, the HFE team still performs the Design Implementation activities. The process follows the HFE V&V process. In this case, the Sampling Process includes identifying any changes to the standardized plant design (documented through engineering configuration control process) and assessing impact on HFE aspects of the design. The Sampling Process also identifies aspects of the design that were not able to be previously verified and validated. Additional items not previously identified for V&V but determined to require it are also added to the scope for Design Implementation HFE V&V.

The list of methods used are similar to those described for the HFE Program activities, particularly HFE V&V, Staffing Analysis, Procedure Development, Training and Qualification Program Development, and HFE Issues Tracking.

At this stage, the plant design is in formal engineering configuration control, as described in Chapter 17.

18.5.3 Results

The results of the Design Implementation activities match the outputs for the same activities conducted during the design phase. They are recorded in a summary report. Any remaining HFE issues or HEDs are recorded in the HFEITS for turnover to the plant operating organization, and further mitigation through operating arrangements as required.

18.6 Human Performance Monitoring

The HPM strategy links HFE methods used during the design with methods for monitoring user task performance during operation. The HPM program is fully developed by the licence applicant as part of the future licencing stage.

18.6.1 Objectives and Scope

The purposes of HPM are:

1. To ensure that the high safety standards established by the HFE Program during the design of the plant are maintained even when changes are made to the plant
2. To detect any deterioration of task performance that may be attributable to latent or slow-developing HFE design issues
3. To provide adequate assurance that the safety bases remain valid during the operational phase of the plant

There is no intent for the HSI designer or the applicant to periodically repeat a full set of ISV activities. The strategy is to provide a monitoring plan, building upon the HFE activities during the design that can be carried forward into the operational phase, using industry accepted methods. HPM incorporates the monitoring strategy into the problem identification and corrective action program, which identifies and classifies human errors, provide for evaluation of the root cause, and supports effectiveness verification and documentation of the corrective action.

The scope of the performance monitoring strategy provides reasonable assurance that:

1. The HSI design is effective during:
 - a. Normal operations
 - b. Maintenance, Inspection, Testing, and Surveillance
 - c. Anticipated Operational Occurrences
 - d. Design Basis Accidents
 - e. Design Extension Conditions
 - f. Severe Accidents
2. Human actions, using HSI information, cues and controls can accomplish tasks while maintaining margin for time and performance criteria.
3. Acceptable performance levels established during the HFE ISV are maintained.
4. Changes made to the initial HSIs, user group definition, job design, procedures, and training do not have adverse effects on personnel task performance (e.g., a change interferes with trained skills, or a fatigue management policy is not implemented, contrary to what was assumed in the HFE COO).

18.6.2 Methodology

The HPM program aligns with the overall quality program and condition reporting methods. The program includes:

- Data collection
- Importance screening
- Event analysis to determine causes

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- Trend analysis
- Corrective action development

The HPM strategy collects data to trend task performance, particularly seeking issues with design root causes due to non-compliance with or inappropriate application of HFE principles. The HPM program uses existing utility or industry programs (e.g., corrective action, programs, or operator training) for data collection where appropriate. The HPM program is designed to ensure that:

1. Human actions are monitored commensurate with their safety importance.
2. Feedback of information and corrective actions are accomplished in a timely manner.
3. Degradation in performance is detected and corrected before plant safety is compromised.

NOTE: The HFE-based HPM does not seek personnel behaviour-based corrective actions. It is focused solely on issues related to the design of HSIs and organizational arrangements that lead to human error.

The HPM program maintains a database of event causes and corrective actions taken. Such data supports trending of performance anomalies.

The HPM identifies and establishes corrective actions that reduce the potential for incident recurrence. The program systematically identifies the cause of the failure or degraded performance. The corrective actions are derived by:

1. Addressing the significance of the failure through application of PRA/HRA importance measures
2. Classifying the causes and circumstances surrounding the failure or degraded human performance
3. Illuminating the characteristics of the failure (e.g., being task specific or due to design issues)
4. Determining whether the failure is isolated or has generic or common cause implications

18.6.3 Results

The HPM program activities and outputs align with the overall condition reporting requirements. They are expected to include specific incident or trend analysis reports, a recommendations and action tracking database, and periodic summary reports.

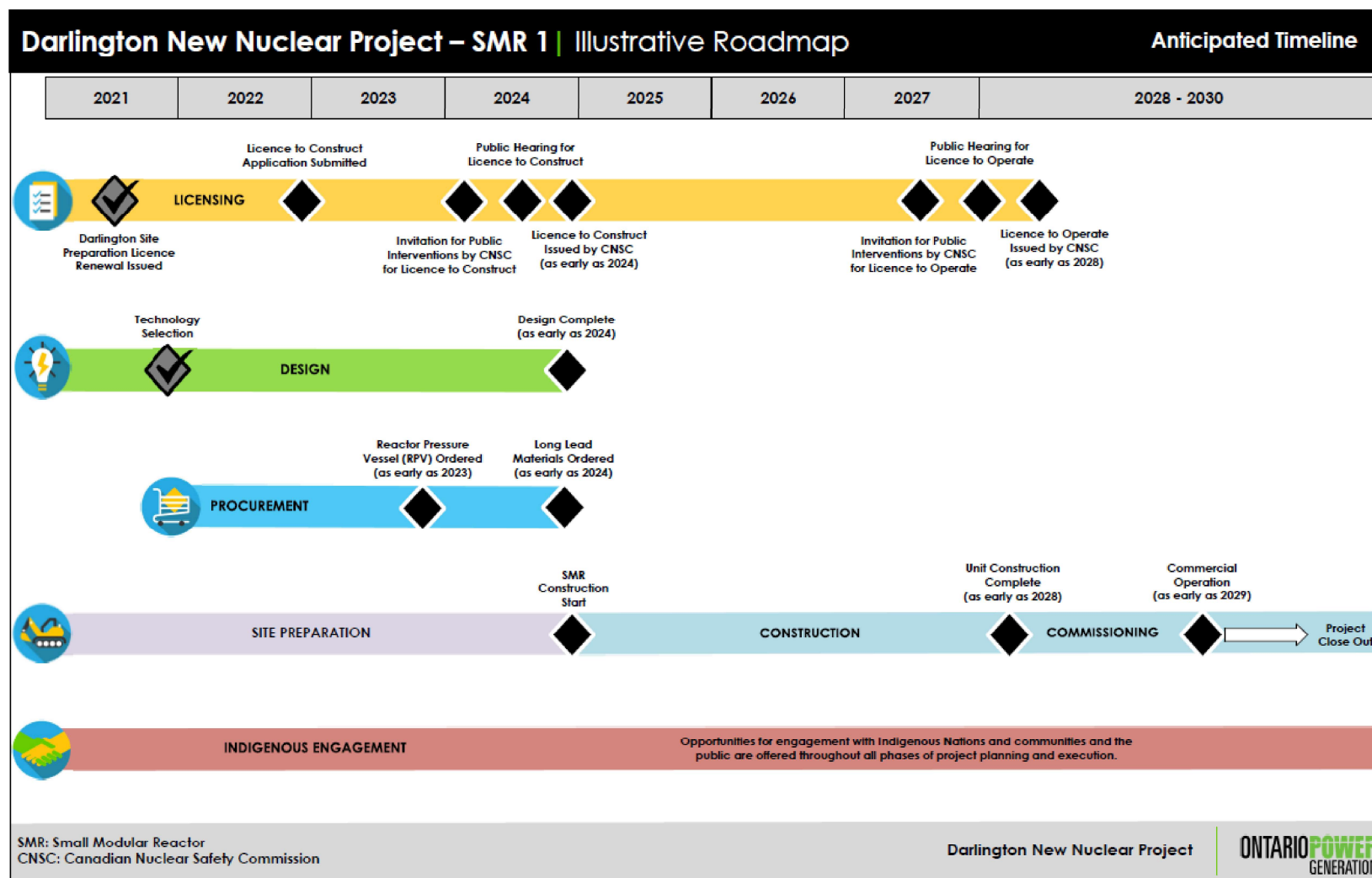


Figure 1.1-2: DNNP Project Roadmap



HITACHI

GE Hitachi Nuclear Energy

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May 2023

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**BWRX-300 Darlington New Nuclear
Project (DNNP)
REGDOC-2.5.2 Compliance Matrix
Report**

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT
Please Read Carefully

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
0	All	Originally Issued as Proprietary Version NEDC-33975P Rev 0
1	All	Initial Issue as Non-Proprietary Version

1.0 INTRODUCTION

The Darlington New Nuclear Project (DNNP) intends to design and construct a BWRX-300 Small Modular Reactor (SMR) for the Darlington Nuclear Site. A Preliminary Safety Analysis Report (PSAR) is required by the *Class I Nuclear Facilities Regulations* (SOR/2000-204) paragraph 5(f) (Reference 3-1) for submission as part of the Licence to Construct (LTC) to the Canadian Nuclear Safety Commission (CNSC). The PSAR demonstrates the adequacy of the design and includes a safety assessment.

REGDOC-1.1.2, *Licence Application Guide: Licence to Construct a Nuclear Power Plant* specifies detailed requirements for a licence application and acknowledges that licence applications may be submitted prior to full design completion (Reference 3-2). REGDOC-1.1.2 points to requirement of REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants* for a complete design of water-cooled nuclear power plants (Reference 3-3).

The PSAR format is developed in accordance with International Atomic Energy Agency (IAEA) Specific Safety Guide SSG-61, *Format and Content of the Safety Analysis Report for Nuclear Power Plants* (Reference 3-4) and demonstrates compliance with CNSC REGDOC-1.1.2 (Reference 3-2) and REGDOC-2.5.2 (Reference 3-3), which establishes a set of comprehensive design requirements and guidance that are risk-informed and align with accepted international codes and practices.

1.1 Purpose

Sections 1 to 11 of REGDOC-2.5.2 are mapped to the PSAR for the purposes of this compliance review (Reference 3-3). The compliance matrix developed in Table 2-1 reflects the current phase in the design process to support the LTC application (Reference 3-5), which is iterative in nature and may be subject to change as design and safety analysis iteration progress into subsequent phases.

Supporting details to substantiate compliance statements may not be available at this time in the design process; however, further details on compliance with REGDOC-2.5.2 requirements will be available in subsequent design phases as referenced in the PSAR. Table 2-1 provides a roadmap to demonstrate that BWRX-300 plant requirements meet the requirement of REGDOC-2.5.2, given the current phase in design. Where a specific REGDOC-2.5.2 requirement is not met as written, an alternative approach is identified in Table 2-1 and justified in NEDC-33974P, BWRX-300 DNNP REGDOC-2.5.2 Alternative Approach Report (Reference 3-6).

The GEH design process also complies with the management system requirements of REGDOC-2.5.2 (Reference 3-3).

1.2 Scope

Sections 1 to 11 of REGDOC-2.5.2 (Reference 3-2) are mapped to the PSAR for the purposes of this compliance review. The compliance matrix developed in Table 2-1 reflects the current phase in the design process, which is iterative in nature and may be subject to change as design and safety analysis iteration progresses into subsequent phases in design.

2.0 COMPLIANCE MATRIX

2.1 Methodology

Each section of REGDOC-2.5.2 is mapped to Chapters in the PSAR in Table 2-1 (Reference 3-3). The REGDOC-2.5.2 content is classified per section in the table as follows:

- Requirement – for regulatory document sections that contain “shall” statements
- For Information to Support Requirement – for regulatory document requirements that do not contain “shall” statements

Guidance is listed in Table 2-1 as considered, if applicable. Compliance notes are used to provide interfacing information or clarification as applicable. The compliance matrix is consistent with the NEDC-33974P, BWRX-300 DNNP REGDOC-2.5.2 Alternative Approach Report, when applicable (Reference 3-6).

2.2 Assessment

The REGDOC-2.5.2 compliance matrix for REGDOC-2.5.2 V1 is presented in Table 2-1 (Reference 3-3).

2.3 Conclusion

The results in Table 2-1 demonstrate compliance with the requirements of REGDOC-2.5.2 V1 (Reference 3-3). Where an alternative approach is identified, an assessment demonstrating compliance with the intent/safety objective of REGDOC-2.5.2 V1 is demonstrated through the PSAR and documented in the NEDC-33974P, BWRX-300 DNNP REGDOC-2.5.2 Alternative Approach Report (Reference 3-6). Table 2-1 contains compliance statements that reflect the current state of design. The compliance notes also describe instances where interfacing information is included in various sections within the PSAR.

Table 2-1: REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants, May 2014, Version 1 Compliance Matrix

REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
1. Purpose	For Information to Support Requirement	The BWRX-300 design complies with Section 1. of REGDOC-2.5.2 V1 as written.	Chapter 1-21, including Security Annex and Safeguards Annex	Refer to Chapter 1, Section 1.11 for conformance with applicable regulations, codes and standards. Refer to Chapter 3, Section 3.11 for compliance with national and international standards.
2. Scope	For Information to Support Requirement	The BWRX-300 design complies with Section 2. of REGDOC-2.5.2 V1 as written.	Chapter 1-21, including Security Annex and Safeguards Annex	Refer to Chapter 1, Section 1.1 for the format of the PSAR. Refer to Chapter 3, Section 3.1 for the general safety design basis and safety goals. Refer to NEDC-33974P, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report.
3. Relevant Legislation	Requirement	The BWRX-300 design complies with Section 3. of REGDOC-2.5.2 V1 as written.	Chapter 1-21, including Security Annex and Safeguards Annex	Various references in PSAR. Refer to Chapter 1, Section 1.1 for the format of the safety analysis report.
4. Safety Objectives and Concepts	For Information to Support Requirement	The BWRX-300 design complies with Section 4. of REGDOC-2.5.2 V1 as written.	Chapter 1-21, including Security Annex and Safeguards Annex	Various references in PSAR. Refer to Chapter 1, Section 1.1 for the format of the safety analysis report.
4.1 General nuclear safety objective	For Information to Support Requirement	The BWRX-300 design complies with Section 4.1 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.1 Chapter 3, Subsection 3.1.1 Chapter 15, Subsection 15.1.1	Refer to Subsection 3.1.1 for the General Nuclear Safety Objective.
4.1.1 Radiation projection objective	Requirement	The BWRX-300 design complies with Section 4.1.1 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.1 Chapter 3, Subsection 3.1.1 Chapter 3, Subsection 3.1.2 Chapter 12, Section 12.1 Chapter 12, Section 12.2 Chapter 12, Section 12.3 Chapter 12, Section 12.5	Refer to Subsection 3.1.2 for the Radiation Protection Objective. Refer to Chapter 12 for As Low As Reasonably Achievable (ALARA) design criteria.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
4.1.2 Technical safety objectives	For Information to Support Requirement	The BWRX-300 design complies with Section 4.1.2 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.1 Chapter 3, Subsection 3.1.1 Chapter 3, Subsection 3.1.4 Chapter 3, Subsection 3.1.5 Chapter 3, Subsection 3.1.6	Refer to Subsection 3.1.1 for the Technical Safety Objective.
4.1.3 Environmental protection objective	Requirement	The BWRX-300 design complies with Section 4.1.3 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.1 Chapter 3, Subsection 3.1.1 Chapter 11 Chapter 20	Refer to Subsection 3.1.1 for an overview of the Environmental Protection Objective. Refer to Chapter 11 for management of radioactive waste. Refer to Chapter 20 for environmental aspects.
4.2 Application of the technical safety objectives	Requirement	The BWRX-300 design complies with Section 4.2 of REGDOC-2.5.2 V1 as written.	Chapter 3, Subsection 3.1.2 Chapter 3, Subsection 3.1.2.3 Chapter 15, Section 15.1 Chapter 15, Section 15.3 Chapter 15, Section 15.7	Refer to Subsection 3.1.2 for radiation protection and radiological acceptance criteria. Refer to Subsection 3.1.2.3 for safety goals. Refer to Chapter 15 for supporting analyses.
4.2.1 Dose acceptance criteria	Requirement	The BWRX-300 design complies with Section 4.2.1 of REGDOC-2.5.2 V1 as written.	Chapter 3, Subsection 3.1.2 Chapter 3, Subsection 3.1.2.2 Chapter 15, Section 15.1 Chapter 15, Section 15.3 Chapter 15, Section 15.7	Refer to Subsection 3.1.2.2 for radiological acceptance criteria. Refer to Chapter 15 for supporting analyses.
4.2.2 Safety goals	Requirement	The BWRX-300 design complies with Section 4.2.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.2.3 Chapter 3, Subsection 3.1.6 Chapter 15, Section 15.3 Chapter 15, Subsection 15.3.2 Chapter 15, Section 15.6	Refer to Subsection 3.1.2.3 for safety goals. Refer to Chapter 15 for supporting analyses, including Section 15.6 for the Probabilistic Safety Assessment (PSA).
4.2.3 Safety analyses	Requirement	The BWRX-300 design complies with Section 4.2.3 of REGDOC-2.5.2 V1 as written.	Chapter 2, Section 2.2 Chapter 3, Subsection 3.1.3 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.5 Chapter 15, Section 15.6	Refer to Chapter 2 for the identification and characterization of external hazards. Key elements of the Safety Strategy Framework are described in Chapter 3. Refer to Subsection 3.1.3 for plant states considered in the design basis. Refer to Chapter 15 for supporting analyses.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
4.2.4 Accident mitigation and management	Requirement	The BWRX-300 design complies with Section 4.2.4 of REGDOC-2.5.2 V1 as written.	Chapter 3, Subsection 3.1.4 Chapter 3, Subsection 3.1.5 Chapter 3, Subsection 3.1.6 Chapter 4-12 Chapter 13, Subsection 13.4.3 Chapter 18, Subsection 18.3.7 Chapter 19, Section 19.1 Chapter 19, Section 19.2	Refer to Subsection 3.1.4 for prevention and mitigation of accidents. Refer to Chapter 4 through 12 for the specific system design and safety functions considered in the BWRX-300 design basis. Refer to Chapter 6 for the Engineered Safety Features (ESFs). Refer to Subsection 13.4.3 for the development of emergency procedures and guidelines.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
4.3 Safety Concepts				
4.3.1 Defence in depth	Requirement	The BWRX-300 design complies with Section 4.3.1 of REGDOC-2.5.2 V1 as written.	Chapter 3, Subsection 3.1.6 Chapter 3, Subsection 3.1.7.5 Chapter 3, Appendix 3A Chapter 15, Section 15.2 Chapter 15, Section 15.7	Defense Lines (DLs) in the plant design envelope for operational and accident conditions are adopted consistent with IAEA SSR-2/1, Safety of Nuclear Power Plants as outlined in Subsection 3.1.6.2 of the PSAR. Subsection 3.1.6 outlines the Fault Evaluation process; the selection and categorization of Postulated Initiating Events (PIEs) and fault sequences for deterministic analysis is described in Section 15.2.
4.3.2 Physical barriers	Requirement	The BWRX-300 design complies with Section 4.3.2 of REGDOC-2.5.2 V1 as written.	Chapter 3, Subsection 3.1.5 Chapter 3, Subsection 3.1.6 Chapter 3, Subsection 3.1.6.1	Refer to Subsection 3.1.6.1 for physical barriers in place to prevent the release of radioactivity. Refer to REGDOC-2.5.2, Section 6.1.1 for further discussion on physical barriers.
4.3.3 Operational limits and conditions	Requirement	The BWRX-300 design complies with Section 4.3.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 13, Subsection 13.3.3 Chapter 16	Refer to Chapter 16 for the basis on which the Operational Limits and Conditions (OLCs) are derived. Refer to Subsection 13.3.3 for surveillance, maintenance, testing and inspection practices.
4.3.4 Interface of safety with security and safeguards	Requirement	The BWRX-300 design complies with Section 4.3.4 of REGDOC-2.5.2 V1 as written.	Chapter 13, Section 13.5 Safeguards Annex, Section 7.0 Safeguards Annex, Section 8.0 Security Annex	Refer to Section 13.5 for nuclear safety and nuclear security interfaces. Refer to Section 7.0 of the Safeguards Annex for nuclear material accountancy information. Refer to Section 8.0 of the Safeguards Annex for safety, safeguards and security interface.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
5. Safety Management in Design	Requirement	The BWRX-300 design complies with Section 5. of REGDOC-2.5.2 V1 as written.	Chapter 14, Section 14.1 Chapter 14, Section 14.2 Chapter 14, Section 14.3 Chapter 17, Section 17.2 Chapter 17, Section 17.3 Chapter 17, Subsection 17.3.1.2 Chapter 17, Section 17.5	Refer to Chapter 14 for the plant construction and commissioning responsibility within the OPG management system. Refer to Subsection 17.3.1.2 for GEH's Quality Management Plan and supporting plans for design management. Refer to Section 17.5 for specific programs associated with Safety Culture elements in OPG's management system.
5.1 Design authority	Requirement	The BWRX-300 design complies with Section 5.1 of REGDOC-2.5.2 V1 as written.	Chapter 13, Subsection 13.3.6 Chapter 17, Section 17.2 Chapter 17, Subsection 17.2.1.1 Chapter 17, Section 17.3 Chapter 17, Subsection 17.3.1.2	Refer to Subsection 17.2.1.1 for roles of the Design Authority (DA). Refer to Section 17.3 for design management plans that cover the DA. Refer to Subsection 17.3.1.2 for GEH's Quality Management Plan and supporting plans for design management.
5.2 Design management	Requirement	The BWRX-300 design complies with Section 5.2 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.2 Chapter 11 Chapter 13, Subsection 13.3 Chapter 14, Section 14.1 Chapter 15 Chapter 16 Chapter 17, Section 17.3 Chapter 17, Subsection 17.3.1.2 Chapter 18 Chapter 20 Security Annex	Refer to Subsection 17.3.1.2 for GEH's Quality Management Plan and supporting plans for design management and control process. Refer to Section 3.2 for the Safety Classification process. Structures, Systems, and Components (SSC) classification includes Safety Class 1, 2, 3 and Non-Safety Class. BWRX-300 does not use "important to safety" terminology as a classification category. Refer to Security Annex for physical protection systems and cyber security programs are provided to address design- basis threats.
5.3 Design control measures	Requirement	The BWRX-300 design complies with Section 5.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Appendix 3B-G Chapter 13, Subsection 13.3.6 Chapter 17, Section 17.3 Chapter 17, Subsection 17.2.1.2 Chapter 17, Subsection 17.3.1.2	Refer to Appendix 3B-G, as applicable. Refer to Subsection 13.3.6 for control of modifications. Refer to Subsection 17.3.1.2 for GEH's Quality Management Plan and supporting plans for design management and control process.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
5.4 Proven engineering practices	Requirement	The BWRX-300 design complies with Section 5.4 of REGDOC-2.5.2 V1 as written.	Chapter 1, Section 1.11 Chapter 3, Subsection 3.1.7 Chapter 3, Section 3.3 Chapter 3, Section 3.4 Chapter 3, Section 3.5 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 3, Section 3.11 Chapter 4-12 Chapter 17, Section 17.3 Chapter 17, Subsection 17.3.1.2	The GEH BWRX-300 utilizes proven SSC designs based on over 60 years of operating experience (OPEX), testing of components, and experimental research of its Boiling Water Reactor (BWR) design as outlined in the PSAR. Refer to Sections 3.3 to 3.9 for design criteria for structures, mechanical systems, electrical systems, Instrumentation and Control (I&C) systems, and equipment qualification. Chapters 4 through 12 describe the SSCs. Refer to Chapter 17 for the BWRX-300 Research and Development Program, which includes OPEX.
5.5 Operational experience and safety research	Requirement	The BWRX-300 design complies with Section 5.5 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 4-12 Chapter 17, Subsection 17.3.1.2 Chapter 17, Section 17.4	The GEH BWRX-300 utilizes proven SSC designs based on over 60 years of OPEX, testing of components, and experimental research of its BWR as outlined in the PSAR. Refer to Chapter 17 for the BWRX-300 Research and Development Program, which includes OPEX.
5.6 Safety assessment	Requirement	The BWRX-300 design complies with Section 5.6 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.1 Chapter 13, Section 13.3 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.4 Chapter 15, Section 15.5 Chapter 15, Section 15.6 Chapter 15, Section 15.7 Chapter 17, Subsection 17.2.1.2	Refer to Subsection 3.1.6 for defence in depth. Refer to Chapter 15 for reference to the iterative nature of design and analyses. An independent peer review of Chapter 15 to meet the requirements of REGDOC-2.5.2 Section 5.6 will be submitted by OPG as part of the LTC application. Refer to the LTC application plan outlined in NK054-CORR-00531- 10667.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
5.7 Design documentation	Requirement	The BWRX-300 design complies with Section 5.7 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.2 Chapter 13 Chapter 4-12 Chapter 17	PSAR is a summary document, that is based on a suite of verified engineering and design documentation. Refer to Chapter 13 and 17 for GEH and OPG configuration management information that will be used to demonstrate the adequacy of design documentation for procurement, construction, commissioning, and safe operation including maintenance, aging management, modification, and eventual decommissioning of the reactor facility.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
6. Safety Requirements				
6.1 Application of defence in depth	Requirement	The BWRX-300 design complies with Section 6.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.1 Chapter 3, Subsection 3.1.6 Chapter 3, Subsection 3.1.6.2 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.4	Refer to Section 3.1 for an overview of defence in depth. Refer to Sections 15.1 to 15.4 for the basis and methodology used in analyses.
6.1.1 Physical barriers	Requirement	The BWRX-300 design complies with Section 6.1.1 of REGDOC-2.5.2 V1 as written.	Chapter 2, Subsection 2.1.2 Chapter 3, Subsection 3.1.6 Chapter 4, Section 4.2 Chapter 4, Section 4.4 Chapter 4, Subsection 4.4.5 Chapter 5, Section 5.1 Chapter 5, Section 5.2 Chapter 5, Section 5.3 Chapter 5, Section 5.4 Chapter 6, Section 6.3 Chapter 15, Section 15.1 Chapter 15, Section 15.3	Physical barriers, as outlined in Subsection 3.1.6.1, include the fuel cladding, Reactor Coolant Pressure Boundary (RCPB), and the containment. Chapter 2, Subsection 2.1.2 contains information on the exclusion zone for DNNP. Refer to Subsection 4.4.5 for the flow stability evaluation.
6.2 Safety functions	Requirement	The BWRX-300 design complies with Section 6.2 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.1 Chapter 3, Subsection 3.1.5 Chapter 3, Subsection 3.1.6 Chapter 3, Section 3.2 Chapter 3, Appendix 3A Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.4 Chapter 15, Section 15.5	Refer to Subsection 3.1.5 for Fundamental Safety Functions (FSFs) for BWRX-300 design. SSC classification includes Safety Class 1, 2, 3 and Non- Safety Class as outlined in Section 3.2. BWRX-300 does not use "safety support system" terminology. SSC safety functions are covered in the PSAR. Refer to Chapter 15 for supporting analyses. Section 15.1 to 15.5 reference operational states, Design Basis Accidents (DBAs) and Design Extension Conditions (DECs).

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
6.3 Accident prevention and plant safety characteristics	Requirement	The BWRX-300 design complies with Section 6.3 of REGDOC-2.5.2 V1 as written.	Chapter 3, Subsection 3.1.4 Chapter 3, Subsection 3.1.6 Chapter 3, Section 3.5 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 3, Appendix 3A Chapter 4-12 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.5 Chapter 15, Section 15.6 Chapter 15, Section 15.7 Chapter 15, Appendix 15B	Refer to Subsection 3.1.4 for prevention and mitigation of accidents, and Subsection 3.1.6 for defence in depth. Refer to PSAR Section 3.5 to 3.9 for design characteristics applicable to REGDOC-2.5.2, Section 6.3. Refer to Chapter 4-12 for specific system design and safety functions that are applied to the DLs. Refer to the Fault Evaluation process as outlined in Section 15.2. Refer to Chapter 15, Appendix 15B for complementary design features.
6.4 Radiation Protection and acceptance criteria	Requirement	The BWRX-300 design complies with Section 6.4 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.2 Chapter 3, Subsection 3.1.6 Chapter 12, Section 12.1 Chapter 12, Section 12.2 Chapter 12, Section 12.3 Chapter 12, Section 12.5 Chapter 15, Section 15.1 Chapter 15, Section 15.3 Chapter 15, Section 15.5 Chapter 15, Section 15.6 Chapter 15, Section 15.7 Chapter 20, Section 20.8 Chapter 20, Section 20.9 Chapter 20, Section 20.11	Refer to Subsection 3.1.2 for the radiation protection and radiological acceptance criteria. Refer to Subsection 3.1.6 for defence in depth in design. The design provisions to ensure radiation exposures are ALARA are included in Section 12.3. Section 15.5 describes the dose calculation methodology used in the Deterministic Safety Analysis (DSA). Results of the analyses are summarized in Section 15.7 demonstrating that the radiological consequences of the analyzed events do not exceed the acceptance criteria for Anticipated Operational Occurrences (AOOs) listed in Chapter 15, Table 15.3-1 and for DBAs listed in Table 15.3- 2. There are no DBAs or AOOs that lead to fission product releases. Refer to Chapter 20 for environment impact of postulated accidents involving radioactive releases.

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6.5 Exclusion zone	Requirement	The BWRX-300 design complies with Section 6.5 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 2, Section 2.1 Chapter 2, Section 2.10 Chapter 19	Refer to Section 2.1 for information on DNNP exclusion zone, including security requirements and measures to minimize environmental impacts. Refer to Section 2.10 and Chapter 19 for emergency preparedness and response and accident management.
6.6 Facility layout	Requirement	The BWRX-300 design complies with Section 6.6 of REGDOC-2.5.2 V1 as written.	Chapter 1, Section 1.4 Chapter 1, Section 1.5 Chapter 1, Section 1.6 Chapter 1, Section 1.7 Chapter 2, Subsection 2.1.2 Chapter 3, Section 3.1.3 Safeguards Annex Security Annex	Refer to Chapter 1 for information on plant layout and design. Refer to Subsection 2.1.2 for the facility layout and exclusion zone. Refer to Subsection 3.1.3 for details on plant states considered in the design basis. Refer to the Safeguards Annex and Security Annex for interfaces between the safety, security and safeguards provisions of the NPP and other aspects of the facility layout.
6.6.1 Requirements for multiple units	Requirement	The BWRX-300 design complies with Section 6.6.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 1, Section 1.1 Chapter 3, Subsection 3.1.11	Refer to Subsection 3.1.11 for details. The intention is to deploy a single BWRX-300 unit at DNNP; however, the effect of common cause events impacting multiple units is considered in the hazard evaluation.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
7. General Design Requirements				
7.1 Safety classification of structures, systems and components	Requirement	The BWRX-300 design complies with the intent/safety objective of REGDOC-2.5.2 V1, Section 7.1 and guidance has been considered.	Chapter 3, Section 3.2 Chapter 3, Subsection 3.2.1 Chapter 3, Subsection 3.2.2 Chapter 3, Appendix 3A Chapter 3, Table 3.12-1 Chapter 7, Subsection 7.1.2 Chapter 15 Chapter 17, Subsection 17.3.1	SSC classification includes Safety Class 1, 2, 3 and Non-Safety Class as outlined in Section 3.2. BWRX-300 does not use "safety support system" terminology. Refer to the safety classification background in Subsection 3.2.1 and process in Subsection 3.2.2. Refer to Chapter 3, Appendix 3A, Table 3.12-1 for the preliminary BWRX-300 classification list. Refer to Subsection 7.1.2 for the I&C System Classification. Refer to Chapter 15 for supporting deterministic design basis analyses, complemented by probabilistic methods and engineering judgement, where appropriate. Refer to NEDC-33974P, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report for further information on applying an alternative approach for seismic design of radwaste SCCs (Seismic Category, RW-IIa).
7.2 Plant design envelope	Requirement	The BWRX-300 design complies with Section 7.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.3 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.4 Chapter 15, Section 15.5 Chapter 15, Section 15.7 Chapter 15, Appendix 15B Chapter 17, Subsection 17.3.1	Refer to Subsection 3.1.3 for details on plant states considered in the design basis. Refer to Figure 3.1-1 for the DLs as they correspond to plant states. Refer to Sections 15.1 through 15.5 for information to support the plant states. Refer to Chapter 15, Appendix 15B for complementary design features. Refer to Subsection 17.3.1.2 for GEH's Quality Management Plan and supporting plans for design management and control process.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
7.3 Plant states	Requirement	The BWRX-300 design complies with Section 7.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 1, Section 1.8 Chapter 3, Subsection 3.1.2 Chapter 3, Subsection 3.1.3 Chapter 3, Section 3.3 Chapter 3, Section 3.4 Chapter 3, Section 3.5 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.5 Chapter 15, Section 15.6 Chapter 15, Section 15.7 Chapter 13 Chapter 16	Refer to Section 1.8 for modes of normal operation of the plant. Refer to Section 3.3 to 3.9 for design criteria for structures, mechanical systems, electrical systems, I&C systems and equipment qualification for plant states. Refer to Subsection 3.1.2 for radiological acceptance criteria for AOOs and DBAs. Refer to Chapter 13 for plant procedures and guidelines. Refer to for Section 15.5 for the DSA and results in Section 15.7. Refer to Chapter 16 for the basis for the OLCs.
7.3.1 Normal Operation	Requirement	The BWRX-300 design complies with Section 7.3.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 1, Section 1.8 Chapter 3, Subsection 3.1.3 Chapter 7, Section 7.3 Chapter 15, Subsection 15.5.2 Chapter 15, Section 15.7 Chapter 13 Chapter 16, Section 16.4	Refer to Section 1.8 for modes of normal operation of the plant. Refer to Section 7.3 for constraints in I&C systems, and Chapter 16 for the basis for the OLCs. Refer to Chapter 13 for plant procedures and guidelines. Refer to Subsection 15.5.2 for analysis of normal operation.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
7.3.2 Anticipated operational occurrences	Requirement	The BWRX-300 design complies with Section 7.3.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.2 Chapter 4-12 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.4 Chapter 15, Section 15.5 Chapter 15, Section 15.7 Chapter 18	Refer to Subsection 3.1.2 for radiological acceptance criteria for AOOs and DBAs. Refer to Chapter 4-12 for specific system design and safety functions that are applied to the DLs that are used to mitigate AOOs. Refer to Chapter 6 for ESFs which mitigate the consequences of AOOs or postulated DBAs without any core damage. Refer to for Section 15.5 for the DSA and results in Section 15.7. Refer to Chapter 18 for Human Factors Engineering (HFE) considerations in design.
7.3.3 Design- basis accidents	Requirement	The BWRX-300 design complies with Section 7.3.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.2 Chapter 3, Subsection 3.1.3 Chapter 4-12 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.4 Chapter 15, Section 15.5 Chapter 15, Section 15.7 Chapter 18	Refer to Subsection 3.1.2 for radiological acceptance criteria for AOOs and DBAs. Refer to for Section 15.5 for the DSA and results in Section 15.7. Refer to Chapter 18 for HFE considerations in design.
7.3.4 Design extension conditions	Requirement	The BWRX-300 design complies with Section 7.3.4 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.3 Chapter 4-12 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.5 Chapter 15, Section 15.6 Chapter 15, Section 15.7 Chapter 15, Appendix 15A Chapter 15, Appendix 15B Chapter 19, Section 19.1	Refer to Chapter 4-12 for details on systems, where applicable. Refer to Chapter 9B, Subsection 9B.2.2 for the Bioshield. Refer to for Section 15.5 for the DSA for DECAs without core damage and results in Section 15.7. Control of combustible gases is outlined in Sections 15.5 and 15.6. Refer to Chapter 15, Appendix 15A for practical elimination of Beyond Design Basis Accident (BDBAs). Refer to Chapter 15, Appendix 15B for complementary design features.

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7.3.4.1 Severe accidents with design extension conditions	Requirement	The BWRX-300 design complies with Section 7.3.4.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.3 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.5 Chapter 15, Section 15.6 Chapter 15, Section 15.7 Chapter 15, Appendix 15A Chapter 15, Appendix 15B Chapter 13, Subsection 13.4.3 Chapter 17, Subsection 17.3.1 Chapter 19, Section 19.1	Refer to Subsection 15.1 for an overview of BDBAs with core damage, referred to as Severe Accidents (SAs). Refer to Section 15.6 for analysis of SAs. Refer to Chapter 13, Subsection 13.4.3 for procedures and guidelines for operating the plant during accidents. Refer to Subsection 17.3.1 for GEH's description of DA management. Refer to Chapter 19, Section 19.1 for establishment of Severe Accident Management Guideline (SAMG). A Probabilistic Safety Assessment Report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.
7.4 Postulated initiating events	Requirement	The BWRX-300 design complies with Section 7.4 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 2, Section 2.2 Chapter 3, Subsection 3.1.3 Chapter 3, Subsection 3.1.6 Chapter 3, Section 3.3 Chapter 3, Section 3.4 Chapter 3, Table 3.12-1 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.5	Refer to Chapter 2 for the evaluation of site-specific hazards. A Hazard Analysis Methodology will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667. Refer to Chapter 3, Appendix 3A, Table 3.12-1 for the preliminary BWRX-300 classification list. The Fault Evaluation process is described in Section 15.2. Refer to Section 15.2 for identification, categorization and grouping of PIEs. Refer to Section 3.3, Section 3.4 and Section 15.1 for the internal and external hazard evaluation.
7.4.1 Internal hazards	Requirement	The BWRX-300 design complies with Section 7.4.1 of REGDOC-2.5.2 V1 and guidance has been considered.	Chapter 3, Subsection 3.1.6 Chapter 3, Section 3.4 Chapter 9A, Section 9A.6 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.5 Chapter 15, Section 15.6	Refer to Section 3.4 for protection against internal hazards. Refer to Section 9A.6 for the fire hazard assessment. A Preliminary Fire Hazards Assessment Report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667. Refer to Section 15.1 for internal hazard evaluation. Refer to Section 15.2 for the Fault Evaluation process.

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7.4.2 External hazards	Requirement	The BWRX-300 design complies with Section 7.4.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 2, Section 2.2 Chapter 2, Section 2.4 Chapter 2, Section 2.5 Chapter 2, Section 2.6 Chapter 2, Section 2.7 Chapter 2, Section 2.8 Chapter 2, Section 2.9 Chapter 2, Section 2.11 Chapter 3, Subsection 3.1.6 Chapter 3, Section 3.3 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.5 Chapter 15, Section 15.6	Refer to Chapter 2 for site-specific hazard evaluation for external hazards. Refer to Section 2.11 for arrangements for the monitoring of site-related parameters. Refer to Section 3.3 for the evaluation of the impact of the site-related hazards. Refer to Section 15.1 for external hazard evaluation. Refer to Section 15.2 for the Fault Evaluation process. The Hazard Analysis Methodology will be submitted as per the LTC application plan outlined in NK054-CORR-00531- 10667.
7.4.3 Combination of events	Requirement	The BWRX-300 design complies with Section 7.4.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.6 Chapter 3, Section 3.3 Chapter 3, Section 3.4 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.5 Chapter 15, Section 15.6	Refer to Subsection 3.1.6 for defence in depth for internal and external hazard evaluations. Refer to Section 15.2 for the Fault Evaluation process.
7.5 Design rules and limits	Requirement	The BWRX-300 design complies with Section 7.5 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.7 Chapter 3, Subsection 3.2.2 Chapter 3, Section 3.3 Chapter 3, Section 3.4 Chapter 3, Section 3.5 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 3, Section 3.10 Chapter 4-12 Chapter 17, Subsection 17.2.1 Chapter 17, Subsection 17.3.1	Refer to Subsection 3.2.2 for the safety classification process. Refer to Sections 3.3 to 3.9 for design criteria for structures, mechanical systems, electrical systems, I&C systems and equipment qualification. Refer to Section 3.10 for in-service monitoring, tests, maintenance and inspections. Chapter 4-12 describe the SSCs. Refer to Chapter 17 for OPG's quality management system that applies to design.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
7.6 Design for reliability	Requirement	The BWRX-300 design complies with Section 7.6 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.2 Chapter 3, Subsection 3.1.7 Chapter 3, Section 3.6 Chapter 3, Subsection 3.7.2 Chapter 4-8 Chapter 12 Chapter 13, Subsection 13.3.2	Subsection 3.7.2 on design for reliability includes pointers to relevant sections in Chapter 7. Refer to Chapter 4-8, 11 and 12 for reliability requirements where applicable. Refer to Subsection 13.3.2 for Reliability Program under Fitness for Service. SSC classification includes Safety Class 1, 2, 3 and Non- Safety Class as outlined in Section 3.2. BWRX-300 does not use "safety support system" terminology.
7.6.1 Common- cause failure	Requirement	The BWRX-300 design complies with Section 7.6.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.6 Chapter 3, Subsection 3.1.7 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 7 Chapter 8, Subsection 8.2.2 Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.5	Refer to Chapter 3 for overview of CCF and its application in Chapter 7. Refer to Section 7.4 for testing. SSC classification includes Safety Class 1, 2, 3 and Non- Safety Class as outlined in Section 3.2. BWRX-300 does not use "safety support system" terminology.
7.6.1.1 Separation	Requirement	The BWRX-300 design complies with Section 7.6.1.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.7 Chapter 3, Section 3.2 Chapter 3, Section 3.3 Chapter 3, Section 3.4 Chapter 3, Section 3.5 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 4-12	Refer to Sections 3.5 to 3.9 for separation in general design criteria, where applicable. Refer to Chapter 4-12 for details on how physical separation is achieved in SSCs, as applicable. Chapter 7, Subsections 7.3.1.3.3, 7.3.2.3.3, and 7.3.3.3.3, for example, contain information on separation for the BWRX-300 I&C design. SSC classification includes Safety Class 1, 2, 3 and Non- Safety Class as outlined in Section 3.2. BWRX-300 does not use "safety support system" terminology.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
7.6.1.2 Diversity	Requirement	The BWRX-300 design complies with Section 7.6.1.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.1.7 Chapter 3, Section 3.2 Chapter 3, Section 3.5 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 4-12	Refer to Sections 3.5 to 3.9 for diversity in general design criteria, where applicable. Refer to Subsection 3.8.3 for diversity in electrical systems and components. Refer to Chapter 4-12 for details on how diversity is achieved in SSCs, as applicable. Chapter 7, for example, Subsections 7.3.1.3.3, 7.3.2.3.3, and 7.3.3.3.3 contain information on diversity for the BWRX-300 I&C design.
7.6.1.3 Independence	Requirement	The BWRX-300 design complies with Section 7.6.1.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.1.7 Chapter 3, Section 3.2 Chapter 3, Section 3.5 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 4-12	Refer to Sections 3.5 to 3.9 for independence in general design criteria, where applicable. Refer to Subsection 3.8.2 for independence in electrical systems and components. Refer to Chapter 4-12 for details on how independence is achieved in SSCs, as applicable. Chapter 7, Subsections 7.1.1, 7.3.1.2, 7.3.1.3.3, 7.3.2.2, 7.3.2.3.3, 7.3.3.2, and 7.3.3.3.3 contain information on independence for the BWRX-300 I&C design.
7.6.2 Single-failure criterion	Requirement	The BWRX-300 design complies with Section 7.6.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.6 Chapter 3, Subsection 3.1.7 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 4-12 Chapter 15, Section 15.2 Chapter 16	Refer to Subsection 3.1.7 for single-failure criterion. Refer to Section 3.6 and 3.7 for application of the single failure criterion in general design criteria, where applicable. Refer to Chapter 4-12 for single failure in SSCs, as applicable. Chapter 7, Subsections 7.1.1, 7.3.1.2, 7.3.1.3.3, 7.3.2.2, 7.3.2.3.3, 7.3.3.2, and 7.3.3.3.3, for example, contain information on single failure for the BWRX-300 I&C design. Refer to Chapter 16 for the basis for OLC requirements.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
7.6.3 Fail-safe design	Requirement	The BWRX-300 design complies with Section 7.6.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.7 Chapter 4-12	Refer to Subsection 3.1.7 for details on fail-safe design. Refer to Chapter 4-12 for fail-safe design in SSCs, as applicable. Chapter 7, for example, subsections 7.1.1, 7.3.1.2, 7.3.1.3.3, 7.3.2.2, 7.3.2.3.3, 7.3.3.2, and 7.3.3.3.3 contain information on fail-safe design for the BWRX-300 I&C systems. Some functions, ex. Diverse Protection System (DPS), are not fail-safe.
7.6.4 Allowance for equipment outages	Requirement	The BWRX-300 design complies with Section 7.6.4 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 4-12 Chapter 13, Section 13.3 Chapter 16, Section 16.4	Refer to Chapter 4-12 for details per SSC, as applicable. Chapter 7, for example, Subsections 7.3.1.2, 7.3.1.3.2, 7.3.2.3.2, 7.3.3.3.2, and 7.3.4.3.2 contain information on equipment outages for the BWRX-300 I&C systems. Refer to Subsection 13.3.3 for Maintenance, Surveillance, Inspection and Testing. Refer to Subsection 13.3.9 for Outages.
7.6.5 Shared systems	Requirement	The BWRX-300 design complies with Section 7.6.5 of REGDOC-2.5.2 V1 as written.	Chapter 3, Subsection 3.1.6 Chapter 3, Section 3.2 Chapter 4-12 Chapter 7, Subsection 7.3.3	Refer to Chapter 4-12 for details, as applicable. Refer to subsection 7.3.3.2 for I&C details. Refer to Subsection 7.3.3.3.5 for a description on 72-hour battery backup power supply.
7.6.5.1 Shared instrumentation for safety systems	Requirement	The BWRX-300 design complies with Section 7.6.5.1 of REGDOC-2.5.2 V1 as written.	Chapter 4-12 Chapter 7, Subsection 7.3.3	Refer to Chapter 4-12 for details, as applicable. Refer to subsection 7.3.3.2 for I&C details.
7.6.5.2 Sharing of SSCs between reactors	Requirement	The BWRX-300 design complies with Section 7.6.5.2 of REGDOC-2.5.2 V1 as written.	Chapter 1, Section 1.1 Chapter 3, Subsection 3.1.11	Refer to Subsection 3.1.11 for details. The intention is to deploy a single BWRX-300 unit at DNNP.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
7.7 Pressure-retaining structures, systems and components	Requirement	The BWRX-300 design complies with Section 7.7 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.2 Chapter 3, Section 3.3 Chapter 3, Section 3.4 Chapter 3, Section 3.5 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 5, Section 5.7 Chapter 6, Section 6.3 Chapter 15, Section 15.7 Chapter 15, Appendix 15B	Refer to Sections 3.5 to 3.9 for design criteria for civil structures, mechanical SSCs, I&C, electrical systems and components, and environmental qualification, as applicable. Refer to Section 3.6 for mechanical systems and components. Refer to Subsection 3.4.4 for pipe breaks. Refer to Subsection 3.5.1 for general design principles for Seismic Category A structures. Refer to Section 5.7 for the Reactor Pressure Control (RPC) System and Section 6.3 for the containment and associated systems. Refer to Appendix 15B, Table 15B-1 for complementary design features mitigating functions which include over-pressure protection.
7.8 Equipment environmental qualification	Requirement	The BWRX-300 design complies with Section 7.8 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.4 Chapter 3, Section 3.9 Chapter 3, Subsection 3.9.4 Chapter 4-12 Chapter 13, Section 13.3	Refer to Section 3.4 on barriers for protection against internal hazards. Refer to Subsection 3.9.4 on details of environmental qualification. Refer to Chapter 4-12 for relevant standards for qualifying equipment in different environments, where applicable. Chapter 7, for example, Subsections 7.3.1.3.1 & 7.3.2.3.1 & 7.3.3.1 contain information on environmental qualification for the BWRX-300 I&C systems.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
7.9 Instrumentation and control				
7.9.1 General	Requirement	The BWRX-300 design complies with Section 7.9.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.7 Chapter 7, Subsection 7.1.1 Chapter 7, Subsection 7.3.1 Chapter 7, Subsection 7.3.2 Chapter 7, Subsection 7.3.3 Chapter 7, Section 7.4 Chapter 18, Section 18.3	Refer to Chapter 7 for detailed information on I&C architecture design. Refer to Subsection 7.3.1.2, 7.3.1.3.2, 7.3.1.3.3, 7.3.1.4, 7.3.2.2, 7.3.2.3.2, 7.3.2.3.3, and 7.3.3.4 for specific details. Refer to Section 18.3 for design of human system interface.
7.9.2 Use of computer-based systems or equipment	Requirement	The BWRX-300 design complies with Section 7.9.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.7 Chapter 3, Section 3.9 Chapter 3, Appendix 3C Chapter 7, Section 7.1 Chapter 7, Section 7.3 Chapter 7, Section 7.4	Refer to Section 7.1 and 7.4 for summary of use of the International Electrotechnical Commission (IEC) standards and the I&C system design assurance plan (with illustration of verification and validation activities). Refer to Chapter 3, Appendix 3C for computer programs used in the design of components, equipment and structures. Refer to Table 7.3-5 for I&C Compliance alignment.
7.9.3 Accident monitoring instrumentation	Requirement	The BWRX-300 design complies with Section 7.9.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 7, Subsection 7.1.1 Chapter 7, Subsection 7.3.1 Chapter 7, Subsection 7.3.3 Chapter 19, Section 19.1 Chapter 19, Section 19.3	Refer to Subsection 7.3.1.2, 7.3.1.4 and 7.3.3.4 for I&C accident monitoring. The associated facilities are described in Chapter 19.
7.10 Safety support system	Requirement	The BWRX-300 design complies with Section 7.10 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 6, Section 6.2 Chapter 6, Section 6.3 Chapter 4-12	Refer to system-specific information in Chapter 4-12 of the PSAR. Refer to Chapter 6, for example, for ESFs and Section 9A.5 for facility ventilation. SSC classification includes Safety Class 1, 2, 3 and Non-Safety Class as outlined in Section 3.2. BWRX-300 does not use "safety support system" terminology.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
7.11 Guaranteed shutdown state	Requirement	The BWRX-300 design complies with Section 7.11 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 4, Section 4.5 Chapter 4, Section 4.6 Chapter 13, Section 13.3	Refer to Subsection 4.6.2 for details of the Control Rod Drive (CRD) system. Refer to Subsection 13.3.9 on the Conduct of Operations covers Outages.
7.12 Fire Safety	Requirement	The BWRX-300 design complies with Section 7.12 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.4 Chapter 9A, Section 9A.6 Chapter 4-12	Refer to Section 9A.6 for fire protection systems. Refer to Chapter 4-12 for details, as applicable. A Preliminary Fire Hazards Assessment Report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.
7.12.1 General	Requirement	The BWRX-300 design complies with Section 7.12.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.4 Chapter 3, Section 3.6 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 9A, Section 9A.6 Chapter 13, Section 13.4 Chapter 19, Section 19.1 Chapter 19, Section 19.5	Refer to Chapter 3 for applicability to general design criteria. Refer to Figure 9A.6.10-1 for BWRX-300 Fire Hazard Assessment methodology. Refer to Subsection 9A.6.6 for fire protection system and equipment operation. Refer to Section 13.4 for fire protection program implementation and Section 19.5 for emergency preparedness for internal hazards. A Preliminary Fire Hazards Assessment Report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.
7.12.2 Safety to life	Requirement	The BWRX-300 design complies with Section 7.12.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 9A, Section 9A.5 Chapter 9A, Section 9A.6 Chapter 13, Section 13.4 Chapter 18, Section 18.3	Refer to Figure 9A.6.10-2 for BWRX-300 Fire Safe Shutdown Analysis Methodology. Refer to Section 9A.5 for facility ventilation. Refer to Section 13.4 for fire protection program implementation. Refer to Section 18.3 for training program development, including fire safety. A Preliminary Fire Safe Shutdown Analysis Report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.

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7.12.3 Environmental protection and nuclear safety	Requirement	The BWRX-300 design complies with Section 7.12.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 9A, Section 9A.6 Chapter 11, Section 11.2 Chapter 11, Section 11.3 Chapter 11, Section 11.5 Chapter 12, Section 12.3 Chapter 20, Section 20.6 Chapter 20, Section 20.8 Chapter 20, Section 20.9	Refer to Section 11.2 and 11.3 for systems for management of liquid and gaseous radioactive waste. Refer to Section 20.9 for environmental impact of postulated accidents involving radioactive releases.
7.13 Seismic qualification and design	Requirement	The BWRX-300 design complies with Section 7.13 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.2 Chapter 3, Subsection 3.2.3 Chapter 3, Subsection 3.3.1 Chapter 3, Subsection 3.9.3 Chapter 3, Table 3.3-1 Chapter 9B	Refer to Subsection 3.2.3 for Seismic Categories. Refer to Subsection 3.3.1.2 for seismic analysis of Seismic Category A structures. Refer to Subsection 3.3.1.3 for seismic analysis of Seismic Category A and B subsystems. Refer to Subsection 3.3.1.5 for seismic instrumentation monitoring. Refer to Subsection 3.9.3 for seismic design considerations in environmental qualification. Refer to Chapter 3, Table 3.3-1 for structures and buildings. Refer to Chapter 9B for civil engineering works and structures.
7.13.1 Seismic design and classification	Requirement	The BWRX-300 design complies with the intent/safety objective of REGDOC-2.5.2 V1, Section 7.13.1 and guidance has been considered.	Chapter 3, Section 3.2 Chapter 3, Subsection 3.2.3 Chapter 3, Subsection 3.3.1 Chapter 3, Section 3.5 Chapter 3, Table 3.3-1 Chapter 9B	Refer to Subsection 3.5.3.6 for testing and in-service requirements. Refer to Subsection 3.5.6 for robustness. Refer to Chapter 3, Table 3.3-1 for structures and buildings. Refer to 9B, Section 9B.1 to 9B.3 for detailed information. Refer to NEDC-33974P, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report for further information on applying an alternative approach for seismic design of radwaste SCCs (Seismic Category, RW-IIa).

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7.14 In-service testing, maintenance, repair, inspection and monitoring	Requirement	The BWRX-300 design complies with Section 7.14 of REGDOC-2.5.2 V1, including Guidance, as written.	Chapter 3, Section 3.2 Chapter 3, Table 3.12-1 Chapter 4-12 Chapter 13, Section 13.3	SSC classification includes Safety Class 1, 2, 3 and Non-Safety Class as outlined in Section 3.2. BWRX-300 does not use "important to safety" terminology as a classification category. Refer to Chapter 3, Appendix 3A, Table 3.12-1 for the preliminary BWRX-300 classification list. Refer to Chapter 4-12 for details per SSC, as applicable.

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7.15 Civil structure				
7.15.1 Design	Requirement	The BWRX-300 design complies with Section 7.15.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.3 Chapter 3, Section 3.4 Chapter 3, Section 3.5 Chapter 3, Table 3.3-1 Chapter 6, Section 6.3 Chapter 9B	Refer to Chapter 3 for the methodology used for civil structures; Chapter 9B applies to the approach and presents results. Buildings and structures are referred to by their Seismic Category and are not classified as "important to safety".
7.15.2 Surveillance	Requirement	The BWRX-300 design complies with Section 7.15.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.5 Chapter 6, Section 6.3 Chapter 13, Section 13.3 Chapter 9B	Refer to Chapter 3 for the methodology used for civil structures; Chapter 9B applies to the approach and presents results. Refer to Subsection 6.3.7 for containment leakage testing. Refer to Section 13.3 for implementation of operational safety programs, including maintenance, surveillance, inspection and testing. Buildings and structures are referred to by their Seismic Category and are not classified as "important to safety".
7.15.3 Lifting and handling of large loads	Requirement	The BWRX-300 design complies with Section 7.15.3 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.4 Chapter 3, Section 3.5 Chapter 9A, Section 9A.8	Refer to Section 3.4 and 3.5 for hard object impact and impactive loads. Refer to Section 9A.8 for overhead lifting equipment.
7.16 Construction and commissioning	Requirement	The BWRX-300 design complies with Section 7.16 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 14, Section 14.1 Chapter 17, Subsection 17.3.1	Refer to Section 14.1 for management oversight of construction phase. Refer to Subsection 17.3.1 for facility development lifecycle phases and activities, including construction and commissioning.

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7.17 Aging and wear	Requirement	The BWRX-300 design complies with Section 7.17 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.7 Chapter 3, Section 3.5 Chapter 3, Section 3.6 Chapter 3, Section 3.7 Chapter 3, Section 3.8 Chapter 3, Section 3.9 Chapter 3, Section 3.10 Chapter 4-12 Chapter 13, Section 13.3 Chapter 17, Subsection 17.3.2	Refer to Sections 3.3 to 3.9 for design criteria for structures, mechanical systems, electrical systems, I&C systems and equipment qualification. Refer to Section 3.10 for in-service monitoring, tests, maintenance and inspections. Chapter 4-12 describe the SSCs. Refer to Section 13.3 for Fitness for Service programs, which include aging management.
7.18 Control of foreign material	Requirement	The BWRX-300 design complies with Section 7.18 of REGDOC-2.5.2 V1 as written.	Chapter 4-7 Chapter 4, Subsection 4.2.2 Chapter 10, Subsection 10.2.3 Chapter 10, Subsection 10.3.2 Chapter 13, Subsection 13.3.3 Chapter 14, Subsection 14.2	Refer to Chapter 4-7 for measures to prevent corrosion of materials, where applicable. Refer to Chapter 4, Subsection 4.2.2 for the Debris Filter Lower Tie Plate. Refer to Subsection 10.3.2 for information on the final feedwater strainer. Refer to Chapter 13 for preventative measures OPG has established, including provisions for foreign material exclusion. Refer to Chapter 14 for foreign material exclusion planning in preparation for construction and commissioning.
7.19 Transport and packaging for fuel and radioactive waste	Requirement	The BWRX-300 design complies with Section 7.19 of REGDOC-2.5.2 V1 as written.	Chapter 9A, Section 9A.1 Chapter 9A, Section 9A.8 Chapter 13, Section 13.3 Chapter 21, Section 21.4 Chapter 21, Section 21.5 Safeguards Annex, Section 5.0	Refer to Section 9A.1 for new fuel and fuel storage and handling. Refer to 9A.8 for overhead lifting equipment. Refer to Safeguards Annex for storage of nuclear fuel. Refer to Chapter 21 for information on radioactive waste. PSAR sections listed are for on-site handling.

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7.20 Escape routes and means of communication	Requirement	The BWRX-300 design complies with Section 7.20 of REGDOC-2.5.2 V1 as written.	Chapter 9A, Subsection 9A.9.1 Chapter 9A, Subsection 9A.9.2 Chapter 13, Subsection 13.4.3 Chapter 19, Section 19.2	Refer to Subsection 9A.9.1 for the communication systems. Refer to Subsection 9A.9.2 for lighting and emergency lighting systems. Refer to Subsection 13.4.3 for procedures and guidelines for operating the plant during accidents. Refer to Section 19.2 for means of communication with emergency support facilities.
7.21 Human factors	Requirement	The BWRX-300 design complies with Section 7.21 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 15, Section 15.4 Chapter 18, Section 18.1 Chapter 18, Section 18.3 Chapter 18, Section 18.4 Chapter 18, Section 18.5	Refer to Section 15.4 for human actions in the DSA and PSA. Refer to Section 18.1 for overview of the Human Factors Engineering Program Plan (HFEPP). A Human Factors Engineering Program Plan will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667. Refer to Section 18.4 for human factors verification and validation.
7.22 Robustness against malevolent acts	Requirement	The BWRX-300 design complies with Section 7.22 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.5.6 Security Annex	Refer to Subsection 3.5.6.2 for design for malevolent acts. Please refer to Security Annex for further information.
7.22.1 Design principles	Requirement	The BWRX-300 design complies with Section 7.22.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.3.7 Chapter 3, Subsection 3.5.6 Security Annex	Refer to Subsection 3.3.7.4 for robustness against malevolent acts. Refer to Subsection 3.5.6.2 for design for malevolent acts. Please refer to Security Annex for further information.
7.22.2 Design methods	Requirement	The BWRX-300 design complies with Section 7.22.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.3.7 Chapter 3, Subsection 3.5.6 Security Annex	Refer to Subsection 3.3.7.4 for robustness against malevolent acts. Refer to Subsection 3.5.6.2 for design for malevolent acts. Please refer to Security Annex for further information.

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7.22.3 Acceptance criteria	Requirement	The BWRX-300 design complies with Section 7.22.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.3.7 Chapter 3, Subsection 3.5.6 Chapter 15, Section 15.2 Security Annex	Refer to Subsection 3.3.7.4 for robustness against malevolent acts. Refer to Subsection 3.5.6.2 for design for malevolent acts. Refer to Section 15.3 for acceptance criteria. The Fault Evaluation process is described in Section 15.2. Malevolent acts are addressed in the Security Annex.
7.22.4 Cyber security	Requirement	The BWRX-300 design complies with Section 7.22.4 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 7, Subsection 7.4.4 Chapter 13, Section 13.5 Security Annex	Refer to Subsection 7.4.4 for the I&C cyber security lifecycle. Refer to Section 13.5 for nuclear safety and nuclear security interfaces, which include cyber security. Additional information can be found in the Security Annex.
7.23 Safeguards	Requirement	The BWRX-300 design complies with Section 7.23 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Safeguards Annex, Section 1.2 Safeguards Annex, Section 6.0 Safeguards Annex, Section 7.0	Refer to the Safeguards Annex for details on compliance with safeguards and non-proliferation requirements to which Canada has agreed.
7.24 Decommissioning	Requirement	The BWRX-300 design complies with Section 7.24 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 12, Section 12.3 Chapter 17, Subsection 17.2.3 Chapter 21, Section 21.2 Chapter 21, Section 21.3 Chapter 21, Section 21.4 Chapter 21, Section 21.5	Refer to Subsection 17.2.3 for OPG's facility end-of-life lifecycle phases and activities. Refer to Section 12.3 for incorporation of decommissioning in the BWRX-300 design.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
8. System-Specific Requirements				
8.1 Reactor core	Requirement	The BWRX-300 design complies with Section 8.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.1.6 Chapter 3, Section 3.6 Chapter 4, Section 4.1 Chapter 4, Section 4.3 Chapter 4, Subsection 4.4.5 Chapter 4, Section 4.5 Chapter 4, Section 4.6 Chapter 6, Section 6.2 Chapter 6, Section 6.3 Chapter 7 Chapter 15	Refer to Section 3.6 for general design aspects and Section 4.3 for reactor core design. Refer to Subsection 4.4.5 on flow stability evaluation. Refer to Section 4.5 for details on the CRD. Refer to Subsection 4.6.2 for details of the CRD system. Refer to Chapter 15 for the supporting analyses to steady state.
8.1.1 Fuel elements, assemblies and design	Requirement	The BWRX-300 design complies with Section 8.1.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 4, Section 4.2 Chapter 4, Section 4.3 Chapter 4, Section 4.4 Chapter 4, Subsection 4.4.5 Chapter 7, Section 7.3 Chapter 13, Section 13.3 Chapter 15, Section 15.5 Chapter 21, Section 21.4	Refer to Section 4.2 for fuel design, including fuel assembly. Refer to Subsection 4.4.5 for flow stability evaluation. Refer to Section 7.3 for the stability monitoring function to detect and suppress instabilities should they occur. Refer to Section 15.5 for analysis of DBAs. Refer to Section 21.4 for considerations for long-term irradiated fuel storage.
8.1.2 Control systems	Requirement	The BWRX-300 design complies with Section 8.1.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 4, Section 4.2 Chapter 4, Section 4.5 Chapter 4, Section 4.6	Refer to Section 4.6 for design of reactivity control systems.
8.2 Reactor coolant system	Requirement	The BWRX-300 design complies with Section 8.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 5, Subsection 5.1.1 Chapter 5, Section 5.2 Chapter 5, Section 5.5 Chapter 5, Section 5.7 Chapter 5, Section 5.11	Currently, there are no pressure relief valves for the BWRX-300. There are no moving components as applicable to REGDOC-2.5.2, Section 8.2. Reactor coolant flow through the BWRX-300 core is by natural circulation (no reactor coolant pumps in BWRX-300 design) as outlined in Section 5.5.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
8.2.1 In-service pressure boundary inspection	Requirement	The BWRX-300 design complies with Section 8.2.1 of REGDOC-2.5.2 V1 as written.	Chapter 5, Section 5.7 Chapter 5, Section 5.11	Refer to Section 5.11 for in-service inspection and surveillance testing.
8.2.2 Reactor coolant system inventory	Requirement	The BWRX-300 design complies with Section 8.2.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 5, Section 5.1 Chapter 5, Section 5.10 Chapter 5, Section 5.11 Chapter 5, Section 5.12 Chapter 15, Section 15.3 Chapter 15, Section 15.5	Refer to Section 5.10 for estimating core coolant inventory. Refer to Section 5.12 for chemical and inventory control systems for the reactor coolant.
8.2.3 Reactor coolant system cleanup	Requirement	The BWRX-300 design complies with Section 8.2.3 of REGDOC-2.5.2 V1 as written.	Chapter 5, Section 5.11 Chapter 5, Section 5.12 Chapter 9A, Section 9A.2.2	Refer to Section 5.11 for water chemistry and monitoring. Refer to Section 5.12 and Section 9A.2.2 for the Reactor Water Cleanup System (CUW).
8.2.4 Removal of residual heat from reactor core	Requirement	The BWRX-300 design complies with Section 8.2.4 of REGDOC-2.5.2 V1 as written.	Chapter 6, Section 6.2 Chapter 5, Section 5.3 Chapter 9A, Section 9A.2.3	Refer to Chapter 6 for information on the Isolation Condenser System (ICS). Refer to Section 5.3 for nuclear boiler system shutdown modes. Refer to Chapter 9A for a description of the Shutdown Cooling System (SDC).

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
8.3 Steam supply system				
8.3.1 Steam lines	Requirement	The BWRX-300 design complies with Section 8.3.1 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.6 Chapter 5, Section 5.1 Chapter 5, Section 5.3 Chapter 5, Section 5.10 Chapter 5, Section 5.11 Chapter 10, Section 10.4	Refer to Section 3.6 for Main Steam Containment Isolation Valve (MSCIV) and Main Steam Reactor Isolation Valve (MSRIV) design criteria. Refer to Section 10.4 for main steam system.
8.3.2 Steam and feedwater system piping and vessel	Requirement	The BWRX-300 design complies with Section 8.3.2 of REGDOC-2.5.2 V1 as written.	Chapter 3, Section 3.7 Chapter 7, Section 7.3 Chapter 8, Section 8.5 Chapter 10, Section 10.3	Refer to Section 7.3 for I&C separation. Refer to Section 8.5 for cable separation. Refer to Section 10.3 for feedwater systems. Auxiliary feedwater is not applicable to the BWRX-300 design.
8.3.3 Turbine generators	Requirement	The BWRX-300 design complies with Section 8.3.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.4 Chapter 10, Section 10.2 Chapter 10, Section 10.9 Chapter 15, Section 15.6	Refer to Section 3.4 for sources of internal hazard. Refer to section 10.2 for turbine generator system. Refer to Subsection 10.2.4 for turbine missile probability analysis. Refer to Section 15.6 which presents the methodology for performing a qualitative assessment for determining the probability of turbine generated missiles and associated results.
8.4 Means of shutdown	Requirement	The BWRX-300 design complies with the intent/safety objective of REGDOC-2.5.2 V1, Section 8.4 and guidance has been considered.	Chapter 4, Section 4.2 Chapter 4, Section 4.3 Chapter 4, Section 4.5 Chapter 4, Section 4.6 Chapter 7, Section 7.3 Chapter 7, Subsection 7.3.3.2 Chapter 15	Refer to Subsection 4.6.2 for details on means of shutdown with the CRD. Refer to NEDC-33974P, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report for further information on applying an alternative approach for means of shutdown independence.
8.4.1 Reactor trip parameters	Requirement	The BWRX-300 design complies with Section 8.4.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 7, Section 7.3 Chapter 15, Section 15.5 Chapter 15, Table 15.5-5 Chapter 16	Refer to Section 15.5 for Deterministic Safety Analysis. Refer to Chapter 5, Table 15.5-5 for DL inputs used in non-Loss-of-Coolant Accident (LOCA) analyses.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
8.4.2 Reliability	Requirement	The BWRX-300 design complies with Section 8.4.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Subsection 3.1.6 Chapter 3, Subsection 3.7.2 Chapter 3, Section 3.10 Chapter 4-12 Chapter 13, Subsection 13.3.2	Refer to Chapter 4-12 for details on system reliability, as applicable. The I&C system presents reliability information, as an example, in Chapter 7.
8.4.3 Monitoring and operator action	Requirement	The BWRX-300 design complies with Section 8.4.3 of REGDOC-2.5.2 V1 as written.	Chapter 4, Section 4.6 Chapter 7, Section 7.3 Chapter 7, Section 7.5 Chapter 7, Section 7.6 Chapter 18, Section 18.3	Refer to Chapter 7 for I&C. Refer to Section 18.3 for design of the MCR and SCR.
8.5 Emergency core cooling system	Requirement	The BWRX-300 design complies with Section 8.5 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 6, Section 6.2 Chapter 6, Subsection 6.2.1 Chapter 6, Subsection 6.2.2 Chapter 6, Subsection 6.2.3 Chapter 6, Section 6.3	Refer to Section 6.2 for ICS which functions as the Emergency Core Cooling System (ECCS).

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
8.6 Containment				
8.6.1 General	Requirement	The BWRX-300 design complies with Section 8.6.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 6, Section 6.3 Chapter 6, Subsection 6.3.2 Chapter 6, Subsection 6.3.3 Chapter 6, Subsection 6.3.4 Chapter 6, Subsection 6.3.5 Chapter 9B, Subsection 9B.2.1 Chapter 9B, Subsection 9B.2.2 Chapter 15, Appendix 15B	Refer to Subsection 6.3.4 for mechanical features of containment, including details on containment isolation. Refer to Chapter 15, Appendix 15B for complementary design features.
8.6.2 Strengthen of the containment structure	Requirement	The BWRX-300 design complies with Section 8.6.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.4 Chapter 3, Section 3.5 Chapter 3, Subsection 3.5.3 Chapter 3, Subsection 3.5.5 Chapter 3, Table 3.3-1 Chapter 6, Section 6.3 Chapter 6, Subsection 6.3.2 Chapter 9B, Subsection 9B.2.1	Refer to Chapter 3 for containment structure design and Chapter 9B for load combinations. Refer to Subsection 3.5.3 and Subsection 3.5.5 for containment seismic design criteria, which is summarized in Table 3.3-1. Refer to Subsection 6.3.2 for the Primary Containment System (PCS).
8.6.3 Capability for pressure test	Requirement	The BWRX-300 design complies with Section 8.6.3 of REGDOC-2.5.2 V1 as written.	Chapter 3, Subsection 3.5.3 Chapter 6, Section 6.3 Chapter 6, Subsection 6.3.7 Chapter 9B, Subsection 9B.2.1.8	Refer to Subsection 3.5.3 for testing and in-service inspection of the containment structure. Refer to Subsection 6.3.7 for containment leakage testing.
8.6.4 Leakage	Requirement	The BWRX-300 design complies with Section 8.6.4 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.5 Chapter 6, Section 6.3 Chapter 6, Subsection 6.3.7 Chapter 13, Section 13.3 Chapter 9B, Subsection 9B.2.1	Refer to Subsection 6.3.6 for containment leakage testing.
8.6.5 Containment penetrations	Requirement	The BWRX-300 design complies with Section 8.6.5 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.4 Chapter 6, Section 6.3 Chapter 6, Subsection 6.3.2 Chapter 6, Subsection 6.3.4 Chapter 9B, Subsection 9B.2.1	Refer to Subsection 6.3.2. for containment penetrations. Refer to Subsection 9B.2.1.3 for the containment structural description, which includes containment penetrations.

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8.6.6 Containment isolation	Requirement	The BWRX-300 design complies with the intent/safety objective of REGDOC-2.5.2 V1, Section 8.6.6.	Chapter 5 Chapter 6, Section 6.3 Chapter 6, Subsection 6.3.4	Chapter 5 addresses pressure boundary and isolation in the context of the RCPB. The containment isolation function of the PCS is outlined in Subsection 6.3.4. Refer to NEDC-33974P, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report for further information on applying an alternative approach for containment isolation.
8.6.7 Containment airlocks	Requirement	The BWRX-300 design complies with Section 8.6.7 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.5 Chapter 6, Section 6.3 Chapter 6, Subsection 6.3.2 Chapter 9B, Subsection 9B.2.1	Refer to Section 6.3.2 on equipment and personnel access.
8.6.8 Internal structures of the containment	Requirement	The BWRX-300 design complies with Section 8.6.8 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.4 Chapter 3, Section 3.5 Chapter 6, Section 6.3 Chapter 6, Subsection 6.3.2 Chapter 9A, Section 9A.4 Chapter 9A, Section 9A.5 Chapter 9B, Section 9B.2	Refer to Chapter 6 and 9 on the structural support function of the PCS. Refer to Section 9A.4 for air and gas system; refer to Section 9A.5 for ventilation.
8.6.9 Containment pressure and energy management	Requirement	The BWRX-300 design complies with Section 8.6.9 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 6, Section 6.3 Chapter 6, Subsection 6.3.3 Chapter 6, Subsection 6.3.5 Chapter 15, Appendix 15B	Refer to Subsection 6.3.3 for the Passive Containment Cooling System (PCCS) which maintains temperature and pressure of the PCS through passive cooling during accident conditions. Refer to Chapter 15, Appendix 15B for containment pressure control venting.
8.6.10 Control and cleanup of the containment atmosphere	Requirement	The BWRX-300 design complies with Section 8.6.10 of REGDOC-2.5.2 V1 as written.	Chapter 6, Section 6.2 Chapter 6, Section 6.3 Chapter 6, Subsection 6.3.2 Chapter 6, Subsection 6.3.4 Chapter 9A, Subsection 9A.4.2 Chapter 15, Section 15.7 Chapter 15, Appendix 15B	Refer to Subsection 9A.4.2 and Subsection 6.3.2 on the Containment Inerting System (CIS).

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8.6.11 Coverings, coatings and materials	Requirement	The BWRX-300 design complies with Section 8.6.11 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 6, Section 6.1 Chapter 6, Section 6.3 Chapter 9B, Subsection 9B.2.1	Refer to Chapter 6 for ESF material, including containment and associated systems. Refer to Chapter 9B, Subsection 9B.2.1.4 for the selection of containment design materials. Refer to Section 6.3 for further details.
8.6.12 Design extension conditions	Requirement	The BWRX-300 design complies with Section 8.6.12 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 6, Section 6.3 Chapter 15, Section 15.5 Chapter 15, Section 15.6 Chapter 15, Section 15.7 Chapter 15, Appendix 15A Chapter 15, Appendix 15B	The leak-tight containment boundary is outlined in Section 6.3. Refer to Section 15.7 for the results of the PSA and DSA. Refer to Chapter 15, Appendix 15A for practical elimination of BDBAs. Refer to Chapter 15, Appendix B for complementary design features for mitigating BDBAs.
8.7 Heat transfer to an ultimate heat sink	Requirement	The BWRX-300 design complies with Section 8.7 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 6, Section 6.2 Chapter 6, Subsection 6.2.3 Chapter 15, Section 15.2 Chapter 15, Section 15.5 Chapter 15, Section 15.6	Refer to Section 6.2 for details on the ICS pool arrangement that provides the ultimate heat sink for protecting the reactor core for any AOO or DBA. Refer to Section 15.2 for the Fault Evaluation process. Functions of the ICS satisfy the analysis assumptions for AOOs and DBAs as described in Chapter 15, Section 15.5. For BDBAs, refer to Chapter 15, Section 15.6.
8.8 Emergency heat removal system	Requirement	The BWRX-300 design complies with Section 8.8 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 6, Section 6.2 Chapter 15, Section 15.2 Chapter 15, Section 15.5 Chapter 15, Section 15.6	Refer to Section 6.2 for details on the ICS. Refer to Section 15.2 for the Fault Evaluation process. Functions of the ICS satisfy the analysis assumptions for AOOs and DBAs as described in Chapter 15, Section 15.5. For BDBAs, refer to Chapter 15, Section 15.6.
8.9 Electrical power systems	Requirement	The BWRX-300 design complies with Section 8.9 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.8 Chapter 8, Section 8.1 Chapter 8, Section 8.2 Chapter 8, Section 8.3 Chapter 8, Section 8.4 Chapter 8, Section 8.5	Refer to Section 3.8 for general design aspects for electrical systems and components. Refer to Subsection 8.1.1.2 for electrical power systems.

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8.9.1 Standby and emergency power systems	Requirement	The BWRX-300 design complies with Section 8.9.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.8 Chapter 8, Subsection 8.1.1 Chapter 8, Section 8.2 Chapter 8, Section 8.3 Chapter 8, Section 8.4 Chapter 9A, Section 9A.7	Refer to Section 3.8 for general design aspects for electrical systems and components. Refer to Subsection 8.1.1.3 and Section 8.4 for the emergency power supply system (batteries). Refer to Section 9A.7 for supporting systems for diesel generators.
8.9.2 DC and uninterruptible power systems	Requirement	The BWRX-300 design complies with Section 8.9.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.8 Chapter 8, Subsection 8.1.1 Chapter 8, Section 8.4 Chapter 8, Subsection 8.4.2	Refer to Section 3.8 for general design aspects for electrical systems and components. Refer to Subsection 8.1.1.4 and Subsection 8.4.2 for on- site power details.
8.9.3 Alternate AC power supply	Requirement	The BWRX-300 design complies with Section 8.9.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 3.8 Chapter 8, Subsection 8.1.1 Chapter 8, Section 8.4 Chapter 8, Subsection 8.4.2	Refer to Section 3.8 for general design aspects for electrical systems and components. Refer to Subsection 8.1.1.5 and Subsection 8.4.2 for on- site power details.

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8.10 Control Facilities				
8.10.1 Main control room	Requirement	The BWRX-300 design complies with Section 8.10.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 2.2 Chapter 3, Section 3.3 Chapter 3, Section 3.4 Chapter 6, Section 6.4 Chapter 7, Subsection 7.3.1 Chapter 7, Subsection 7.3.2 Chapter 7, Section 7.5 Chapter 8, Section 8.5 Chapter 9A, Section 9A.5 Chapter 9A, Section 9A.6 Chapter 9A, Section 9A.9 Chapter 12, Section 12.3 Chapter 12, Section 12.4 Chapter 18, Subsection 18.3.5	Refer to Section 6.4 for Control Room Habitability (CRH), which are served by a combination of individual systems in Chapter 9A and 12. Refer to Section 7.5 for and I&C in the Main Control Room (MCR). Refer to Subsection 18.3.5 for human factors design of the MCR.
8.10.1.1 Safety parameter display system	Requirement	The BWRX-300 design complies with Section 8.10.1.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 7, Subsection 7.3.3 Chapter 7, Section 7.7 Chapter 18, Section 18.3 Chapter 19, Section 19.2 Chapter 19, Section 19.3	Refer to Chapter 7 for I&C in the emergency response facilities. The I&C overview for the Safety Parameter Display System (SPDS) is outlined in Subsection 7.3.3.2. Refer to Section 19.2 for emergency response facilities.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
8.10.2 Secondary control room	Requirement	The BWRX-300 design complies with Section 8.10.1.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3, Section 2.2 Chapter 3, Section 3.3 Chapter 3, Section 3.4 Chapter 6, Section 6.4 Chapter 7, Subsection 7.3.1 Chapter 7, Subsection 7.3.2 Chapter 7, Section 7.6 Chapter 8, Section 8.5 Chapter 9A, Section 9A.5 Chapter 9A, Section 9A.6 Chapter 9A, Section 9A.9 Chapter 12, Section 12.3 Chapter 12, Section 12.4 Chapter 18, Subsection 18.3.6	Refer to Section 6.4 for CRH, which are served by a combination of individual systems in Chapter 9A and 12. Refer to Section 7.6 for and I&C in the Secondary Control Room (SCR). Refer to Subsection 18.3.6 for human factors design of the SCR.
8.10.3 Emergency support facilities	Requirement	The BWRX-300 design complies with Section 8.10.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 7, Subsection 7.3.3 Chapter 7, Section 7.7 Chapter 9A, Subsection 9A.9.1 Chapter 13, Section 13.3 Chapter 13, Section 13.4 Chapter 18, Section 18.2 Chapter 18, Section 18.3 Chapter 19, Section 19.2	Refer to Chapter 7 for I&C in the emergency response facilities. Refer to Subsection 9A.9.1 for communication systems, which include provisions to support emergency preparedness and response requirements. Refer to Subsection 13.4.3 for procedures and guidelines for operating the plant during accidents. Refer to Chapter 19 for emergency response facility details.
8.10.4 Credit for operator action	Requirement	The BWRX-300 design complies with Section 8.10.4 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 7, Subsection 7.3.1 Chapter 7, Subsection 7.3.2 Chapter 7, Subsection 7.3.3 Chapter 13, Section 13.3 Chapter 13, Section 13.4 Chapter 15, Section 15.1 Chapter 15, Section 15.4 Chapter 15, Section 15.5 Chapter 15, Section 15.6 Chapter 18, Section 18.3 Chapter 18, Section 18.4 Chapter 19, Section 19.3	Refer to Subsection 7.3.1.4, 7.3.2.4 and 7.3.3.4 for I&C operator interface. Refer to Subsection 13.4.2 for operating procedures. There are no operator actions credited in responding to the events analyzed in Section 15.5. Operator actions credited for PSA are discussed in Section 15.6. Refer to Chapter 18 for HFE design of the human system interface. Refer to Section 19.3 for goals of the accident management procedures and guidelines.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
8.11 Waste treatment and control	Requirement	The BWRX-300 design complies with the intent/safety objective of REGDOC-2.5.2 V1, Section 8.11.	Chapter 11, Section 11.2 Chapter 11, Section 11.3 Chapter 12, Section 12.1 Chapter 12, Section 12.2 Chapter 12, Section 12.3 Chapter 20, Section 20.6 Chapter 20, Section 20.8	Refer to Section 11.2 and 11.3 for management of liquid and gaseous radioactive waste, respectively. Refer to Section 12.1 for ALARA design considerations. Refer to NEDC-33974P, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report for further information on applying an alternative approach for seismic design of radwaste SCCs (Seismic Category, RW-IIa). Radioactive Waste Management Plan will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.
8.11.1 Control of liquid releases to the environment	Requirement	The BWRX-300 design complies with Section 8.11.1 of REGDOC-2.5.2 V1 as written.	Chapter 11, Section 11.2 Chapter 12, Section 12.1 Chapter 12, Section 12.2 Chapter 12, Section 12.3 Chapter 20, Section 20.6 Chapter 20, Section 20.8	Refer to Section 11.2 for management of liquid radioactive waste. Refer to Subsection 12.2.3 for airborne and liquid sources for environmental considerations.
8.11.2 Control of airborne material within the plant	Requirement	The BWRX-300 design complies with Section 8.11.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 9A, Section 9A.5 Chapter 11, Section 11.3 Chapter 12, Section 12.1 Chapter 12, Section 12.2 Chapter 12, Section 12.3 Chapter 20, Section 20.6 Chapter 20, Section 20.8	Refer to Section 9A.5 for facility ventilation. Refer to Section 11.3 for management of gaseous radioactive waste. Refer to Subsection 12.3.13 for radiation protection design features for ventilation. Refer to Subsection 12.2.3 for airborne and liquid sources for environmental considerations. Refer to Subsection 20.8.1 for airborne and effluent releases.
8.11.3 Control of gaseous releases to the environment	Requirement	The BWRX-300 design complies with Section 8.11.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 9A, Section 9A.5 Chapter 11, Section 11.3 Chapter 12, Section 12.1 Chapter 12, Section 12.2 Chapter 12, Section 12.3 Chapter 20, Section 20.6 Chapter 20, Section 20.8	Refer to Section 9A.5 for facility ventilation. Refer to Section 11.3 for management of gaseous radioactive waste. Refer to Subsection 12.3.13 for radiation protection design features for ventilation. Refer to Chapter 20 for plant features minimizing environmental impact.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
8.12 Fuel handling and storage	Requirement	The BWRX-300 design complies with Section 8.12 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 4, Subsection 4.2.5 Chapter 9A, Section 9A.1 Chapter 12, Section 12.3 Chapter 21, Section 21.5 Chapter 21, Section 21.7 Safeguards Annex, Section 5.0 Safeguards Annex, Section 7.0	Refer to Subsection 4.2.5.2, which includes inspection and testing for fuel system components and parts. Refer to Section 9A.1 for fuel handling and storage. Refer to Section 21.5 for long-term irradiated fuel storage. Fuel Design Reports and a Fuel Qualification Report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667. Refer to Section 5.0 and 7.0 for the flow and storage of nuclear fuel, and nuclear material accountancy, respectively.
8.12.1 Handling and storage of non-irradiated fuel	Requirement	The BWRX-300 design complies with Section 8.12.1 of REGDOC-2.5.2 V1 as written.	Chapter 4, Subsection 4.2.5 Chapter 9A, Section 9A.1 Safeguards Annex, Section 5.0 Safeguards Annex, Section 7.0	Refer to Subsection 4.2.5.2, which includes inspection and testing for fuel system components and parts. Refer to Section 9A.1 for handling and storage of non- irradiated fuel. Refer to the Subsection 5.1.2 of the Safeguards Annex for fresh fuel staging and storage. Fuel Design Reports and a Fuel Qualification Report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.
8.12.2 Handling and storage of irradiated fuel	Requirement	The BWRX-300 design complies with Section 8.12.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 4, Subsection 4.2.5 Chapter 9A, Section 9A.1 Chapter 9A, Subsection 9A.8.1 Safeguards Annex, Section 5.0 Safeguards Annex, Section 7.0	Refer to Subsection 4.2.5.2, which includes inspection and testing for fuel system components and parts. Refer to Section 9A.1 for fuel storage and handling, including the Fuel Pool Cooling and Cleanup System (FPC). Fuel Design Reports and a Fuel Qualification Report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
8.12.3 Detection of failed fuel	Requirement	The BWRX-300 design complies with Section 8.12.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 4, Section 4.6.8 Chapter 9A, Subsection 9A.1.2 Chapter 11, Section 11.3 Chapter 12, Section 12.3	Refer to Section 4.6.8 for core monitoring, which includes failed fuel prevention techniques. Refer to Subsection 9A.1.2.3.8 for overview of failed fuel considerations. Refer to the offgas system in Section 11.3 and associated radiation monitoring equipment in Section 12.3. Fuel Design Reports and a Fuel Qualification Report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.
8.13 Radiation protection	Requirement	The BWRX-300 design complies with Section 8.13 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 9A, Section 9A.5 Chapter 12, Section 12.3 Chapter 12, Section 12.4 Chapter 12, Section 12.6 Chapter 12, Section 12.7	Refer to Section 9A.5 for facility ventilation. Refer to Section 12.3 for design features for radiation protection. Refer to Section 12.4 for shielding design information and Section 12.7 for the radiation protection program.
8.13.1 Design for radiation protection	Requirement	The BWRX-300 design complies with Section 8.13.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 12, Section 12.3 Chapter 12, Section 12.4 Chapter 12, Section 12.6	Refer to 12.4 for shielding design information.
8.13.2 Access and movement control	Requirement	The BWRX-300 design complies with Section 8.13.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 12, Section 12.3 Chapter 12, Section 12.4 Chapter 12, Section 12.6 Chapter 12, Section 12.7	Refer to Section 12.6 for radiation zones and access requirements.
8.13.3 Radiation monitoring	Requirement	The BWRX-300 design complies with Section 8.13.3 of REGDOC-2.5.2 V1 as written.	Chapter 2, Section 2.9 Chapter 12, Section 12.3 Chapter 12, Subsection 12.3.14	Refer to Subsection 2.9.2 for radiation monitoring systems. Refer to Subsection 12.3.14 for instrumentation for radiation levels and airborne radioactivity.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
8.13.4 Sources of radiation	Requirement	The BWRX-300 design complies with Section 8.13.4 of REGDOC-2.5.2 V1 as written.	Chapter 12, Section 12.2 Chapter 12, Section 12.7	Refer to Section 12.2 for sources of radiation. Refer to Section 12.7 for radiation protection program.
8.13.5 Monitoring environmental impact	Requirement	The BWRX-300 design complies with Section 8.13.5 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 2, Section 2.9 Chapter 11, Section 11.5 Chapter 12, Section 12.7 Chapter 20, Section 20.6 Chapter 20, Section 20.8 Chapter 20, Section 20.11	Refer to Subsection 2.9.2 for radiation monitoring systems. Refer to Section 11.5 for process and effluent radiological monitoring. Refer to Section 12.7 for radiation protection program. Refer to Section 20.8 for control of radioactive discharges to the environment, and Section 20.11 for environmental measurements and monitoring programs.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
9. Safety Analysis				
9.1 General	Requirement	The BWRX-300 design complies with Section 9.1 of REGDOC-2.5.2 V1 as written.	Chapter 3 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.4 Chapter 15, Section 15.5 Chapter 15, Section 15.6 Chapter 15, Section 15.7	SSC classification includes Safety Class 1, 2, 3 and Non-Safety Class as outlined in Section 3.2. BWRX-300 does not use "important to safety" terminology as a classification category. Refer to Chapter 15 for safety analysis information in the PSAR to support the LTC. A subsequent safety analysis report will be submitted in support of the Licence to Operate application. Refer to Section 15.5 and 15.6 for DSA and PSA, respectively. Results are presented in Section 15.7. A Probabilistic Safety Assessment Methodology report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.
9.2 Analysis objectives	Requirement	The BWRX-300 design complies with Section 9.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 3 Chapter 4-12 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.4 Chapter 15, Section 15.5 Chapter 15, Section 15.6 Chapter 16, Section 15.7 Chapter 16, Section 16.4	SSC classification includes Safety Class 1, 2, 3 and Non-Safety Class as outlined in Section 3.2. BWRX-300 does not use "important to safety" terminology as a classification category. Refer to Chapter 15 for safety analysis information in the PSAR to support the LTC. A subsequent safety analysis report will be submitted in support of the Licence to Operate application. Refer to Section 15.5 and 15.6 for DSA and PSA, respectively. Results are presented in Section 15.7. A Probabilistic Safety Assessment Methodology report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.
9.3 Hazard analysis	Requirement	The BWRX-300 design complies with Section 9.3 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 15, Section 15.1 Chapter 15, Section 15.2 Chapter 15, Section 15.3 Chapter 15, Section 15.4 Chapter 15, Section 15.5	Refer to Sections 15.1 to 15.5 for hazard analysis. A Hazard Analysis Methodology and Hazard Analysis Results Reports will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
9.4 Deterministic safety analysis	Requirement	The BWRX-300 design complies with Section 9.4 of REGDOC-2.5.2 V1 as written.	Chapter 15, Section 15.5 Chapter 15, Section 15.7	Refer to Section 15.7 for results of the DSA.
9.5 Probabilistic safety assessment	Requirement	The BWRX-300 design complies with Section 9.5 of REGDOC-2.5.2 V1 as written.	Chapter 15, Section 15.6 Chapter 15, Section 15.7	Refer to Section 15.6 for the PSA. Refer to Subsection 15.7.11 for the results of the PSA. A Probabilistic Safety Assessment Methodology report will be submitted as per the LTC application plan outlined in NK054-CORR-00531-10667.

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REGDOC-2.5.2 V1 Section	Classification	Statement of Compliance	PSAR Reference	Compliance Notes or Comments (if applicable, including limitation on scope, interfacing information or clarification)
10. Environmental Protection and Mitigation				
10.1 Design for environmental protection	Requirement	The BWRX-300 design complies with Section 10.1 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 20, Section 20.2 Chapter 20, Section 20.5 Chapter 20, Section 20.6 Chapter 20, Section 20.11	Refer to Section 20.2 for the Environmental Impact Statement (EIS). Refer to Section 20.5 for site characteristics important to environmental impact. Refer to Section 20.6 for the best available technology and techniques economically available design features.
10.2 Release of nuclear and hazardous substances	Requirement	The BWRX-300 design complies with Section 10.2 of REGDOC-2.5.2 V1 as written and guidance has been considered.	Chapter 2, Subsection 2.8.2 Chapter 12, Section 12.3 Chapter 13, Section 13.3 Chapter 20, Section 20.4 Chapter 20, Section 20.6 Chapter 20, Section 20.8 Chapter 20, Section 20.10 Chapter 20, Section 20.11	Refer to Subsection 2.8.2 for hydrology considerations at the DNNP site. Refer to Section 12.3 for design features for radiation protection. Refer to Section 20.6 for best available technology and techniques economically available design features.
11. Alternative Approaches	Requirement	The BWRX-300 design complies with Section 11 of REGDOC-2.5.2 V1 by taking alternative approaches when required, meeting the intent/safety objective of the clause.	N/A	Alternative approaches are minimized in the BWRX-300 design. Where a design feature does not comply with a requirement in the applicable regulatory document as written, an assessment is performed. Refer to NEDC-33974P, BWRX-300 Darlington New Nuclear Project (DNNP) REGDOC-2.5.2 Alternative Approach Report for further information on alternative approaches.

3.0 REFERENCES

- 3-1 Government of Canada SOR/200-204, Class 1 Nuclear Facilities Regulations, June 2017.
- 3-2 Canadian Nuclear Safety Commission REGDOC-1.1.2, Licence Application Guide: Licence to Construct a Reactor Facility, Version 2, November 2021 Draft.
- 3-3 Canadian Nuclear Safety Commission REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants, Version 1, May 2014.
- 3-4 International Atomic Energy Agency, SSG-61, Format and Content of the Safety Analysis Report for Nuclear Power Plants, September 2021.
- 3-5 OPG Document NK054-PLAN-01210-00007, Darlington New Nuclear Project Licence to Construct Application Plan, May 2022.
- 3-6 GE Hitachi Nuclear Energy, NEDC-33974P, BWRX-300 DNNP REGDOC-2.5.2 Alternative Approach Report, September 2022.



HITACHI

GE Hitachi Nuclear Energy

NEDO-33974

Revision 2

May 2023

Non-Proprietary Information

**BWRX-300 Darlington New Nuclear
Project (DNNP)
REGDOC-2.5.2 Alternative Approach
Report**

NEDO-33974 Revision 2
Non-Proprietary Information

This is a non-proprietary version of the GE Hitachi Nuclear Energy (GEH) document NEDC-33974P Revision 2, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT
Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or, except as specified above, any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

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None.

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
0	All	Originally Issued as Proprietary Version NEDC-33974P Rev 0
1	All	Initial Issue as Non-Proprietary Version
2	Abstract, 1.1, 2.4, Ref. 21 and Ref. 22.	Included Abstract. Revised to incorporate recommendations from Independent Peer Review report (Ref. 21) related to means of shutdown.
2	2.2.2	Updated to reflect submission of LTC application.
2	2.2.3	Updated to reflect completed Phase 1 and Phase 2 Vendor Design Review.
2	2.5.1-2.7.1	Changed 2.6.1 to 2.5.1 and 2.7.1 to 2.6.1 to reflect deletion of the Means of Shutdown Independence as an alternative approach.
2	2.5.1	Added section heading and leading paragraph in Section 2.5.1. Modified CIS description in Section 2.5.1.3.2. for consistency with PSAR Section 6.3.4.3, Containment Isolation Valves.
2	Appendix	Added Appendix for details on deletion of Means of Shutdown Independence as an alternative approach.
2	All	Minor formatting updates throughout, including tables and references.
2	All	Initial Issue as Non-Proprietary Version NEDO-33974 Revision 2.

ABSTRACT

This report addresses alternative approaches for meeting the requirements of the Canadian Nuclear Safety Commission regulatory document for design of nuclear power plants as related to the BWRX-300. The report applies to the Ontario Power Generation (OPG) Darlington New Nuclear Project.

This report includes information on proposed alternative approaches that may be used in the design and justified in accordance with Section 11, Alternative Approaches, in REGDOC-2.5.2. Specifically, it includes information on proposed alternative approaches and summarizes the justification for acceptance of the alternative approaches. The summary descriptions are not intended to be detailed justifications for the alternative approaches because the technical basis for an alternative approach is part of the progression of the design process and details are documented in engineering reports and specifications. Accordingly, alternative approaches identified in this report are subject to change, and other alternative approaches may be identified in the design process. The summary descriptions address the items that are currently identified as alternative approaches that are being reviewed for implementation in the design process at the time of issuance of the report.

The BWRX-300 is a small modular reactor with passive safety features and is cooled using natural circulation. The design uses a risk-informed decision-making process for applying a graded approach to meet requirements and, where a graded approach relates to the alternatives, it is identified.

An initial goal for the BWRX-300 design is that it complies with and meets the intent of the regulatory requirements, both internationally and in the country where a project is located. Alternative approaches are, therefore, minimized in the design. An alternative approach is proposed for a design feature that does not strictly comply with a requirement in the applicable regulatory document. This report includes a description of alternative approaches. The alternative approaches identified in the report are related to the following design features:

- Containment isolation valves
- Radwaste structures, systems, and components

Following an Independent Peer Review (IPR) effort conducted in accordance with REGDOC-2.5.2, Section 5.6, it has been determined that the BWRX-300 meets applicable requirements and guidance related to means of shutdown. Therefore, the proposed alternative approach in Revision 0 of this document related to means of shutdown independence has been removed as an alternative approach. Refer to the Appendix of this report for further details.

This report does not specifically address alternatives to industry codes and standards, as those alternatives are evaluated using processes that are identified in the specific codes and standards, or in the applicable regulatory requirements related to applying those codes and standards. However, an industry code or standard may relate to an alternative approach. This report includes details on the industry codes and standards that may apply specifically to the alternative approach.

1.0 INTRODUCTION

This report addresses alternative approaches for meeting the requirements of the Canadian Nuclear Safety Commission (CNSC) regulatory document for design of nuclear power plants as related to the BWRX-300. Specifically, this report addresses design requirements in CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants (Reference 5-1). The report applies to the Ontario Power Generation Darlington New Nuclear Project.

Detailed descriptions of design information related to the alternative approaches are found in the BWRX-300, Darlington New Nuclear Project (DNNP) Preliminary Safety Analysis Report (PSAR), with references identifying the applicable PSAR Report number.

1.1 Scope

This report addresses the application of an overall graded approach for the design and identifies requirements for applying an alternative approach to certain specific design requirements in accordance with REGDOC-2.5.2, Section 11, Alternative Approaches. It includes information regarding proposed alternative approaches in the design and describes in brief the justification for acceptance of these alternative approaches. The technical basis for an alternative approach is based on the current stage of the design process. Accordingly, alternative approaches identified in this report are subject to change, and other alternative approaches may be identified through the design process.

Summary descriptions address the items that are currently identified as alternative approaches that are being implemented in the design process at the time of issuance of this report. The level of detail supporting application of the alternative approach is commensurate with the level of design information available during the current phase of design. Further details on compliance with REGDOC-2.5.2 requirements will be available in subsequent design phases. Where a specific graded approach applies to an alternative approach, it is identified in the summary description. A graded approach is applied to the safety assessment framework in the overall design process, based on a defence-in-depth approach for identifying safety categories of functions and safety class of structures, systems, and components.

2.0 DESCRIPTION

Alternative approaches are identified and a summary description of the basis for acceptance of each of the alternative approaches is provided. The CNSC regulatory requirements related to alternative approaches are described as they are applied and implemented in the BWRX-300 design for the Darlington New Nuclear Plant. Ontario Power Generation (OPG) is the applicant for the reactor facility and GEH is the BWRX-300 developer and the design authority.

2.1 Assumptions and Limitations

The alternative approaches identified in this report are those that relate to specific requirements in REGDOC-2.5.2 (Reference 5-1) as applied to the current design phase. As the design progresses, there may be changes to the alternative approaches, or additional alternatives may be identified. Information will be provided in a revision to this report or in an appropriate licensing document. A comprehensive review of design and regulatory documents, as compared to requirements in REGDOC-2.5.2, was performed to identify the alternative approaches included at this stage of the design.

This report does not address in detail the application of industry codes and standards and the alternatives that may be applied in how these are specifically implemented in the design. Neither does this report describe in detail the application of graded approaches (including risk-informed activities), other than to describe the concepts for a graded approach and how these may apply to the alternative approaches addressed in this report, and as the safety assessment framework applying a defence-in-depth (D-in-D) approach is considered in evaluating the alternative approaches. To the extent that an alternative approach relies on an assumption, it is stated in the section related to the alternative approach. Assumptions for applying industry codes and standards and a graded approach are addressed below.

2.1.1 Applying Industry Codes and Standards

The BWRX-300 applies industry codes and standards consistent with the requirements and guidance in REGDOC-2.5.2 (Reference 5-1). These industry codes and standards may include provisions for, or regulations may dictate, a process to apply alternatives when implementing the industry codes and standards. The BWRX-300 design organization identifies the applicable editions and addenda and alternatives for these industry codes and standards as part of the design process and program development. This report does not address alternative approaches related to industry codes and standards but may refer to industry codes and standards as related to alternative approaches that implement certain portions of such codes and standards to explain or justify an alternative approach.

2.1.2 Applying a Graded Approach

The BWRX-300 design principles and criteria are based on a graded approach across the facility consistent with REGDOC-3.5.3, CNSC Processes and Practices, Regulatory Framework (Reference 5-2), and as described in CNSC REGDOC-1.1.5, Supplemental Information for Small Modular Reactor Proponents (Reference 5-3). The graded approach is a method or process by which elements such as the level of analysis, the depth of documentation and the scope of actions necessary to comply with requirements are commensurate with the following:

1. The relative risks to health, safety, security, the environment, and the implementation of international obligations to which Canada has agreed
2. The characteristics of a facility or activity

For the BWRX-300, application of a graded approach is not specifically considered an alternative approach under REGDOC-2.5.2 requirements. A graded approach is, rather, a process for implementing the requirements considering the relative risks to health, safety, security, the environment, and safeguards, and is used as part of the overall design process and program development and in applying risk-informed decision making.

For the BWRX-300, a graded approach may be applied to design features, safety analyses, or programs. The concepts for a graded approach may include such programs and processes as assessment of risk; quality category, program, or process; applicable regulatory requirements and guidance; availability of operating experience; and potential restraints or constraints that may apply. Implementation of a graded approach to meet requirements provides an acceptable means of compliance and allows for an appropriate focus on risk and safety aspects.

More specifically to the evaluations for alternative approaches, the D-in-D approach in the overall safety assessment framework is applied in the design by identifying along defence lines (DLs) the safety categories of functions and the safety classification of structures, systems, and components (SSC). The D-in-D approach used in the BWRX-300 design is described in detail in NEDO-33952, DNNP PSAR, Chapter 3 (Reference 5-4).

Except as identified and addressed in this report, no other specific applications of a graded approach are identified as alternatives for the current design phase. If alternatives involving a graded approach are identified as the design progresses, they will be evaluated appropriately.

2.2 CNSC Design Requirements and Guidance

REGDOC-2.5.2 (Reference 5-1) provides requirements applicable to the design of the BWRX-300. This report focuses on the requirements where an alternative approach for certain of those design requirements is identified. REGDOC-2.5.2 guidance applicable to a specific alternative approach is also discussed in this report. The subsections below discuss the requirements for alternative approaches, application of the safety assessment framework and a risk-informed decision-making approach, and early engagement with regulatory agencies in pre-licensing activities.

2.2.1 REGDOC-2.5.2 Requirements for Alternative Approaches

REGDOC-2.5.2 (Reference 5-1) specifies requirements for alternative approaches to the design requirements included therein. The requirements for identifying and justifying alternative approaches are as follows:

“Section 11, Alternative Approaches

The requirements in this regulatory document are intended to be technology neutral for water-cooled reactor designs. It is recognized that specific technologies may use alternative approaches.

The CNSC will consider alternative approaches to the requirements in this document where:

1. the alternative approach would result in an equivalent or superior level of safety
2. the application of the requirements in this document conflicts with other rules or requirements
3. the application of the requirements in this document would not serve the underlying purpose, or is not necessary to achieve the underlying purpose

Any alternative approach shall demonstrate equivalence to the outcomes associated with the use of the requirements set out in this regulatory document.”

2.2.2 Safety Assessment Framework and Risk-Informed Graded Approach

OPG submitted a Licence to Construct (LTC) Application for the BWRX-300. The BWRX-300 is a light-water cooled design that employs passive safety features and, in some areas, first-of-a-kind engineering applications. The safety assessment framework for the BWRX-300 is based on the intent of the International Atomic Energy Agency (IAEA) safety requirements and guidance in the following documents:

- IAEA GSR Part 4, Safety Assessment for Facilities and Activities (Reference 5-5)
- IAEA SSR-2/1, Safety of Nuclear Power Plants: Design (Reference 5-6)
- IAEA SSG-30, Safety Classification of Structures, Systems, and Components (SSCs) in Nuclear Power Plants (Reference 5-7)

The CNSC indicates in REGDOC-2.5.2 (Reference 5-1), Section 2, Scope, that “[t]o a large degree, this document represents the CNSC’s adoption of the principles set forth in the IAEA document SSR-2/1 (Reference 5-6), and the adaptation of those principles to align with Canadian practices.” Accordingly, the BWRX-300 safety assessment framework for the design is consistent with the principles and practices established in REGDOC-2.5.2.

The BWRX-300 safety assessment framework, which is based on IAEA SSR-2/1 (Reference 5-6), uses risk insights, and applies a graded approach. Applying a graded approach to safety analysis is discussed in CNSC REGDOC-2.4.1, Deterministic Safety Analysis (Reference 5-8), Section 6, which indicates that a graded approach is a method in which the stringency of the design measures and analyses applied are commensurate with the level of risk posed by the reactor facility. Applying a graded approach based on risk insights is not, therefore, considered an alternative approach. As described above, the safety assessment framework implements a D-in-D approach using defence lines for safety categories and safety class.¹ This safety approach and the defence lines, which apply to the overall BWRX-300 design process, are discussed in the evaluations of the alternative approaches.

The BWRX-300 is also designed using risk insights to apply the appropriate focus on safety of the design. There is no specific alternative approach for using risk-informed decision-making in the design because, as acknowledged in the Preface, REGDOC-2.5.2 establishes a set of comprehensive design requirements and guidance that are risk-informed and that align with accepted international codes and practices (Reference 5-1).

2.2.3 BWRX-300 Vendor Design Review and Regulatory Collaboration

GEH has engaged with the CNSC in a Vendor Design Review effort for the BWRX-300 design, as well as with the U.S. Nuclear Regulatory Commission (NRC), for certain design features that are described in this report as alternative approaches. These pre-licensing interactions are encouraged, as explained in REGDOC-1.1.5, Supplemental Information for Small Modular Reactor Proponents (Reference 5-3), which describes a process for providing regulatory clarity in pre-licensing for review of a vendor’s reactor design.

¹ As used in the BWRX-300 classification system, “Safety Category” refers to the classification applied to functions to reflect their role in ensuring plant safety and “Safety Class” refers to the classification applied to the structures, systems, and components to reflect their role in ensuring plant safety based on their functions and their safety significance.

Also, the CNSC and the NRC are working collaboratively under a 2017 Memorandum of Understanding (Reference 5-9) and a 2019 Joint Memorandum of Cooperation (Reference 5-10) to review reactor designs, as described in a Joint Report on GE Hitachi's Containment Evaluation Method (Reference 5-11).

It is expected that the collaborative efforts will continue and that licensing reviews involving the BWRX-300 design benefit from this cooperation between regulatory agencies. Such collaboration may involve further interactions related to the alternative approaches identified in this report, or to alternatives which may be identified in the future. CNSC did not identify any fundamental barriers to licensing through the combined Phase 1 and 2 Vendor Design Review efforts.

2.3 Applying an Alternative Approach

As explained above in the Scope section, supporting details to substantiate compliance statements are commensurate with the level of design progress in the current design phase. Alternative approaches identified at the current stage of the design are discussed in this report. If later design details identify that an alternative is no longer necessary, then the design will otherwise meet regulatory requirements. If additional alternative approaches are identified, then they will be evaluated appropriately.

Reasons for an alternative approach vary. As an example, an alternative approach may be proposed in a case where a design feature is considered appropriate because of industry experience, among other aspects. An alternative approach may be applied where a different means of complying is determined to be an acceptable approach from a safety standpoint, or because there are factors that are considered while maintaining safety in the design. An alternative approach, especially for the BWRX-300 small modular reactor, may be applied because there is a novel feature or a complementary method, or based on risk-informed decision making. As the design progresses, if an alternative approach is identified, it will be justified and evaluated, as appropriate.

2.4 Outline for Proposed BWRX-300 Design Alternative Approaches

The BWRX-300 design team proposes to apply an alternative approach for meeting the regulatory requirements in the areas identified below. This report provides a summary description of the REGDOC-2.5.2 related requirement, the proposed alternative, and the justification for applying the alternate, consistent with a process for evaluating alternative approaches using the REGDOC-2.5.2 Section 11 guidance as a format.

The summary description of an alternative presented in this report is based on information consistent with the current status of the design phase. These proposed alternative approaches may be modified, or additional items may be identified as the project proceeds and the detailed design progresses.

The format for the sections below is as follows:

- REGDOC-2.5.2 requirement is identified.
- Proposed alternative approach is described.
- Criteria for alternative approaches are listed and marked as appropriate.
- Justification is summarized.

Alternative approaches described below are related to the following design features:

- Containment isolation valves

- Radwaste structures, systems, and components

2.5 Containment Isolation Valves

REGDOC-2.5.2 includes requirements for the design of containment isolation valves (CIVs).

2.5.1 Specific Descriptions of CIV Requirements and Design

The following information describes the requirements for CIV and the specific design features.

2.5.1.1 REGDOC-2.5.2 Requirements

REGDOC-2.5.2 includes the following requirements for the design of containment isolation:

“8.6.6 Containment isolation

Each line of the reactor coolant pressure boundary that penetrates the containment, or that is connected directly to the containment atmosphere, shall be automatically and reliably sealed. This requirement is essential to maintaining the leak tightness of the containment in the event of an accident and preventing radioactive releases to the environment that exceed prescribed limits.

Automatic isolation valves shall be positioned to provide the greatest safety upon loss of actuating power.

Piping systems that penetrate the containment system shall have isolation devices with redundancy, reliability, and performance capabilities that reflect the importance of isolating the various types of piping systems. Alternative types of isolation may be used where justification is provided.

Where manual isolation valves are used, they shall be readily accessible and have locking or continuous monitoring capability.

Reactor coolant system auxiliaries that penetrate containment

Each auxiliary line that is connected to the reactor coolant pressure boundary, and that penetrates the containment structure, shall include two isolation valves in series. The valves shall be normally arranged with one inside and one outside the containment structure.

Where the valves provide isolation of the heat transport system during normal operation, both valves shall be normally in the closed position.

Systems directly connected to the reactor coolant system that may be open during normal operation shall be subject to the same isolation requirements as the normally closed system, with the exception that manual isolating valves inside the containment structure will not be used. At least one of the two isolation valves shall be either automatic or powered, and operable from the main and secondary control rooms.

For any piping outside of containment that could contain radioactivity from the reactor core, the following requirements shall apply:

1. The design parameters shall be the same as those for a piping extension to containment and are subject to the requirements for metal penetrations of containment.
2. All piping and components that are open to the containment atmosphere shall be designed for a pressure greater than the containment design pressure.

3. The piping and components shall be housed in a confinement structure that prevents leakage of radioactivity to the environment and to adjacent structures.
4. This housing shall include detection capability for leakage of radioactivity and shall include the capability to deal safely with the leakage.

Systems connected to containment atmosphere

Each line that connects directly to the containment atmosphere, that penetrates the containment structure and is not part of a closed system, shall be provided with two isolation barriers that meet the following requirements:

1. two automatic isolation valves in series for lines that may be open to the containment atmosphere
2. two closed isolation valves in series for lines that are normally closed to the containment atmosphere
3. the line up to and including the second valve is part of the containment envelope

Closed systems

All closed piping service systems shall have at least one single isolation valve on each line penetrating the containment, with the valve being located outside of, but as close as practicable to, the containment structure.

Where failure of a closed loop is assumed to be a PIE [postulated initiating event] or the result of a PIE, the isolations appropriate to the system shall apply.

Closed piping service systems whether inside or outside the containment structure which form part of the containment envelope, require no further isolation if

1. they meet the applicable service piping standards and codes
2. they can be continuously monitored for leaks”

2.5.1.2 Proposed Alternative Approach

The Containment pressure boundary function is to provide a leak-tight barrier, preventing the release of radioactive material in the event of a failure of the Reactor Coolant Pressure Boundary (RCPB). Confinement is a fundamental safety function. The BWRX-300 confinement function is achieved by the Primary Containment System and other systems with which it interfaces. Accordingly, as to the general requirements for containment in REGDOC-2.5.2, Section 8.6.1, it is considered that the overall BWRX-300 Containment complies with the requirements. To the extent that an issue may be identified that impacts the overall containment, it would be evaluated to determine if it should be addressed as an alternative approach. REGDOC-2.5.2, Section 8.6.1, lists “Additional Information” and identifies applicable codes. The BWRX-300 containment design addresses applicable codes and standards. In the current stage of design, however, certain containment isolation valve configurations are the focus of this report as alternatives to the requirements in REGDOC-2.5.2, Section 8.6.6.

Containment Isolation Valves (CIVs) provide the necessary isolation of the Primary Containment in the event of accidents or other conditions and prevent the unfiltered release of containment contents that would exceed regulatory limits. Certain cases discussed in this section are identified and evaluated as an alternative means of meeting REGDOC-2.5.2 requirements for the reasons explained below.

2.5.1.3 Justification Assessment

Criteria for Justifying Alternative Approach	Yes	No
The alternative approach would result in an equivalent or superior level of safety.	X	
The application of the requirements in this document conflicts with other rules or requirements.		X
The application of the requirements in this document would not serve the underlying purpose or is not necessary to achieve the underlying purpose.		X

Detailed basis for alternative:

- Address one or more of the criteria marked “Yes” above.
- Describe how the alternative approach demonstrates equivalence to the outcomes in related the section(s) of REGDOC-2.5.2.
- Include additional details sufficient to justify the alternative approach.

2.5.1.3.1 Containment Description

The BWRX-300 containment comprises a Steel-Plate Composite Containment Vessel (SCCV), a steel containment closure head, and other components. As described in NEDO-33952, DNNP PSAR, Chapter 3, Section 3.5.3 (Reference 5-4), the design of the containment conforms to the applicable industry codes and standards and complies with the regulatory requirements in REGDOC-2.5.2. Details of the containment systems design are described in NEDO-33955, DNNP PSAR, Chapter 6, Section 6.3.2 (Reference 5-14). The alternative approach focuses on certain specific configurations of the containment isolation system.

2.5.1.3.2 Containment Isolation System

Details of the containment isolation system are in NEDO-33955, DNNP PSAR, Chapter 6, Section 6.3.4 (Reference 5-14), which describes the isolation of mechanical systems penetrating primary containment. To support isolation of lines penetrating containment, containment isolation valves (CIVs) and penetrations are included in the system lines. Redundant CIVs are included in each system line near the containment boundary and which close on predefined parameters to prevent release from containment (with certain exceptions discussed herein). Penetrations are included at the boundary between the line and the containment to limit leakage at the connection and are designed to maintain structural integrity and to protect against leakage during the extreme environmental conditions as the result of an accident.

Piping located between the CIV and penetration is considered a part of the containment boundary as well, and therefore, supports the containment isolation function. The CIVs, piping, and the penetrations are included as a part of the containment boundary, and therefore, they support the containment in limiting release of fission product leakage during and after an accident.

The arrangement of CIVs for the BWRX-300 design is developed in accordance with the requirements set forth in U.S. NRC regulations in 10 Code of Federal Regulations (CFR) Part 50, Appendix A, General Design Criteria (GDC) (Reference 5-15) as follows: 10 CFR 50 Appendix A (GDC 55), Reactor Coolant Pressure Boundary Penetrating Containment, 10 CFR 50 Appendix A (GDC 56), Primary Containment Isolation, and 10 CFR 50 Appendix A (GDC 57), Closed System Isolation Valves, with the following system CIVs acceptable “on some other defined basis”

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cases that otherwise meets GDC 55 or GDC 56 intent (note that GDC 57 does not apply to these five systems):

- Isolation Condenser System (ICS)
- Boron Injection System (BIS)
- Control Rod Drive (CRD) System
- Containment Inerting System (CIS)
- Equipment and Floor Drain System (EFS)

These five systems and containment isolation configurations are evaluated as an alternative approach to the specific requirements in REGDOC-2.5.2 considering the current stage of design. The GDC for CIVs are as identified in Table 2.5-1:

Table 2.5-1: U.S. NRC 10 CFR Part 50, Appendix A, GDC 55, 56, and 57

GDC 55	GDC56	GDC 57
<p>Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:</p> <p>(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or</p> <p>(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or</p> <p>(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or</p> <p>(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.</p> <p>Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to</p>	<p>Criterion 56—Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:</p> <p>(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or</p> <p>(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or</p> <p>(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or</p> <p>(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.</p> <p>Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.</p>	<p>Criterion 57—Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.</p> <p>[NOTE: This Criterion 57 is not applicable to the containment isolation configurations addressed in the alternative approaches described below.]</p>

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GDC 55	GDC56	GDC 57
<p>take the position that provides greater safety.</p> <p>Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.</p>		

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The following table (Table 2.5-2) identifies, for each of the five systems, whether or not the specific requirement is part of the proposed alternative approach for that system. Details of the containment isolation configuration for these five systems are described in NEDO-33955, DNNP PSAR, Chapter 6, Section 6.3.4.3 (Reference 5-14).

Table 2.5-2: Comparison to REGDOC-2.5.2, Section 8.6.6, Requirements

NOTE: Responses in this table discuss design for each of the five systems as related to the U.S. Nuclear Regulatory General Design Criteria and how these criteria apply to the containment isolation configuration in REGDOC-2.5.2, Section 8.6.6. Note that this information is not described directly in the DNNP PSAR. The responses are subject to change as the design progresses.

REGDOC-2.5.2 Requirements 8.6.6 Containment Isolation	General	Isolation Condenser System (ICS) Alternative (GDC 55 Other Defined Basis)	Boron Injection System (BIS) Alternative (GDC 55 Other Defined Basis)	Control Rod Drive (CRD) Hydraulic System Alternative (GDC 55 Other Defined Basis)	Containment Inerting System (CIS) Alternative (GDC 56 Other Defined Basis)	Equipment and Floor Drain System (EFS) Alternative (GDC 56 Other Defined Basis)
		[[
Each line of the reactor coolant pressure boundary that penetrates the containment, or that is connected directly to the containment atmosphere, shall be automatically and reliably sealed. This requirement is essential to maintaining the leak tightness of the containment in the event of an accident, and preventing radioactive releases to the environment that exceed prescribed limits.	[[
Automatic isolation valves shall be positioned to provide the greatest safety upon loss of actuating power.						

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REGDOC-2.5.2 Requirements 8.6.6 Containment Isolation	General	Isolation Condenser System (ICS) Alternative (GDC 55 Other Defined Basis)	Boron Injection System (BIS) Alternative (GDC 55 Other Defined Basis)	Control Rod Drive (CRD) Hydraulic System Alternative (GDC 55 Other Defined Basis)	Containment Inerting System (CIS) Alternative (GDC 56 Other Defined Basis)	Equipment and Floor Drain System (EFS) Alternative (GDC 56 Other Defined Basis)	
		[[]]
Piping systems that penetrate the containment system shall have isolation devices with redundancy, reliability, and performance capabilities that reflect the importance of isolating the various types of piping systems. Alternative types of isolation may be used where justification is provided.							
Where manual isolation valves are used, they shall be readily accessible and have locking or continuous monitoring capability.							
<i>Reactor coolant system auxiliaries that penetrate containment</i>							
Each auxiliary line that is connected to the reactor coolant pressure boundary, and that penetrates the containment structure, shall include two isolation valves in series. The valves shall be normally arranged with one inside and one outside the containment structure.							

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REGDOC-2.5.2 Requirements 8.6.6 Containment Isolation	General	Isolation Condenser System (ICS) Alternative (GDC 55 Other Defined Basis)	Boron Injection System (BIS) Alternative (GDC 55 Other Defined Basis)	Control Rod Drive (CRD) Hydraulic System Alternative (GDC 55 Other Defined Basis)	Containment Inerting System (CIS) Alternative (GDC 56 Other Defined Basis)	Equipment and Floor Drain System (EFS) Alternative (GDC 56 Other Defined Basis)	
		[[]]
Where the valves provide isolation of the heat transport system during normal operation, both valves shall be normally in the closed position.							
Systems directly connected to the reactor coolant system that may be open during normal operation shall be subject to the same isolation requirements as the normally closed system, with the exception that manual isolating valves inside the containment structure will not be used. At least one of the two isolation valves shall be either automatic or powered, and operable from the main and secondary control rooms.							

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REGDOC-2.5.2 Requirements 8.6.6 Containment Isolation	General	Isolation Condenser System (ICS) Alternative (GDC 55 Other Defined Basis)	Boron Injection System (BIS) Alternative (GDC 55 Other Defined Basis)	Control Rod Drive (CRD) Hydraulic System Alternative (GDC 55 Other Defined Basis)	Containment Inerting System (CIS) Alternative (GDC 56 Other Defined Basis)	Equipment and Floor Drain System (EFS) Alternative (GDC 56 Other Defined Basis)	
		[[]]
For any piping outside of containment that could contain radioactivity from the reactor core, the following requirements shall apply: 1. The design parameters shall be the same as those for a piping extension to containment and are subject to the requirements for metal penetrations of containment. 2. All piping and components that are open to the containment atmosphere shall be designed for a pressure greater than the containment design pressure. 3. The piping and components shall be housed in a confinement structure that prevents leakage of radioactivity to the environment and to adjacent structures. 4. This housing shall include detection capability for leakage of radioactivity and shall include the capability to deal safely with the leakage.							
<i>Systems connected to containment atmosphere</i>							

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REGDOC-2.5.2 Requirements 8.6.6 Containment Isolation	General	Isolation Condenser System (ICS) Alternative (GDC 55 Other Defined Basis)	Boron Injection System (BIS) Alternative (GDC 55 Other Defined Basis)	Control Rod Drive (CRD) Hydraulic System Alternative (GDC 55 Other Defined Basis)	Containment Inerting System (CIS) Alternative (GDC 56 Other Defined Basis)	Equipment and Floor Drain System (EFS) Alternative (GDC 56 Other Defined Basis)	
		[[]]
Each line that connects directly to the containment atmosphere, that penetrates the containment structure and is not part of a closed system, shall be provided with two isolation barriers that meet the following requirements: 1. two automatic isolation valves in series for lines that may be open to the containment atmosphere 2. two closed isolation valves in series for lines that are normally closed to the containment atmosphere 3. the line up to and including the second valve is part of the containment envelope							
Closed systems							
All closed piping service systems shall have at least one single isolation valve on each line penetrating the containment, with the valve being located outside of, but as close as practicable to, the containment structure.							
Where failure of a closed loop is assumed to be a PIE or the result of a PIE, the isolations appropriate to the system shall apply.							

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REGDOC-2.5.2 Requirements 8.6.6 Containment Isolation	General	Isolation Condenser System (ICS) Alternative (GDC 55 Other Defined Basis)	Boron Injection System (BIS) Alternative (GDC 55 Other Defined Basis)	Control Rod Drive (CRD) Hydraulic System Alternative (GDC 55 Other Defined Basis)	Containment Inerting System (CIS) Alternative (GDC 56 Other Defined Basis)	Equipment and Floor Drain System (EFS) Alternative (GDC 56 Other Defined Basis)
		[[
Closed piping service systems whether inside or outside the containment structure which form part of the containment envelope, require no further isolation if: 1. they meet the applicable service piping standards and codes 2. they can be continuously monitored for leaks]]

For these five systems, the following Table 2.5-3 provides the valve positions under normal, shutdown, post-accident, and loss of motive power.

Table 2.5-3: Mechanical Containment Isolation Valve Positions

System	CIV Description	Valve Position			
		Normal	Shutdown	Post-Accident	Loss of Motive Power
ICS	[[
]]
	Row deleted.				
BIS	[[
CRD					
CIS					
EFS					
]]

(1) [[]].

[[]].

2.5.1.3.2.1 Isolation Condenser System

The Isolation Condenser System (ICS) functions as the Emergency Core Cooling System (ECCS) for core cooling in the event, as described in NEDO-33955, DNNP PSAR, Chapter 6, Section 6.2.1 (Reference 5-14).

The BWRX-300 is designed for RPV inventory preservation with passive Defense Line 3 (Safety Class 1) safety features. [[

]]

- [[]].
- [[]].
- [[]].

The relatively large RPV volume with the relatively tall chimney region provides a substantial reservoir of water above the core. [[

]].

The ICS functions as the Emergency Core Cooling System (ECCS) in the BWRX-300 design in support of the preservation of reactor coolant inventory.

The containment isolation design for the ICS steam supply and condensate return lines ensure that the valve positions maintain the system capabilities to provide emergency core cooling, rather than closing on containment isolation. A single failure does not disable the containment isolation function. The two in-series reactor vessel isolation valves that function as CIVs remain open during accident conditions allowing the ICS to function as ECCS, as described in NEDO-33955, DNNP PSAR, Chapter 6, Section 6.3.4 (Reference 5-14).

The ICS valve configuration alternative approach provides an acceptable, essentially equivalent, level of safety for containment isolation, while also considering the ICS functions for providing emergency core cooling. The ICS containment isolation configuration is considered an “other defined basis” under NRC GDC-55, providing an acceptable level of safety as an alternative containment isolation configuration.

2.5.1.3.2.2 Boron Injection System

The Boron Injection System (BIS) injection line including the injection valve penetrates containment and is directly connected to the reactor vessel through the ICS. Therefore, this portion of the system has requirements for containment isolation and is a portion of the Reactor Coolant Pressure Boundary (RCPB).

The line has a remote manual, air-operated outboard CIV and a check valve as the inboard containment and isolation valve. The injection location is into the condensate return line of the ICS “C” loop downstream of the two ICS condensate return valves, providing a direct flow path into the reactor. The injection location into the ICS condensate return line requires a separate containment penetration for the BIS injection line. The air-operated injection valve, when open, provides a flow path to the reactor when injecting either the boron solution or demineralized water into the reactor. The injection valve, when closed after an injection event, allows for re-establishment of containment isolation. This satisfies 10 CFR Part 50 Appendix A (GDC 55) requirement based on “other defined basis.” This “other defined basis” is determined acceptable because locating a valve inside containment when needed for accident mitigation is not practical and should not be isolated when it is needed for accident mitigation. The injection line has a remote manual, air-operated outboard CIV to satisfy 10 CFR Part 50 Appendix A (GDC 55). The injection line also has a check valve inboard to satisfy 10 CFR Part 50 Appendix A (GDC 55). Design of the BIS containment isolation configuration is as described in NEDO-33955, DNNP PSAR, Chapter 6, Section 6.3.4 (Reference 5-14).

On this basis, the BIS containment isolation configuration provides an acceptable level of safety as an alternative containment isolation configuration. It is considered an “other defined basis” configuration under NRC GDC-55.

2.5.1.3.2.3 Control Rod Drive System

Hydraulic lines for the hydraulic scram function use penetrations without isolation valves based on the closed system piping outside primary containment and the RCPB isolation uses internal ball check valves in the drive design. The CRD system and the associated hydraulic insertion line performs a safety critical function by providing high-pressure water to produce a reactor scram. Each hydraulic unit includes an integral ball check valve at the drive flange insert port. The check ball operation serves to plug the insert line port and limits the reactor coolant

discharged in the event of an insert line break. Additionally, manual isolation valves may be used to further isolate the HCU from the hydraulic insertion line if needed. To summarize:

- In the control rod drive hydraulic lines scram function, the containment penetrations do not have automatic isolation valves. This is based upon these lines being closed piping systems outside containment and having reactor coolant pressure boundary isolation (internal ball check valves) in the design of the control rod drives.
- HCUs meet the REGDOC-2.5.2 requirements as an alternate configuration by forming a closed system outside containment, except that the closed system does not include outside containment isolation valves.

This closed-loop piping outside containment can prevent radioactive releases outside containment to satisfy the U.S. NRC GDC-55, requirement based on “other defined basis.” The control rod drive system containment isolation configuration is as described in NEDO-33955, DNNP PSAR, Chapter 6, Section 6.3.4 (Reference 5-14). The design features for the containment isolation configurations of the CRD system provide an acceptable and essentially equivalent level of safety for containment isolation while preserving functionality of the systems and the containment penetration can be isolated, if necessary, for assuring a leak-tight containment and maintaining necessary connections to the reactor coolant pressure boundary inside containment or in an extension of containment.

2.5.1.3.2.4 Containment Inerting System

The Containment Inerting System (CIS) is designed to establish and maintain an inert atmosphere (nitrogen) within containment.

The Containment Inerting System containment isolation configuration complies with the REGDOC-2.5.2 requirements for systems open to the containment, except that both CIVs are located outside of containment. Each CIS line that penetrates containment is provided with CIVs and connects directly to containment atmosphere. Both isolation valves on these lines are located outside the containment vessel to remove them from the harsh containment environment and protect them from the effects of flood and dynamic effects of pipe breaks.

As described in NEDO-33955, DNNP PSAR, Chapter 6, Section 6.3.4 (Reference 5-14), for containment isolation, each line in the CIS that connects directly to the containment atmosphere and penetrates primary reactor containment is provided with CIVs. Both isolation valves on these lines are located outside of the containment. The valves are located as close as practical to the containment vessel. The piping from the containment vessel to and including both valves is an extension of the primary containment boundary and is designed in accordance with code requirements. The arrangement of the isolation valves and connecting piping is such that a single active failure of an inboard valve, or a single active or passive failure in the connecting piping or an outboard valve, cannot prevent isolation of the CIS containment penetrations. This satisfies NRC 10 CFR Part 50 Appendix A, GDC 56 requirements based on “other defined basis.”

Based on the information provided above, the CIS containment isolation design meets the requirements of REGDOC-2.5.2, Section 8.6.6, except that the configuration includes design features for severe accident mitigation. These DL4b design features perform a function to relieve containment pressure, either by a passive rupture disc or by manual opening of a bypass valve, with discharge into the Reactor Building (RB) equipment pool. It is important that this function be available if needed. Once this important design feature has completed its function to lower containment pressure, the containment overpressure protection air-operated isolation valve can be closed to re-establish the containment pressure boundary.

On this basis, the CIS containment isolation configuration provides an acceptable level of safety as an alternative containment isolation configuration. The configuration is considered an “other defined basis” under NRC GDC-56.

2.5.1.3.2.5 Equipment and Floor Drain System

Water from various plant areas is sent to the Liquid Waste Management collection tanks located in the Radwaste Building by the Equipment and Floor Drain System (EFS). The EFS is a non-Safety Class system with the exception of the containment isolation valves upstream of the pressurized sump. As described in NEDO-33955, DNNP PSAR, Chapter 6, Section 6.3.4 (Reference 5-14), the EFS line that penetrates containment is provided with CIVs and connects directly to containment atmosphere. There are two CIVs placed in series, located outside containment that are placed as close to the primary containment wall as practical. It is impractical to have an isolation valve inside containment above the floor elevation. The pressurized containment sump tank is credited for containment leak detection. The isolation valves on the EFS containment drain line are normally open to connect the drain line to the pressurized sump. The EFS CIVs close automatically. Upon loss of actuating power, the automatic isolation valves fail closed, providing a greater safety position.

These valves are consistent with and satisfy the NRC 10 CFR Part 50 Appendix A GDC 56 requirements based on “other defined basis.” One EFS line connects directly to the containment atmosphere and penetrates primary reactor containment is provided with CIVs. Two automatic isolation valves are located outside containment.

On this basis, the EFS containment isolation configuration provides an acceptable level of safety as an alternative containment isolation configuration. The configuration is considered an “other defined basis” under NRC GDC-56.

2.5.2 Overall Summary for Alternate Containment Isolation Configurations

Based on the information presented, for the current phase of design and analyses, the five systems that are addressed in this section as alternative approaches to the REGDOC-2.5.2 Section 8.6.6 requirements are designed for the systems to be configured to [[

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considering the system functions. Each system configuration provides an acceptable level of safety as to containment isolation in that there is protection from radioactive releases, either as directly configured for containment isolation, or as may be remote manually configured, considering the system functions that may be needed in accident conditions, including design extension conditions and severe accidents in certain scenarios. Therefore, it is concluded that the alternate containment isolation configurations provide an acceptable level of quality and safety.

2.6 Radwaste Structures, Systems, and Components

REGDOC-2.5.2 includes requirements for classification and design of radwaste structures, systems, and components.

2.6.1 REGDOC-2.5.2 Requirements

REGDOC-2.5.2 requirements related to radioactive waste treatment systems classification and design include the following:

“Section 7.1 Safety classification of structures, systems and components

The design authority shall classify SSCs using a consistent and clearly defined classification method. The SSCs shall then be designed, constructed, and

maintained such that their quality and reliability is commensurate with this classification.

Section 7.13.1 Seismic design and classification

The design authority shall ensure that seismically qualified SSCs important to safety are qualified to a design-basis earthquake (DBE) and ensure that they are categorized accordingly. This shall apply to:

1. SSCs whose failure could directly or indirectly cause an accident leading to core damage
2. SSCs restricting the release of radioactive material to the environment
3. SSCs that assure the subcriticality of stored nuclear material
4. SSCs such as radioactive waste tanks containing radioactive material that, if released, would exceed regulatory dose limits

Section 8.11 Waste treatment and control

The design shall include provisions to treat liquid and gaseous effluents in a manner that will keep the quantities and concentrations of discharged contaminants within prescribed limits, and that will support application of the ALARA principle. The design of the NPP shall minimize the generation of radioactive and hazardous waste. The design shall also include adequate provision for the safe onsite handling and storage of radioactive and hazardous wastes, for a period of time consistent with options for offsite management or disposal.”

2.6.2 Proposed Alternative Approach

The BWRX-300 design for radwaste structures, systems, and components (SSC) includes systems and components housed in a Radwaste Building (RWB) structure. The main function of the RWB is to process and house liquid, solid, and gaseous radwaste. The RWB houses the off-gas system, refueling water storage tanks, and rooms and equipment for handling, processing, and packaging liquid and solid radioactive wastes.

The alternative approach proposes applying a special seismic category for the RWB, which is also used to identify the building structural analysis requirements and the safety classification of Radwaste SSC. The alternative approach applies U.S. NRC guidance for structures housing radioactive waste material and processing and the related systems and components (as applicable). The alternative approach is described below.

2.6.3 Justification Assessment

Criteria for Justifying Alternative Approach	Yes	No
The alternative approach would result in an equivalent or superior level of safety.	X	
The application of the requirements in this document conflicts with other rules or requirements.		X
The application of the requirements in this document would not serve the underlying purpose or is not necessary to achieve the underlying purpose.		X

Detailed basis for alternative:

- Address one or more of the criteria marked “Yes” above.
- Describe how the alternative approach demonstrates equivalence to the outcomes in related the section(s) of REGDOC-2.5.2.
- Include additional details sufficient to justify the alternative approach.

The Radwaste Building (RWB) is part of the BWRX-300 standard design for the power block buildings. It is located next to the Reactor Building (RB), Turbine Building (TB), and Control Building. From a seismic perspective, it is evaluated for seismic interactions with these buildings, as well as being evaluated for the effects due to a seismic event on the RWB structure itself and the systems and components housed within the RWB.

The alternative approach focuses on the seismic design category of the RWB, which includes protection for the systems and components within the RWB, considering the current design phase. The seismic category designation also defines the treatment of natural phenomena for the RWB, where applicable, such as wind and missiles, to ensure that the RWB and the systems and components housed in the RWB are protected.

The alternative approach proposes to apply U.S. NRC guidance in Regulatory Guide (RG) 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water Cooled Nuclear Power Plants (Reference 5-16), for the seismic category of the RWB. The U.S. NRC reviewed Licensing Topical Report (LTR) NEDO-33914, submitted by GEH “to present design, analysis, and monitoring guidelines and requirements for construction of a BWRX-300 Small Modular Reactor (SMR) using innovative and comprehensive approaches that ensures safe operation throughout the life of the plant” (Reference 5-17). The U.S. NRC issued a Safety Evaluation for the LTR, noting that “the CB and TB structures are considered non-seismic and the RWB structure is considered RW-IIa category, as specified in RG 1.143” and that the [NRC] “staff finds this classification of these structures reasonable” (Reference 5-17).

The proposed alternative approach would classify the RWB and the applicable systems and components housed in the RWB as “RW-IIa” for consistency in applying RG 1.143 guidance based on the radiological significance of the functions of the systems and components, which is a graded, risk-informed approach.

2.6.3.1 RG 1.143 Seismic Category RW-IIa

NEDO-33952, DNNP PSAR, Chapter 3, Section 3.2.3 (Reference 5-4), identifies the seismic categories used in the BWRX-300 design. It explains that SSC for management and storage of radiological material that, if released, would exceed the dose limits defined in REGDOC-2.5.2, Section 4.2.1, are categorized as Seismic Category RW-IIa per guidance in RG 1.143. These RW-IIa SSC are seismically qualified for one-half of the site-specific Design Basis Earthquake (DBE). This approach is in accordance with REGDOC-2.5.2, Section 7.13.1, which permits the use of the ASCE/SEI 43 (Reference 5-18) graded approach for seismic classification of SSC with justification. Based upon the consequences of failure, one-half of the site-specific DBE is justified as it would bound the ground motion spectra for seismic categories identified in ASCE/SEI 43 (Reference 5-18) for SSCs used for handling and storage of highly radioactive materials. Additional design information regarding the RWB and SSC housed within the RWB is described in NEDO-33952, DNNP PSAR, Chapter 3, Section 3.3.1 (Reference 5-4) and NEDO-33959, DNNP PSAR, Chapter 9B, Section 9B.3.1 (Reference 5-19).

More specifically, the structures housing SSC for storage and processing of radioactive materials are classified based on the total design basis unmitigated radiological release considering the maximum inventory caused by the SSCs failure. Per RG 1.143, the High Hazard (RW-IIa) classification is assigned to structures hosting SSC for storage and processing of radiological

materials if failure of the SSC, considering the maximum inventory, results in a total design basis unmitigated radiological release greater than:

- 500 millirem or 5 mSv per year at the boundary of the unprotected area, or
- 5 rem or 250 mSv per year within the protected area.

Seismic category reflects SSC requirements during and after a seismic event and governs how the SSC is designed and qualified. Detailed design loads of the SSC for the RW-IIa category are developed in the site-specific building design specifications as the design progresses. For RW-IIa category structures, wind and tornado loading demands are per the detailed site-specific wind and tornado loading specification for RW-IIa category structures developed as the design progresses.

2.6.4 REGDOC-2.5.2 Guidance

The guidance in REGDOC-2.5.2, Section 7.13.1 discusses applying the seismic design categories (SDC) in ASCE/SEI 43, stating that the “seismic design of structural systems should be categorized according to seismic design category (SDC) 1 to 5 as per ASCE 43-05” (Reference 5-1). The Seismic Category RW-IIa RWB design is in accordance with applicable industry codes and standards, with seismic demand developed in accordance with RG 1.143 using the procedures outlined in ASCE/SEI 43, including the applicable requirements in related industry codes and standards, to comply with the regulatory requirements in Section 7.13.1 of REGDOC-2.5.2.

This industry standard establishes five levels of seismic design categories based on the severity of adverse effects. ASCE/SEI 43, Section 1.3, Seismic Design Criteria, provides criteria for seismic design of safety-related SSC to achieve target performance goals in nuclear facilities. It explains that the standard presents graded seismic design criteria, with more stringent design criteria used to achieve better seismic performance for SSC that have more serious failure consequences, as recognized by the designation of a seismic design basis, which varies by SSC.

2.6.5 Summary

Based on the information presented, for the current phase of design and analyses, applying a graded, alternative approach following U.S. NRC guidance in RG 1.143 provides an acceptable level of safety in the BWRX-300 design for categorizing and classifying radwaste SSC. In summary, following RG 1.143 to identify seismic design criteria for Radwaste SSC is considered essentially an equivalent level of safety for complying with the REGDOC-2.5.2 requirements.

3.0 REQUIREMENTS

This report describes and identifies certain proposed alternative approaches for the requirements of REGDOC-2.5.2 (Reference 5-1) and, where applicable, a graded approach for meeting such requirements. No requirements are specifically imposed by this report.

4.0 ACRONYMS, DEFINITIONS, AND SYMBOLS

4.1 Acronyms

Acronym	Explanation
AOO	Anticipated Operational Occurrence
ASCE	American Society of Civil Engineers
BDBA	Beyond Design Basis Accident
BIS	Boron Injection System
BWR	Boiling Water Reactor
BWRX-300	Boiling Water Reactor, 10 th Design – 300 MWe
CB	Control Building
CCF	Common Cause Failure
CFR	Code of Federal Regulation
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CNSC	Canadian Nuclear Safety Commission
CRA	Control Rod Assembly
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DEC	Design Extension Condition
D-in-D	Defence-in-Depth
DL	Defence Line
DNNP	Darlington New Nuclear Project
DPS	Diverse Protection System
ECCS	Emergency Core Cooling System
EFS	Equipment and Floor Drain System
GDC	General Design Criterion or General Design Criteria
GEH	GE Hitachi
HCU	Hydraulic Control Unit
IAEA	International Atomic Energy Agency
ICS	Isolation Condenser System
IPR	Independent Peer Review
LTC	Licence to Construct
LTR	Licensing Topical Report
NRC	U.S. Nuclear Regulatory Commission

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Acronym	Explanation
OPG	Ontario Power Generation
PIE	Postulated Initiating Event
PSAR	Preliminary Safety Analysis Report
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
REGDOC	Regulatory Document (Canadian Nuclear Safety Commission)
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
RW	Radwaste
RWB	Radwaste Building
SCCV	Steel-Plate Composite Containment Vessel
SDC	Seismic Design Category (as used in ASCE/SEI 43)
SEI	Structural Engineering Institute
SMR	Small Modular Reactor
SSC	Structures, Systems, and Components (may also be SSCs)
TB	Turbine Building
UPS	Uninterruptable Power Supply

4.2 Definitions

Term	Definition
Alternate approach	Any alternative approach shall demonstrate that safety and security protections are maintained or improved. Where risk characteristics contain uncertainties, the amount of evidence required for the applicant to demonstrate a credible decision increases. Suitable evidence may include results of research and development, computer modelling and consideration of operating experience, and the evidence must be demonstrated to be relevant to the specific proposal. All of these types of evidence should be documented, traceable and quality-assured. A proponent that is considering a licence application for an SMR is encouraged to engage with the CNSC early on, well in advance of submitting the application, in order to understand CNSC expectations for management systems and quality assurance. This will inform research and development work, with a view to supporting a potential future licence application. (REGDOC-1.1.5, Reference 5-3)
Graded approach	<p>A method or process by which elements such as the level of analysis, the depth of documentation and the scope of actions necessary to comply with requirements are commensurate with:</p> <ul style="list-style-type: none">• the relative risks to health, safety, security, the environment and the implementation of international obligations to which Canada has agreed• the particular characteristics of a nuclear facility or licenced activity. (REGDOC-3.6, Reference 5-20)

4.3 Symbols

None.

5.0 REFERENCES

- 5-1 Canadian Nuclear Safety Commission, REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants, May 2014.
- 5-2 Canadian Nuclear Safety Commission, REGDOC-3.5.3, CNSC Processes and Practices, Regulatory Framework, January 2021.
- 5-3 Canadian Nuclear Safety Commission, REGDOC-1.1.5, Supplemental Information for Small Modular Reactor Proponents, August 2019.
- 5-4 GE Hitachi Nuclear Energy, NEDO-33952, BWRX-300, Darlington New Nuclear Project Preliminary Safety Analysis Report, Chapter 3, Safety Objectives and Design Rules for Structures, Systems and Components, Revision 1, March 2023
- 5-5 IAEA GSR Part 4, Safety Assessment for Facilities and Activities, 2016.
- 5-6 IAEA SSR-2/1, Safety of Nuclear Power Plants: Design, 2016.
- 5-7 IAEA SSG-30, Safety Classification of Structures, Systems, and Components (SSCs) in Nuclear Power Plants, 2014.
- 5-8 Canadian Nuclear Safety Commission, REGDOC-2.4.1, Safety Analysis, Deterministic Safety Analysis, May 2014.
- 5-9 Memorandum of Understanding Between the United States Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission for the Exchange of Technical Information and Cooperation in Nuclear Safety Matters, August 2017.
- 5-10 Memorandum of Cooperation on Advanced Reactor and Small Modular Reactor Technologies Between the United States Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission, August 2019.
- 5-11 Joint Report on GE Hitachi's Containment Evaluation Method, A Collaborative Review by the U.S. Nuclear Regulatory Commission and the Canadian Nuclear Safety Commission, April 2022.
- 5-12 GE Hitachi Nuclear Energy, NEDO-33953, BWRX-300, Darlington New Nuclear Project Preliminary Safety Analysis Report, Chapter 4, Reactor, Revision 0, September 2022.
- 5-13 GE Hitachi Nuclear Energy, NEDO-33956, BWRX-300, Darlington New Nuclear Project Preliminary Safety Analysis Report, Chapter 7, Instrumentation and Control, Revision 0, September 2022.
- 5-14 GE Hitachi Nuclear Energy, NEDO-33955, BWRX-300, Darlington New Nuclear Project Preliminary Safety Analysis Report, Chapter 6, Engineered Safety Features, Revision 0, September 2022.
- 5-15 U.S. Nuclear Regulatory Commission, Regulations, 10 Code of Federal Regulations (CFR) Part 50, Appendix A, General Design Criteria.
- 5-16 U.S. Nuclear Regulatory Commission, Regulatory Guide (RG) 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water Cooled Nuclear Power Plants, Revision 2, November 2001.
- 5-17 GE Hitachi Nuclear Energy, NEDO-33914-A, Licensing Topical Report, BWRX-300 Advanced Civil Construction and Design Approach, Revision 2, June 2022.
- 5-18 ASCE/SEI 43, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, 2019.

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- 5-19 GE Hitachi Nuclear Energy, NEDO-33959, BWRX-300, Darlington New Nuclear Project Preliminary Safety Analysis Report, Chapter 9B, Civil Engineering Works and Structures, Revision 2, March 2023.
- 5-20 Canadian Nuclear Safety Commission, REGDOC-3.6, Glossary of CNSC Terminology, February 2022.
- 5-21 OPG NK054-REP-03500-00001, R002, Independent Peer Review of the Preliminary Safety Analysis Report (PSAR) for the Darlington New Nuclear Project, February 24, 2023.
- 5-22 GE Hitachi Nuclear Energy, NEDO-33965, BWRX-300, Darlington New Nuclear Project Preliminary Safety Analysis Report, Chapter 15, Safety Analysis, Revision 0, September 2022.

A6.0 APPENDIX, MEANS OF SHUTDOWN INDEPENDENCE

The alternative approach for means of shutdown independence has been deleted. It is determined that the BWRX-300 design meets the requirements and guidance for independence of two means of shutdown.

REGDOC-2.5.2 provides specific requirements with respect to independent and diverse means of shutdown. A proposed alternative approach in Revision 0 of this document has been reviewed and determined to be unnecessary, as discussed below. Because the BWRX-300 design conforms with CNSC requirements and guidance, no alternative approach is considered necessary.

An Independent Peer Review (IPR) of the BWRX-300 PSAR, documented in OPG Report NK054-REP-03500-00001, Independent Peer Review of the Preliminary Safety Analysis Report (PSAR) for the Darlington New Nuclear Project (Reference 5-21), was conducted in accordance with REGDOC-2.5.2, Section 5.6, which specifies that: "Before the design is submitted, an independent peer review of the safety assessment shall be conducted by individuals or groups separate from those carrying out the design" (Reference 5-1). The IPR concluded that, based on a review and comparison to CNSC guidance and requirements, the two means of shutdown meet the REGDOC-2.5.2, Section 8.4, independence requirements.

OPG plans to address the IPR report recommendations. The alternative approach report revision is one element of this effort.

A6.1 REGDOC-2.5.2 Requirements and Guidance

REGDOC-2.5.2 provides the following requirements and guidance with respect to defining separate, independent, and diverse means of shutdown.

8.4 Means of shutdown

...

The design shall include two separate, independent, and diverse means of shutting down the reactor.

Section 8.4, Guidance, indicates that:

For the two means of shutting down the reactor to be independent of each other, they do not share components. If both means act inside the core and complete separation is not possible, adequate separation of ex-core components should be demonstrated.

In addition to the REGDOC-2.5.2 requirements, REGDOC-2.4.1, Deterministic Safety Analysis, (Reference 5-8), Section 4.4.4.4, provides guidance for means of shutdown systems that informs this proposed alternative approach. It explains that two broad categories of reactors are considered:

- Reactors with inherent safety: designs that demonstrate that any [anticipated operational occurrence] AOO or [design basis accident] DBA with failure of the fast-acting shutdown means (anticipated transient without reactor trip type analysis) does not lead to severe core damage and a significant early challenge to containment.
- Reactors with engineered safety: designs that cannot demonstrate that any AOO or DBA with failure of the fast-acting shutdown means does not lead to severe core damage and a significant early challenge to containment.

The BWRX-300 design is a reactor with inherent safety, as described in DNNP PSAR, Chapter 4, Reactor (Reference 5-12). The REGDOC-2.4.1 guidance indicates that, for performing a

deterministic safety analysis, such a reactor with inherent safety would use the following approach:

- For the first shutdown means, which is fast-acting, the analysis should demonstrate that the criteria applicable to the initiating event class (AOO or DBA [Anticipated Operational Occurrence or Design Basis Accident], as applicable) are met. Operator actions to supplement the fast-acting shutdown means may be credited, provided that the conditions for manual reactor trip are satisfied.
- For the second shutdown means (that may be manually initiated), the frequency of occurrence of an AOO and the failure frequency of the fast-acting shutdown means may result in a combined frequency that falls in the DBA range, in which case the applicable limits are the DBA dose limits. If the designer can demonstrate a very high reliability for the fast-acting shutdown means, it may be acceptable to use [Beyond Design Basis Accident] BDBA limits (i.e., the safety goals).
- The frequency of a DBA and the failure frequency for the fast-acting shutdown means may result in a combined frequency that falls in the BDBA range, in which case the applicable limits are the safety goals.

As quoted, the guidance indicates that there would be one fast-acting shutdown means and a second shutdown means. This guidance applies to the BWRX-300 design in that it employs a fast-acting scram using hydraulic control units, with a second means of shutdown using a fast motor run-in feature of the control rod drive system. Because the BWRX-300 design meets the guidance for implementing the means of shutdown requirements, the proposed alternative approach is deemed unnecessary.

A6.2 Design Features for Means of Shutdown

GE Boiling Water Reactor (BWR) designs have historically, in over 60-years of operating experience, used a single set of control rods for both reactivity control and reactor shutdown. While there are components within the shutdown systems that are separate and independent, the control rods inside the core are shared between the two shutdown means. However, the BWRX-300 design includes two diverse and separate means of control rod insertion that, while sharing the components within the reactor core (most particularly the control rods), are adequately separated outside the reactor core. By relying on the fast-acting hydraulic control units as the first means of shutdown, and on the electric fast motor run-in insertion as the second means of shutdown, the requirement for two diverse means of shutdown is met, using the same control rods inside the reactor core. This design is consistent with guidance in REGDOC-2.4.1.

Accordingly, no alternative approach is proposed for the independence requirements set forth in REGDOC-2.5.2, Section 8.4, because the current configuration is fully compliant with the specific requirements and guidance.

A6.2.1 Core Design

Design of the BWRX-300 reactor core is described in the DNNP PSAR Chapter 4, Section 4.3.2 (Reference 5-12). As described, the BWRX-300 core design is a reactor with inherent safety features. Consistent with the guidance in REGDOC-2.4.1, Section 4.4.4.4, it is acceptable to employ a fast-acting means of shutdown and a second means of shutdown.

A6.2.2 Control Rods

Details of the control rods design and functional requirements are described in DNNP PSAR Chapter 4, Section 4.2 (Reference 5-12). The control rods are designed to control the fission chain reaction. The control rods, along with the control rod drive system, provide stable and

automatic control of reactor core power during normal operation. The control rods also shut down the reactor and maintain the core subcritical.

A6.2.3 Control Rod Drives

Details of the control rod drives system design and functional requirements are described in DNNP PSAR Chapter 4, Section 4.2 (Reference 5-12). Each Control Rod Assembly (i.e., control rod) is coupled to a Control Rod Drive (CRD) which is used to position the control rod.

A6.2.4 Two Means of Shutdown

The BWRX-300 design is consistent with the guidance in REGDOC-2.4.1, Section 4.4.4.4 (Reference 5-8), for a reactor core with inherent safety, and having two means of shutdown: (1) a fast-acting shutdown and (2) a second means of shutdown. The first means of shutdown to meet the requirements for a fast-acting insertion of the control rods is the hydraulic scram using the hydraulic control system. The second means of shutdown is the CRD electric fast motor run-in.

As described in the sections below, insertion of the control rods includes several diverse and independent control signals and systems.

1. The first means is the Hydraulic Control System, which inserts the control rods rapidly through a hydraulic scram system.
2. The second means is the fast motor-driven ("run-in") insertion of the control rods by means of the CRD motors with Control Rod Drive Mechanism (CRDM) run-in initiation on high flux after scram signal.

The instrumentation and control functions are described in the DNNP PSAR, Chapter 7, Instrumentation and Control, Section 7.1.1 (Reference 5-13).

A6.2.4.1 Control Rod Drive Hydraulic Scram System

In BWR designs, a fast insertion of control rods using stored hydraulic energy is referred to as a scram or hydraulic scram. The force required for hydraulic scram is provided by 29 HCUs that include nitrogen-charged accumulators (for the 57 control rods, [[]]). A hydraulic scram is initiated by opening 29 scram valves, one on each accumulator water discharge path. The Hydraulic Scram Function is described in detail in DNNP PSAR, Chapter 7, Instrumentation and Control, Section 7.3 (Reference 5-13).

A6.2.4.2 CRD Motor Run-In

The Diverse Protection System (DPS) provides the logic for the diverse protection functions, which includes initiation of the CRD motor run-in. As explained in PSAR Section 7.3.2.2, the DPS fast motor run-in is independent and serves as the backup to the mechanical/hydraulic portions of the CRD system.

The CRD fast motor run-in instrumentation and controls, the scram motor follow, and the Uninterruptable Power Supplies (UPSs) are described in DNNP PSAR, Chapter 7, Instrumentation and Control, Section 7.3 (Reference 5-13).

A6.2.5 System and Component Separation Design Features

REGDOC-2.5.2, Section 8.4, guidance indicates that:

For the two means of shutting down the reactor to be independent of each other, they do not share components. If both means act inside the core and complete

separation is not possible, adequate separation of ex-core components should be demonstrated.

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PSAR Section 7.3 (Reference 5-13) describes the separation of controls. [[

]].

The separation of these two diverse means of shutdown (HCU scram and electric fast motor run-in) conforms to the REGDOC-2.5.2 guidance for those cases where there is a lack of independence inside the core, but there is adequate separation outside the core. The power supplies for the 57 CRD electric motors are separated into four groupings (or four power divisions) to minimize the impact of a power supply failure for one group of drives, as described in DNNP PSAR, Chapter 7, Instrumentation and Control, Section 7.3 (Reference 5-13).

A6.3 Safety Analysis for Means of Shutdown

The BWRX-300 safety goals are established as in PSAR Table 15.3-3, Probabilistic Safety Goals, with the core damage frequency derived quantitative acceptance criteria as follows:

The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than $1E-5/rx\text{-}yr$.

To address the REGDOC-2.4.1 guidance for performing a deterministic safety analysis for a reactor with inherent safety, the BWRX-300 fault evaluation indicates that the AOO with the failure of the fast-acting shutdown means is a DEC (BDBA) and safety goals are met.

Analyses presented in the PSAR demonstrate that an AOO with the failure of the fast-acting shutdown means meet the DBA acceptance criteria, which assures safety goals would be met for this set of events. Results of the safety analyses for failure of HCU scram scenarios are described in the following PSAR Section 15.5 (see Reference 5-22):

- PSAR Section 15.5.5.2.1, Closure of One Main Steam Reactor Isolation Valve
 - The analysis assumes a Common Cause Failure (CCF) hydraulic scram failure. The control rods enter the core using the CRDM run-in function. This event demonstrates that the CRDM run-in function performs the Fundamental Safety Function control of reactivity without hydraulic scram. The CRDM run-in initiation is on high flux after scram signal.
- PSAR Section 15.5.5.2.3, Loss of Condenser Vacuum with CCF Hydraulic Scram

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- A CCF results in failure of hydraulic scram. The control rods enter the core using the CRDM run-in function. No other failures are assumed. CRDM run-in initiation occurs on high flux after scram signal.
- PSAR Section 15.5.5.2.4, Loss-of-Preferred Power with CCF Hydraulic Scram
 - A CCF results in failure of hydraulic scram. The control rods enter the core using the CRDM run-in function. No other failures are postulated in the event. CRDM run-in initiation is on high flux after scram signal.

Regarding the frequency of a DBA and the failure frequency for the fast-acting shutdown means, the REGDOC-2.4.1 guidance indicates that the second shutdown means may result in a combined frequency that falls in the BDBA range, in which case the applicable limits are the safety goals. For the BWRX-300 design, the DBA combined with a failure of the fast-acting shutdown is an even lower frequency than the AOO with a failure of the fast-acting shutdown. This combination would also be a BDBA and the applicable criteria, per the guidance of REGDOC-2.4.1, would be that the safety goals are met.

A6.4 Summary

As explained above, the separation of these two diverse means of shutdown (HCU scram and electric fast motor run-in) conforms to the REGDOC-2.5.2 guidance for those cases where there is a lack of independence inside the core, but there is adequate separation outside the core.

Separate, independent, and diverse control systems for means of shutdown using control rod insertion are described in more detail above in the referenced DNNP PSAR documents.

- Hydraulic scram is the fast rod insertion.
- Diverse Protection System electrical motor CRD run-in.

A6.5 Conclusions

Based on the information presented, for the current phase of design and analyses, the two means of shutdown (fast-acting scram by HCUs and second means of electric fast motor run-in of the CRD system) provide an acceptable level of safety in accordance with CNSC guidance and meet the stated requirement for two independent means of shutdown for an inherently safe reactor core. The controls rods are not independent inside the reactor core, but the two means of shutdown are adequately separated outside the reactor. Based on REGDOC-2.4.1 and REGDOC-2.5.2 guidance, the implementation of the guidance demonstrates that the BWRX-300 design meets the specific requirements for two independent means of shutdown.

4.7.3 The Operational Radiation Protection Program

The future radiation protection program for DNNP will provide the necessary oversight and control to ensure occupational and public exposure to radiation during operations remains ALARA, taking into account social and economic factors. Per regulatory requirements, the radiation protection program will also include a set of action levels for radiation doses or other parameters that, if reached, may indicate a loss of control of part of the program and trigger a requirement for specific action to be taken.

The following will be included in the future radiation protection program for DNNP:

- Management control over work practices;
- Personnel qualification and training;
- Control of occupational and public exposure to radiation;
- Planning for unusual situations;
- Organization and administration for radiation protection
- Radiation protection training and qualification
- Classification of radiation zones
- Radiation exposure and dose control
- Worker dose management
- Radiation protection equipment and instrumentation
- Contamination control
- Radiation protection program oversight
- Dose to the public

4.7.4 Projected BWRX-300 Occupational Exposures

Based on the BWRX-300 conceptual design, the annual collective occupational dose during the operations phase is estimated to be approximately 490 person-mSv. This estimate is less than the 1 person-Sv annual collective dose target OPG established for this reactor, due to the design features that support the ALARA principle. It is also

relatively low compared to typical annual collective doses at currently operating BWRs and is expected to be further reduced as additional design details are confirmed. Individual doses will be maintained below the regulatory dose limits for Nuclear Energy Workers and will be estimated prior to the operations phase when information such as anticipated staffing levels is established.

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**INDEPENDENT PEER REVIEW OF THE PRELIMINARY SAFETY ANALYSIS REPORT (PSAR)
FOR THE DARLINGTON NEW NUCLEAR PROJECT**

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**Independent Peer Review of the
Preliminary Safety Analysis Report
(PSAR) for the Darlington New Nuclear
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Executive Summary

An independent peer review (IPR) of the BWRX-300 Preliminary Safety Analysis Report (PSAR) Rev. 0 was completed to:

- Increase the level of Ontario Power Generation's (OPG) confidence that an adequate level of safety has been achieved for the BWRX-300 design to support the "Licence to Construct" (LTC) application, and
- Satisfy the Canadian Nuclear Safety Commission (CNSC) requirement for an independent peer review of the safety assessment of the selected design, consistent with Clause 5.6 of REGDOC-2.5.2, "Design of Reactor Facilities".

To achieve these objectives, OPG engaged both external subject matter experts in the nuclear industry, and internal OPG Nuclear Safety Division staff that have not been otherwise involved in the Darlington New Nuclear Project (DNNP).

The IPR scope was focused on a review of the PSAR, particularly Chapter 15, and was conducted against the regulatory requirements and guidance in CNSC Regulatory Documents, particularly REGDOC-2.5.2 and REGDOC-2.4.1. Where additional information or clarifications are found outside of the PSAR, they are generally considered out of scope. It is acknowledged that resolution of IPR review comments may be available in current or planned supplementary documents.

BWRX-300 embodies engineered, passive and inherent safety characteristics that provide excellent safety. These include multiple means of reactor shutdown, inherently safe features (strongly negative neutronic feedback from coolant voiding and fuel temperature), and an excellent operating track record for BWRs. Advanced features in BWRX-300 include passively driven core flows to eliminate the need for core internal recirculation jet pumps, and significant simplification in the number of systems, components, pipes and valves. The IPR team also notes that the Defence Line 2 (DL2) systems in BWRX-300 are more comprehensive than comparable safety features in the CANDU. In addition, the Isolation Condenser System (ICS) and Reactor Pressure Vessel (RPV) isolation¹ provide a unique means for long term core cooling in the event of a Loss of Coolant Accident (LOCA) as compared to other reactor designs which require complex engineered Emergency Core Cooling (ECC) measures. Finally, the BWRX-300 passive containment cooling design is expected to improve the reliability of containment for many accidents. This IPR report highlights many of these positive features in Section 4.0.

The IPR team identified the following broad categories as potential areas of concern:

1. "Newness" factors associated with Canadian licensing requirements,
2. Safety analysis gaps in PSAR Rev. 0,

¹ Reliability of these new RPV isolation valves (First of a Kind) is to be demonstrated.

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3. “Newness” factors associated with “First-Of-A-Kind” (FOAK) approaches in the BWRX design, and
4. Incorporation of lessons learned from the 2011 accident at the Fukushima Daiichi power plant.

The following points summarize the IPR team’s conclusions in these areas:

- The licensing process for the BWRX-300 reactor for DNNP represents the first opportunity to put into practice the technology neutral REGDOCs for a new power reactor application. The CNSC requirements and guidance in the new REGDOCs differ from those used to license CANDU reactors in the past. For example, in the past, strict independence of process, control and safety systems, and among safety systems, has been a Canadian regulatory requirement. Given that BWRX-300 shares the same control rods for normal reactor power control and accident mitigation, BWRX-300 would have been non-compliant with past Canadian regulations, even though current Canadian requirements (e.g., Clauses 7.6.1 and 7.6.5 of REGDOC-2.5.2) now allow limited sharing of process/safety equipment between levels of defense and among safety systems.
- Similarly, because of the positive void reactivity coefficient inherent in CANDU reactors, Canadian reactor licensing has historically required two Shutdown Systems, e.g., see [R-9]. Instead, REGDOC-2.5.2 Clause 8.4 requires two appropriately redundant and diverse means of shutdown but does not presume that every candidate design will have reactivity characteristics that require two fully effective and independent Shutdown Systems. The BWRX design combines an inherently safe reactor design with appropriately redundant and diverse engineered means of shutdown that include:
 - the automatic DL2 anticipatory trip functions to mitigate AOOs;
 - automatic, triplicated, single-failure-proof DL3 hydraulic scram for DBAs – considered to be the fast-acting shutdown means by the IPR team;
 - automatic DL4a fast control rod insertion via electric motor that uses independent sensors/logic and in addition an alternate hydraulic system initiation (ARI) – considered to be the second shutdown means by the IPR team; and
 - a manually operated DL4b Boron Injection System (BIS) for extremely low probability Beyond Design Basis Accident (BDBA) sequences.

Except for the BIS, all of these means use the same 57 control rod blades. However, sharing of these in-core components is explicitly allowed in REGDOC-2.5.2 Clause 8.4. A strong case can be made that the BWRX-300 means of shutdown described may already be compliant with the relevant requirements in REGDOC-2.5.2. The CN-DBA analysis in Chapter 15 demonstrates effectiveness of only the fast-acting shutdown means (DL3). Table 2 in Section 3.5.1 of this report summarizes potential compliance gaps associated with this issue and is followed by recommendations to address them. Provided that the gaps identified in Table 2 are satisfactorily addressed (primarily through documentation updates and additional safety analysis), the IPR Team views that BWRX-300 means of shutdown not only meet but also exceed the REGDOC-2.5.2 and REGDOC-2.4.1 shutdown requirements and guidance for inherently safe reactors. An

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analysis of overall shutdown reliability can also help demonstrate that the intent of the shutdown requirements is also met.

- Both REGDOC-2.5.2 and REGDOC-2.4.1 require one or two diverse trip parameters for each shutdown means. The inherently safe BWRX-300 design must demonstrate that the trip coverage requirements of REGDOC-2.5.2 and 2.4.1 (shown in Table 1 and Table 2 of this report) are met in the design or via a documented alternative approach to the REGDOCs utilizing Clause 11 of REGDOC-2.5.2 and/or Graded Approach in REGDOC-2.4.1. The IPR team recommends that the CN-DBA analysis in Chapter 15 provides documented justification for the first trip parameter to be a direct trip parameter separately for each of the credited shutdown means (DL3 and DL4a), or otherwise is carried out in such a manner that the first trip signal in each of the two shutdown means is not credited to shutdown the reactor. Additionally, consideration should be given to credit only the second trip parameter for each shutdown means for events with positive pressure feedback that involve simultaneous positive reactivity insertion and impaired heat removal. Provided that the gaps identified in Table 2 are satisfactorily addressed (primarily through documentation updates and additional safety analysis), the IPR team views that BWRX-300 trip parameter coverage meets or exceeds REGDOC-2.5.2 and REGDOC-2.4.1 requirements and guidance for inherently safe reactors.
- PSAR R0 provides no details on many of the complementary design features and safety-grade (DL4b) water injection which may be required for unforeseen events, or for inventory make-up for unisolable leaks or small loss of coolant accidents (DBA) in the longer term. Chapter 15.6 of the PSAR provides only a high-level description of many important functions (such as CRD purge water injection). It is not clear if the DL4b injection function is functionally and physically separated from the systems intended for other DL functions, defense lines or whether it uses diverse and flexible equipment and portable components.
- An important gap identified by the team relates to small pipe breaks with loss of preferred power and the need for emergency coolant makeup. The analysis in Chapter 15 of the PSAR concludes that fuel is adequately cooled for the first 3 days of the event. However, the analysis terminates with significantly reduced (collapsed) water level below the Top of Active Fuel in the downcomer. The analysis should be extended to summarize the steps needed to bring the reactor to an acceptable end state; restoration of water level in the RPV and leak isolation. For DEC sequences, this includes the PSAR discussing the potential for severe core damage occurring between 72 hours and 7 days (after which CRD makeup can be credited). The later and concluding stages of the accident progression need to be better described in the PSAR. In addition, for breaks within containment, if continuous water injection to the RPV is required, continuous water discharge through the break into containment may pose containment integrity challenges or equipment operability issues due to flooding in the containment. Furthermore, once the RPV and containment pressures equalize, there is a potential for non-condensable nitrogen gas to enter the RPV and impair the effectiveness of ICS heat removal capability.
- The BWRX design employs several FOAK design features. These include Steel Bricks™ and reactor pressure vessel (RPV) isolation valves, and ICS condensate discharge location into the chimney. The IPR Team concludes that additional analysis and qualification testing will be required to confirm these FOAK features as acceptable design approaches.

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Although RPV isolation is identified here as a FOAK and hence an area of potential concern, the IPR team has also identified it as a strength. Current Canadian nuclear plant designs include high-capacity ECCS pumps or accumulators to restore cooling to uncovered/uncooled fuel after postulated pipe breaks. The alternative approach in BWRX-300 is to start with a high heat transport water inventory in the RPV and to preserve adequate water in the core using RPV isolation valves that close following postulated failure of piping. It is intuitively preferable to not uncover fuel during postulated accidents by starting with more coolant and limiting its loss. This approach avoids the difficulty of rewetting hot surfaces after inventory loss and fuel heat up; it also reduces concerns associated with recovering, circulating outside containment and re-injecting large volumes of water that would spill indefinitely from breaks that are not isolated (e.g., recovery strainer plugging issues).

- Finally, the possibility of severe core damage leading to containment failure and large release is an area of potential concern. For DEC, BDBA and severe accident sequences, BWRX-300 incorporates many, significant improvements for emergency cooling of the core as well as for containment emergency cooling and venting. With these improvements, plant states leading to significant release of radioactivity have been practically eliminated, as is required by Clause 7.3.4 of REGDOC-2.5.2. Nonetheless, given the significant impact of the Fukushima accident, the BWRX-300 design features need to be summarized to demonstrate that a repeat of an accident like Fukushima is precluded.

Overall, the IPR team concludes that PSAR Rev. 0 provides sufficient detail to allow meaningful assessment of BWRX-300 safety design features. The IPR team also identified throughout this report numerous gaps, areas of further work and recommendations mostly in the safety analysis and documentation. These are documented in Section 5.0, Table 2, Table 3 and Appendix B, and can be addressed in an updated Pre-Operational Safety Analysis Report (POSAR) or Final Safety Analysis Report (FSAR). Further, the team concludes that GE-H has successfully adapted established BWR design approaches to meet the new Canadian licensing requirements. The plant safety features described and analyzed in PSAR Rev. 0 demonstrate that the BWRX-300 design is safe, and PSAR Rev. 0 provides sufficient detail to support a Licence-to-Construct application.

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AOO	Anticipated Operational Occurrence
ATR	Advanced Test Reactor
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design Basis Accident
BL-	Baseline (Deterministic Safety Analysis)
BLW	Boiling Light Water
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium Uranium
CCF	Common Cause Failure
CHF	Critical Heat Flux
CIS	Containment Inerting System
CN-	Conservative (Deterministic Safety Analysis)
CNSC	Canadian Nuclear Safety Commission
CPR	Critical Power Ratio
CR	Control Rod
CSA	Canadian Standards Association
CVC	Compliance Verification Criteria
DBA	Design Basis Accident
DEC	Design Extension Conditions
DiD	Defence in Depth
DNNP	Darlington New Nuclear Project
DSA	Deterministic Safety Analysis
ECCS	Emergency Core Cooling System
EIS	Environmental Impact Statement
EOC	End of Cycle
EOR	End of Rated
EX-	Extended (Deterministic Safety Analysis)
FDC	Fuel Damage Category

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FLE	Fuel Load Error
FMCRD	Fine Motion Control Rod Drive
FOAK	First Of A Kind
FOM	Figure of Merit
FSAR	Final Safety Analysis Report
FW	Feedwater
GE-H	General Electric – Hitachi Nuclear Energy
GNF	Global Nuclear Fuels
HCU	Hydraulic Control Unit
HDFS	High Density Fuel Storage
IC	Isolation Condenser
ICRW	Inadvertent Single Control Rod Withdrawal at Power
ICS	Isolation Condenser System
IE	Initiating Event
IPR	Independent Peer Review
LCH	Licence Conditions Handbook
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LRF	Large Release Frequency
LTC	License to Construct
MCCI	Molten Corium Concrete Interaction
MOC	Middle of Cycle
MOX	Mixed Oxide
MS	Main Steam
NPP	Nuclear Power Plant
NRX	National Research Experimental
NSCA	Nuclear Safety and Control Act
OLC	Operating Limits and Conditions
OOCC	Out-of-Core Criticality
OPEX	Operating Experience

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OPG	Ontario Power Generation
PARS	Passive Autocatalytic Recombiner System
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PCV	Primary Containment Vessel
PIE	Postulated Initiating Event
POSAR	Pre-Operational Safety Analysis Report
PROL	Power Reactor Operating License
PRSL	Power Reactor Site Preparation License
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PHWR	Pressurized Heavy Water Reactor
QA	Quality Assurance
REGDOC	Regulatory Document
RLC	Reactor Level Control
RPC	Reactor Pressure Control
RPV	Reactor Pressure Vessel
SA	Severe Accident
SCCV	Steel-Plate Composite Containment Vessel
SCDF	Severe Core Damage Frequency
SFP	Spent Fuel Pool
SGHWR	Steam-Generating Heavy Water Reactor
SMR	Small Modular Reactor
TAF	Top of Active Fuel
TBV	Turbine Bypass Valve
TSV	Turbine Stop Valve
TT	Turbine Trip
USL	Upper Subcritical Limit
US NRC	United States Nuclear Regulatory Commission

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1.0 INTRODUCTION

Ontario Power Generation (OPG) holds a Nuclear Power Reactor Site Preparation Licence (PRSL18.00/2031) for the Darlington New Nuclear Project (DNNP) located on the Darlington site in the Municipality of Clarington, in the Regional Municipality of Durham. In December 2021, OPG announced its selection of the technology for deployment at that site: a 300 MWe direct cycle light water-cooled, natural circulation Small Modular Reactor (SMR) designed by General Electric Hitachi (GE-H) Nuclear Energy, denoted as BWRX-300 [O-1], [O-2].

To support DNNP activities for obtaining a Licence to Construct (LTC) this new reactor, OPG has performed a limited-scope independent peer review (IPR) of “Preliminary Safety Analysis Report (PSAR) Rev. 0” [G-1], in alignment with the regulatory requirements from REGDOC-1.1.2, “Licence Application Guide: Licence to Construct a Reactor Facility” [R-1], and REGDOC-2.5.2, “Design of Reactor Facilities” [R-2] and the OPG guide [O-3].

The purpose of this report is to summarize the results and key findings of this independent peer review of the safety case (“safety assessment” presented by the designer). The reviewers involved in this report have not been previously involved in design selection for DNNP or OPG’s review and acceptance of associated design documents or other deliverables.

2.0 OBJECTIVES, SCOPE AND METHODOLOGY

2.1 IPR Objectives

The objectives of the IPR are twofold. The primary objective of this IPR is to increase the level of OPG confidence that an adequate level of safety has been achieved for the BWRX-300 design to support a licence to construct application.

The second objective satisfies the requirement in Clause 5.6 of CNSC REGDOC-2.5.2 [R-2]:

“Before the design is submitted, an independent peer review of the safety assessment shall be conducted by individuals or groups separate from those carrying out the design.”

2.2 IPR Scope

The scope of the independent peer review team is to review the Preliminary Safety Analysis Report, Rev. 0, with a focus on the Safety Analysis described in Chapter 15. The content of the PSAR was evaluated at a high level against the requirements and guidance provided in the following regulatory documents²:

- REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants³ [R-2],
- REGDOC-2.4.1, Deterministic Safety Analysis [R-3],

² Including complementary CSA Standards

³ Specific sections of REGDOC-2.5.2 considered by IPR Team are listed in Section 3.4

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- REGDOC-2.4.2, Probabilistic Safety Assessment for Reactor Facilities [R-4], and
- REGDOC-2.4.3, Nuclear Criticality Safety [R-5].

Although the PSAR Rev. 0 was the main document utilized by the IPR team to perform this review, some other documents from the US-NRC or GE-H licensing topical reports for BWRX-300 were used to understand the technology or fill in some of the information gaps in the text of the PSAR. In addition, at the time of writing this report, GE-H has begun to issue topical reports to address issues that are important and unique for Canadian licensing. One such report (Reference [G-14]) addresses the acceptability of alternative design approaches. This report has not yet been fully incorporated into the PSAR. However, it is relevant to IPR activities and conclusions. Therefore, it was included in IPR team reviews.

Various IPR team members also performed specific portions of the review that are compiled in this report. These include an accident-by-accident review of the postulated initiating events and the results described in the PSAR, an assessment of the core stability analysis, a specific review of the containment design and response, and an evaluation of the methodology presented in the out-of-core criticality safety assessment. A limited review of the BWR Operating Experience (OPEX) is also considered as part of the scope of the IPR team review.

2.2.1 Limitations of IPR Scope

A clause-by-clause compliance review of Regulations, Regulatory Documents, or CSA standards is outside the IPR scope. While the design process itself and compliance with design quality standards are not within the scope for this review, some high-level comments are provided on design features important to nuclear safety. A review of the computer code applicability and validation of the safety analysis codes utilized in preparation of the PSAR is also not in the scope of IPR unless the information is included already in the PSAR itself. While beyond the scope, a finding on code applicability and validation is provided in Section 5.5.

It is acknowledged that, in many instances, the PSAR is a high-level summary of other documents that have been referenced within the PSAR. It is important to emphasize that the IPR team scope of work included only the information presented within the PSAR, with a specific focus on Chapter 15: Safety Analysis. The IPR team did not perform an exhaustive review of other PSAR Chapters and references.

As such, there may be instances when the IPR team comments may be addressed via other related documents that have been prepared or are being prepared. If the IPR team reviewed specific PSAR references, they have been explicitly noted within this report and their contents reflected in the IPR team's review. Otherwise, the IPR team review is based solely on the content presented within PSAR Rev. 0, with a focus on the content of Chapter 15.

2.3 IPR Methodology

The methodology adopted for the independent peer review includes:

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- Forming a team of internal and external experts to conduct the IPR. This team is independent of the OPG team involved in review and acceptance of deliverables for DNNP,
- Familiarization with and review of the features of the GE-Hitachi BWRX-300 reactor,
- Reviewing the results of the safety analysis against the requirements and guidance in Canadian regulatory documents, with a focus on the Safety Analysis methodology and results in Chapter 15 of the Preliminary Safety Analysis Report (PSAR) Rev. 0 [G-1],
- Identifying findings, areas of improvement or areas requiring clarification,
- Identifying strengths and key items that require further work (areas for improvement), and
- Preparing a report summarizing the findings (this document).

Because of the tight timelines for conducting this review, the IPR team performed an in-depth first round of review on a draft version of the PSAR (Rev. A) which was available at the time of IPR team formation in Q2-2022. Subsequent to this review, Revision 0 of the PSAR [G-1] was issued and submitted to the CNSC as part of the LTC application. The IPR team then re-assessed its review based on the updated contents of PSAR Rev. 0 [G-1]. The result of this second-round review is presented herein, consistent with the IPR plan.

During the review of PSAR Rev. 0, the IPR team had an opportunity to have three topical meetings with GE-H to discuss: 1) PSA and Acceptance Criteria, 2) Thermalhydraulics, Reactor Physics and Stability, and 3) Alternate Approaches and Code Accuracy and Applicability. Via these discussions, clarifications were provided and some of the original IPR team comments on PSAR Rev. A were resolved. Although GE-H may have work in progress to address some of the IPR review comments, if a comment is not currently explicitly addressed within the current PSAR Rev. 0 document, the IPR team has included a review comment within this document. The IPR team also acknowledges that certain parts of the design, safety analyses or safety analysis codes may have been accepted or approved for use by the US NRC but have not taken that into consideration as part of its review.

3.0 CANADIAN NUCLEAR SAFETY REGULATORY FRAMEWORK

3.1 Nuclear Safety Control Act

The Nuclear Safety and Control Act (NSCA) is a federal law passed by Parliament that came into force on May 31, 2000 and established the Canadian Nuclear Safety Commission (CNSC). The NSCA establishes the statutes that define the CNSC's mandate, responsibilities, and powers.

The Act provides the CNSC with the authority to regulate the development, production and use of nuclear energy, the possession and use of nuclear substances, prescribed equipment, and prescribed information, for the protection of the health, safety, and security of Canadians

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and the environment. Figure 1 is a basic illustration of the CNSC's regulatory framework consisting of laws passed by Parliament that govern the regulation of Canada's nuclear industry. The CNSC in-turn, will issue regulations, licences & certificates, and regulatory documents it uses to regulate the industry. More details can be found on the CNSC website (<https://www.cnsccsn.gc.ca/>).

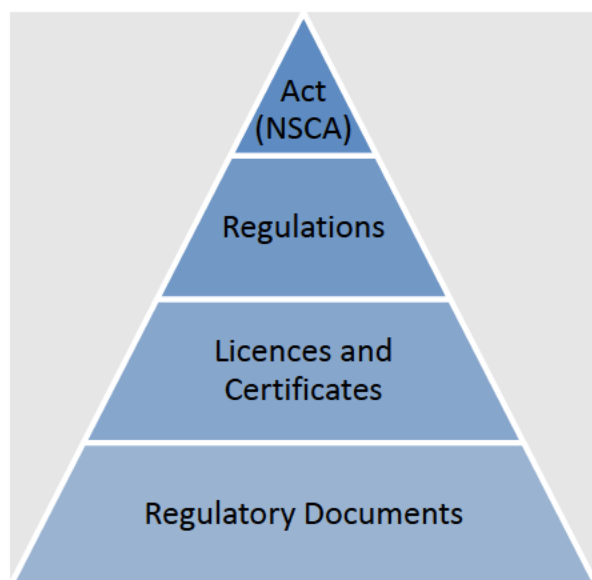


Figure 1: Canadian Regulatory Framework

The Canadian Standards Association (CSA Group) also issues Standards, developed through a consensus standards development process that supports safe and reliable nuclear power generation in Canada. CSA Standards often refer to other international standards for more detailed requirements. These standards are complementary and integrated into the national regulatory framework, and are often referenced in regulatory documents, licenses, and license conditions handbooks for nuclear facilities across Canada.

3.2 Nuclear Power Plant Licensing

In Canadian nuclear power plant licensing, general requirements for a proposed plant design of a new facility are tied to detailed requirements for supporting safety analysis for that design. Two broad types of safety analysis, namely deterministic safety analysis (DSA) and probabilistic safety assessment (PSA), are needed. Requirements and guidance for the deterministic analysis and probabilistic safety assessment for nuclear power plants are provided in REGDOC-2.4.1 [R-3] and REGDOC-2.4.2 [R-4], respectively. CNSC updates their REGDOCs periodically, and the latest formally issued versions are used for this IPR, e.g., [R-1] to [R-5].

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Regulatory documents generally include two kinds of information: requirements and guidance. When included in the licensing basis, requirements shall be met by any licensees wishing to obtain (or retain) a licence or certificate to use nuclear substances or to operate a nuclear facility. Regulatory documents may allow for alternate approaches or graded application of the requirements to be used in satisfying explicit requirements.

Guidance provides clarification to licensees and applicants on how to meet the requirements. It also provides more information about approaches used by CNSC staff during the review of licence applications. Licensees are expected to review and consider guidance in CNSC REGDOCs; if they choose not to follow it, they should explain how their selected approach still meets regulatory requirements.

3.3 Satisfying Regulatory Requirements

As indicated above, one of the objectives of the IPR is to independently assess the BWRX-300 design and safety analysis within the CNSC regulatory framework. As Canada embarks on expanding its nuclear fleet beyond the CANDU design, the CNSC has endeavoured to define licensing requirements in “technology neutral” REGDOCs for new water-cooled reactor facilities.

In Canada, previous licensing experience is with CANDU reactor designs. OPG has only operated CANDU Pressurized Heavy Water Reactors (PHWRs) and is in the process of learning and adapting to the many differences in the BWRX-300 design philosophy.

The starting point for this IPR was to assess direct (literal) compliance with REGDOC requirements. Any gap to direct compliance is identified as a potential gap and recorded as a finding in this IPR report. If the IPR reviewer concludes that the safety assessment includes a defensible “alternative approach” per REGDOC-2.5.2 Clause 11, that is also noted in Section 5.0 of this report.

This report assesses the content of the BWRX-300 PSAR per the requirements as written in the REGDOCs but recognizes that the focus of PSAR is on establishment of the design-basis requirements for the items important to safety, and further details are to be provided in the Final Safety Analysis Report.

3.4 REGDOC-2.5.2 *Design of Reactor Facilities*: Nuclear Power Plants

REGDOC-2.5.2 describes the requirements and guidance for the design of new water-cooled reactor facilities, with the general nuclear safety objective that, *“the reactor facility will be designed and operated in a manner that will protect individuals, society, and the environment from harm”*. Meeting this objective relies on the establishment and maintenance of effective defenses against the (unique) radiological hazards that materialize during the entire life cycle of operating a nuclear power plant.

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REGDOC-2.5.2 defines technical safety objectives and concepts that pertain to design and safety analysis, among them are the establishment of dose acceptance criteria and qualitative and quantitative safety goals. Safety concepts like defense-in-depth (establishing the five levels of defense), physical barriers, and establishing operating limits and conditions (OLCs) are instituted in the REGDOC.

The safety analyses examine the performance of the plant for the following defined states: normal operation, anticipated operational occurrences (AOO), design basis accidents (DBA), and design extension conditions (DEC). DECs are a subset of beyond-design-basis accidents (BDBA). The results of the safety analysis should confirm the capability of the design to withstand postulated initiating events and demonstrate the effectiveness of the defense line functions.

Note that only the safety analysis-related sections of REGDOC-2.5.2 have been the focus in this review, namely the requirements and guidance in the following clauses:

- 7.1 Safety classification of structures, systems and components⁴,
- 7.3 Plant States
- 7.4 Postulated initiating events,
- 8.1 Reactor core,
- 8.2 Reactor coolant system
- 8.4 Means of shutdown
- 8.4.1 Reactor trip parameters,
- 8.5 Emergency core cooling system,
- 8.6 Containment,
- 8.7 Heat transfer to an ultimate heat sink, and
- 8.8 Emergency heat removal system.

A literal interpretation of these clauses assessed versus the BWRX-300 design, without the benefit of Clause 11, Alternative Approaches, has inevitably led the IPR team to find potential compliance gaps against the REGDOC-2.5.2 requirements, as detailed in Section 5.2 and Appendix B of this report.

It is also noted that BWRX-300 design embodies specific REGDOC-2.5.2 recommendations, e.g., Clause 6.3 “Accident prevention and plant safety characteristics” in [R-2] which states:

“The design shall apply the principles of defence in depth to minimize sensitivity to PIEs. Following a PIE, the plant is rendered safe by:

- 1. inherent safety features*
- 2. passive safety features*
- ...*

With inherent and passive safety design features, the safety analysis of the BWRX-300 design shows a remarkable response to many PIEs. Given these features, there is strong evidence

⁴ Including guidance from IAEA Safety Guide No. SSG-30 “Safety Classification of Structures, Systems and Components in Nuclear Power Plants” [M-8]

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that the overall design can meet the regulatory requirements with respect to safety standards, although direct compliance with regulatory requirements may not be practical for all design features. Hence, the IPR team believes GE-H and OPG should prepare a detailed justification for the alternative approach and explicitly request alternative consideration to the clauses in the regulations early in the licensing process. Section 3.4.1 can be used as a starting point to make the overall safety case for alternative approaches.

3.4.1 Use of Alternative Compliance Strategies

To date, all power reactors in Canada have been Canadian Deuterium Uranium (CANDU) designs licensed in accordance with CANDU-specific design and safety requirements.

In recent years, many key CNSC regulatory documents (REGDOCs) have been revised to make them “technology neutral” and, thereby, facilitate licensing of non-CANDU designs. In many cases, the new REGDOCs explicitly allow proven Boiling Water Reactor (BWR) approaches that are safe and appropriate for BWRs, but which would not have been compliant with previous licensing rules that were developed specifically for CANDUs.

In addition to being a technology neutral licensing document, REGDOC-2.5.2 allows “alternative approaches”. Clause 11 of REGDOC-2.5.2 outlines the steps that a licensee needs to take to qualify an alternative approach that is not fully REGDOC compliant. These steps are described in the following excerpt:

“The requirements in this regulatory document are intended to be technology neutral for water-cooled reactor designs. It is recognized that specific technologies may use alternative approaches.

The CNSC will consider alternative approaches to the requirements in this document where:

- 1. the alternative approach would result in an equivalent or superior level of safety*
- 2. the application of the requirements in this document conflicts with other rules or requirements*
- 3. the application of the requirements in this document would not serve the underlying purpose, or is not necessary to achieve the underlying purpose*

Any alternative approach shall demonstrate equivalence to the outcomes associated with the use of the requirements set out in this regulatory document.”

The IPR team recommends that OPG and GE-H perform a detailed clause-by-clause review of REGDOC-2.5.2 requirements and identify potential gaps to compliance to be closed via the application of Clause 11, where appropriate. The basis for an alternative approach, for each potential gap that cannot be practically closed, should be documented for CNSC review and acceptance. A similar approach can be followed for REGDOC-2.4.1 and also for CSA Standards that are referenced within REGDOC-2.4.1 or REGDOC-2.5.2, where direct compliance with the requirements cannot be readily shown.

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At the time of writing this report, GE-H has just begun the process of compiling relevant alternative approaches and documenting their justifying basis (Reference [G-14]). Reference [G-14] does not address all “alternative approach” issues for BWRX-300. However, it is an important first step that explicitly addresses the review recommendation made in this report. Specifically, [G-14] summarizes the process that GE-H will use to justify alternative approaches and then applies that process to specific alternative approaches for means of shutdown, containment isolation and radioactive waste system design.

3.5 **REGDOC-2.4.1 *Deterministic Safety Analysis***

REGDOC-2.4.1 Part I describes the requirements and guidance for the preparation and presentation of the deterministic safety analysis that demonstrates the safety of an NPP. Safety analysis is an analytical, quantitative study performed to demonstrate the safety of an NPP, and the adequacy of its design and performance. The primary objective of deterministic safety analysis included in the PSAR is to confirm that the design of an NPP meets design and safety requirements.

REGDOC-2.4.1 also details the application of the safety concepts introduced in REGDOC-2.5.2 to performing safety analysis, including defining the levels of defense in depth and the aims of each defense level. The REGDOC also details the requirements for analysis to be performed for the various plant states as shown in Figure 2 below.

Operational states		Accident conditions		
Normal operation	Anticipated operational occurrence	Design-basis accident	Beyond-design-basis accidents	
			Design-extension conditions	Practically eliminated conditions
			No severe fuel degradation	Severe accidents
Design basis		Design extension		Not considered as design extension
Reducing frequency of occurrence →				

**Figure 2: Defined Plant States for Deterministic Safety Analysis
(from [R-2]/[R-3])**

Section 4 of REGDOC-2.4.1 provides the details of the requirements for DSA and guidance for meeting them, with specific clauses for a systematic process for event identification and classification, establishing acceptance criteria for each classified postulated initiating event in terms of dose limits and derived acceptance criteria, and prescribes analysis methodology and where conservative or best-estimate assumptions are to be used. Table 3 includes an

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overall summary of how the safety analyses included in PSAR Rev. 0 compare against REGDOC-2.4.1 requirements.

Responsibilities of a Licensee under the Canadian Regulatory framework include maintaining the capability to perform or procure safety analysis, establishing a formal process to assess and update the safety analysis, and most critically, establishing a formal quality assurance process (QA) that meets the QA Standards in CSA N286.7-16, *Quality Assurance of Analytical, Scientific and Design Computer Programs* [R-6]. To this end, Clause 4.4.5 in the REGDOC provides specific guidance for the use, development, and validation of computer codes used in safety analysis. A demonstration of code applicability and code accuracy is expected for each postulated accident analyzed for BWRX-300. The PSAR contains only a high-level summary for code qualification and refers to GE-H proprietary reports which have not been included within the scope of the IPR.

The findings contained in this report have documented the degree to which the deterministic safety analysis presented in the PSAR meets the requirements and guidance in REGDOC-2.4.1. The postulated accident-by-accident review summarized in Section 6.1 and in Appendix A of this report provides a summary of the initiating event, the accident progression, and the credited defense line functions and trip/scram parameters used. Suggestions for improvements to the analysis presented in the PSAR are also included in Section 6.1.

Similar to REGDOC-2.5.2, REGDOC-2.4.1 is technology neutral and allows for a graded approach when applying the requirements and guidance in the document. The REGDOC explicitly states that:

“A graded approach, commensurate with risk, may be defined and used when applying the requirements and guidance contained in this regulatory document. The use of a graded approach is not a relaxation of requirements. With a graded approach, the application of requirements is commensurate with the risks and particular characteristics of the facility or activity.

An applicant or licensee may put forward a case to demonstrate that the intent of a requirement is addressed by other means and demonstrated with supportable evidence.”

Finally, trip coverage for shutdown means is a particularly important topic for the DSA. In this context, “trip coverage” refers to the diversity and redundancy of means of shutdown and to the diversity and redundancy of the trip parameters, instrumentation, and logic that the design uses to initiate shutdown.

Detailed requirements and guidance for DSA trip coverage for shutdown means are provided in both REGDOC-2.5.2 and REGDOC-2.4.1. BWRX-300 is considered a reactor with inherent safety as described in REGDOC-2.5.2 and as described in Chapter 3 of PSAR. Table 3 of REGDOC-2.4.1 which is partially reproduced below as Table 1 presents the trip coverage expectations for reactors with inherent safety.

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**Table 1: Performance Objectives for DSA Trip Coverage
for Reactors with Inherent Safety (from Table 3 in REGDOC-2.4.1)**

Reactor design	Means of shutdown (SD)	Ideal trip parameter (TP) expectation	Is a direct trip parameter available?	Performance objective	Trip parameter total
Reactors with inherent safety	One fast-acting SD means	One direct TP per event	Yes	One direct TP per event	One TP
			No	Two diverse indirect TPs per event	Two TPs
	Second SD means	One direct TP per event	Yes	One direct TP per event	One TPs
			No	Two diverse indirect TPs per event	Two TPs

Notwithstanding the above, CNSC expectations provided for the second shutdown means in this table may not be applicable to the second shutdown means. "Guidance for shutdown means for reactors with inherent safety" in Clause 4.4.4.4 of REGDOC-2.4.1 can be interpreted to suggest that the second (non-fast-acting) shutdown means is not required to comply with the Public Dose limits, and by extension the Derived Acceptance Criteria for DBAs. This is conditional on the reliability of the fast-acting shutdown means and the initiating event frequency. If the event frequency combined with the failure probability of the fast-acting shutdown means falls below $1\text{E-}5$ /yr, then the applicable limits for analysis become safety goals. For safety goals, best-estimate analysis with realistic modelling and without application of conservative analysis rules such as the SFC will then be appropriate for demonstrating effectiveness of the second shutdown means. Hence, the EX-DEC analysis in Chapter 15 crediting DL4a shutdown means may already be in compliance with the CNSC requirements and expectations for the second shutdown means.

3.5.1 Regulatory Requirements and Guidance for Shutdown Means

Because of its critical importance, IPR team performed a high-level assessment of BWRX-300 compliance with the requirements and expectations for shutdown means in REGDOC-2.5.2. This assessment is documented below in Table 2 and is followed by recommendations.

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Table 2: Compliance Assessment for BWRX-300 Shutdown Means against Requirements and Guidance in REGDOC-2.5.2

	Reference Regulatory Document	IPR Team Assessment
Design Features		
Design: Two shutdown means are required to mitigate DBAs	8.4 Requirement: The design shall provide means of reactor shutdown capable of reducing reactor power to a low value ... The design shall include two separate, independent, and diverse means of shutting down the reactor.	BIS (DL4b) may not meet CNSC requirements DL3 hydraulic scram does (the fast-acting SD means) DL4a fast rod run-in can (the non-fast-acting SD means) The requirements for the fast-acting shutdown means are more stringent than those for the second (non-fast-acting) shutdown means. BWRX-300 design is likely already in compliance.
Design: One shutdown means will need to be fast-acting and single-failure proof (essentially triplicated)	8.4 Requirement: At least one means of shutdown shall be independently capable of quickly rendering the nuclear reactor subcritical from normal operation in AOOs and DBAs, by an adequate margin, on the assumption of a single failure. Redundancy shall be provided in the fast-acting means of shutdown ...	DL3 meets the SFC – redundancy provided already, triplicated, etc. DL4 may not meet the SFC – nor is required to. BWRX design is likely already in compliance. Redundancy is required only for the fast-acting shutdown means (DL3)
Design: Sharing in-core components (such as CRs) are allowed	8.4 Guidance: For the two means of shutting down the reactor to be independent of each other, they do not share components. If both means act inside the core and complete separation is not possible, adequate separation of ex-core components should be demonstrated.	In-core components (e.g., CRs) are shared. For DL4a, equipped with an independent fast rod run-in (electro-mechanical), and ARI hydraulic shares some out of core components (accumulators, piping, valves) BWRX design is likely already in compliance.
Design: Adequacy of design needs to be demonstrated through safety analysis.	8.4 Guidance: [For DBAs], a shutdown means is considered to be effective if the safety analysis acceptance criteria are met.	DL3 effectiveness is demonstrated in the PSAR (notwithstanding numerous IPR comments). Gap: If applicable, DL4a effectiveness to mitigate DBAs as the second shutdown means is uncertain in the PSAR. EX-DEC DL4a shutdown effectiveness for prevention of progression to severe core

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	Reference Regulatory Document	IPR Team Assessment
		damage is included in the PSAR but may not address DBA design/analysis requirements depending on the combined initiating event frequency and failure probability of the fast-acting shutdown means.
Design: Effectiveness of shutdown means through safety analysis is required.	8.4.1 Requirement: The design authority ... shall perform a safety analysis to demonstrate the effectiveness of the means of shutdown.	DL3 – See above. Gap: DL4a – See above.
Design/Analysis: Reliability and Independence of shutdown means	8.4.2 Guidance: ... the cumulative frequency of failure to shutdown ... is less than 10^{-5} failures per demand ... This ... recognizes that the two shutdown means may not be completely independent.	Gap: Reliability analysis is missing in the PSAR. Demonstrating overall reliability of the shutdown means may be a part of Alternate Approach, if required.
Trip Coverage		
Design/Safety Analysis: Trip coverage requirements	8.4.1 Requirement: For each credited means of shutdown, the design shall specify a direct trip parameter to initiate reactor shutdown for all ... DBAs.	AOO: not a concern – DL2 is good Gap: DL3 – justification for direct trip missing in Chapter 15 Gap: DL4a – if applicable, analysis is uncertain for DBAs and justification for direct trip missing in Chapter 15 IPR Team interpretation: This clause is applicable to Reactors with Engineered Safety (e.g., CANDU). Also see the requirement below.
Design/Safety Analysis: Diverse trip coverage	8.4.1 Requirement: For all AOOs and DBAs, there shall be at least two diverse trip parameters unless it can be shown that failure to trip will not lead to unacceptable consequences.	This is open to interpretation for BWRX-300. It is clearly a requirement for CANDU. For BWRX-300, this requirement is not consistent with the previous requirement and REGDOC-2.4.1 trip coverage table. IPR Interpretation: This is not considered applicable for BWRX-300 as long as a direct trip is effective.
Design/Safety Analysis: Requires adequate trip coverage	8.4.1 Guidance: [...] through safety analysis performed in accordance with REGDOC-2.4.1, Deterministic Safety Analysis ... Trip coverage should be demonstrated ... for all credited	Gap: Chapter 15 of PSAR needs to establish BWRX-300 as an inherently safe reactor and establish that the credited trip for DBAs is a direct trip for each shutdown means (DL3 and

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	Reference Regulatory Document	IPR Team Assessment
	shutdown means ... the number of credited trip parameters can vary with the event, the reactor design, and whether there is a direct trip available.	DL4a, if applicable). If not, demonstrate dual trip coverage for each shutdown means.

The BWRX-300 design features appear to already comply with the CNSC requirements and guidance for two separate, independent, and diverse shutdown means, but this needs to be documented and confirmed by closing the identified gaps above. Notwithstanding the discussion above regarding CNSC guidance for the second shutdown means depending on the combined initiating event frequency and the failure probability of the fast-acting shutdown means, IPR Team recommendations for consideration are as follows:

- Declare DL4a as the second shutdown means – without any design change or classification. DL4b BIS provides additional coverage for extremely low frequency BDBA scenarios but is not one of the two shutdown means credited to mitigate DBAs. Document justification that DL4a as is meets REGDOC-2.5.2 requirements for the non-fast-acting shutdown means,
- Perform additional CN-DBA safety analysis, or reorganize the existing EX-DEC analysis in Chapter 15, crediting DL4a shutdown means only for mitigation of DBAs,
- Provide justification in Chapter 15 that BWRX-300 is an inherently safe reactor,
- Provide justification that credited trip parameter is a direct trip parameter for both DL3 and DL4a shutdown means – otherwise safety analysis showing dual trip coverage for each shutdown means is required, and
- Perform and document reliability analysis to demonstrate that the failure to shutdown on demand is less than $1E-5$ to address any residual concerns related to independence and separation.

3.5.2 Requirements for Safety Analysis in REGDOC-2.4.1

Table 3 summarizes the REGDOC-2.4.1 requirements evaluated against the safety analysis presented in the PSAR.

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Table 3: REGDOC-2.4.1 Requirements vs. BWRX-300 PSAR Safety Analysis

	Normal Operation	AOO	DBA	BDBA	Observations on BWRX-300 PSAR Rev 0
IE Frequency	NA	>1E-2 /year	>1E-5 /year & <1E-2 /year	<1E-5 /year	<p>Loss of reactivity/power control initiating events leading to power increase also need to be considered as AOO & DBA as per Table 1 and Table 2 of REGDOC-2.4.1, respectively. As an example, control rod withdrawal events are listed both in Table 15.2-1 (event groups) and Table 15.6-5 (generic BWR initiating event), yet there is no AOO or DBA event analyzed in Chapter 15 related to inadvertent rod withdrawal.</p> <p>Several IE frequencies are provided in the PSAR (Tables 15.5-7 and -8 for PSA) but do not appear to be comprehensive or used as the basis of Event Classification for AOO and DBA classification. As an example, initiating event frequency for inadvertent rod withdrawals is not shown in these tables.</p> <p>It is not clear if BWRX-300 design complies with the frequency-based Event Classification scheme for External Hazards in REGDOC-2.4.1.</p>
Initial Operating Conditions	Realistic	Realistic values	Limiting values	Realistic	The limiting initial operating conditions are not always identified for DBAs, and they do not appear to be used in all CN-DBA analysis, particularly for non-LOCA DBAs. While Table 15.5-1 provides limiting values for some operating parameters for LOCAs, nominal values seem to be used in CN-DBA non-LOCA analysis (Table 15.5-3). It is not clear how the Safe Operating Envelope (CSA N290.15) will be determined without using limiting values in CN-DBA analyses.
Public Dose Limits	1 mSv (DRL)	0.5 mSv	20 mSv	None (Input to PSA)	Compliant

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	Normal Operation	AOO	DBA	BDBA	Observations on BWRX-300 PSAR Rev 0
Derived Acceptance Criteria	Same or tighter than AOO (ASME SL A)	Return-to-Service/Fuel-Fit-for-Service (No damage to fuel or structures) (ASME SL B)	No failure of Pressure Boundary (ASME SL C) For most IE scenarios, no fuel failure.	(ASME SL D) (Input to PSA)	The AOO and DBA Acceptance Criteria in Table 15.3-1 and Table 15.3-2 of the PSAR do not meet some expectations in REGDOC-2.4.1 (Clause 4.3.2, Clause 4.3.4, and Table B.1 and Table B.2 for AOO and DBA). As an example, fuel must remain fit for use following an AOO. In addition, economic interests of OPG dictate no damage to fuel or reactor because of frequent initiating events such as Turbine Trip/Load Rejection such that the reactor can be restarted in a timely manner without extensive inspections or repairs.
The SFC	no	no	yes	no	While BWRX-300 appears to be compliant, the limiting component that is assumed to fail needs to be justified with documented rationale for each acceptance criterion for each event. In addition to the SFC, Canadian regulations contain Reliability requirements for safety functions (REGDOC-2.6.1) to ensure adequate redundancy in design.
Analysis Methods: conservatism in codes and models	Best Estimate	Best Estimate	Need to account for code and modelling uncertainties	Best Estimate	Both BL-AOO and CN-DBA analyses need more evidence that the code is qualified for the specific application for a given reactor design. Code qualification should include a BWRX-300 specific assessment of "Code Applicability" and "Code Accuracy" for each analyzed PIE. For both AOOs and DBAs, biases in code prediction for Figure of Merit (FOM) parameters and "safety system initiation" parameters need to be accounted for. For DBAs, in addition to the operational variations that need to be conservatively accounted for, code prediction uncertainties also need to be accounted for. The PSAR does not provide sufficient evidence that AOO and DBA analyses meet the requirements and expectations regarding Code Applicability and Code Accuracy. Instead, high-level

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	Normal Operation	AOO	DBA	BDBA	Observations on BWRX-300 PSAR Rev 0
					descriptions are provided with references to GE-H proprietary reports that have not been reviewed by the IPR team.
Systems Credited	Level 1 only (DL1)	Level 2 only (DL2)	Level 3 (DL3) & possibly DL4a No credit for DL1/DL2 unless DL1/DL2 actions result in more adverse results	DL4a, DL4b including EME [DL2 and DL3, if qualified].	For DBAs, no credit should be taken for DL2 systems if DL2 system(s) help mitigate consequences of an accident. However, “a <i>DL2 system action that adversely affects the accident consequences needs to be included in CN-DBA analysis</i> ” - see REGDOC-2.4.1 Clause 4.4.4 Item 3. The CN-DBA analysis in the PSAR assumes a common cause failure of DL2 functions such as pressure control, level control and power control. For some DBAs, crediting a subset of DL2 (or DL1) functions may affect the accident consequences and hence need to be considered for selecting event scenarios to be analyzed. Examples are provided in detailed comments in Appendix B.
Shutdown Means	N/A	DL2 only	DL3 (hydraulic scram), and possibly DL4a	DL4a, DL4b	For mitigation of DBAs, Both REGDOC-2.4.1 and REGDOC-2.5.2 require two independent shutdown means. BWRX-300 has one DL3, one DL4a and one DL4b means of shutdown. AOO mitigation is provided by DL2 shutdown function. Boron Injection system (DL4b) is a means to add negative reactivity but has no dedicated instrumentation or automated actions and can only be credited after 7 days and is not considered one of the two shutdown means credited to mitigate DBAs by the IPR team. In addition to the fully automated hydraulic scram (DL3), BWRX-300 also provides a strong DL2 and DL4a shutdown safety functions. Furthermore, the DL2 shutdown function appears to have high reliability. Also see discussion in Section 3.5.1 related to compliance of BWRX-300 with the requirements and guidance for shutdown means.

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	Normal Operation	AOO	DBA	BDBA	Observations on BWRX-300 PSAR Rev 0
Shutdown Means: Trip Parameter			Up to two effective trip parameters for each shutdown means		This comment applies to CN-DBA analysis only. The IPR Team considers the BWRX-300 design as a Reactor with Inherent Safety (Chapter 3) for application of Table 3 of REGDOC-2.4.1. Depending on the type of trip parameter (Direct or Indirect), one or two effective Trip Parameters are expected for each shutdown means for Reactors with Inherent Safety. BWRX-300 credits the first trip parameter without justification for the credited trip parameter to be a direct trip parameter for each CN-DBA analysis. Also see discussion in Section 3.5 for requirements and guidance on trip coverage.
BWRX-300 PSAR safety analysis	Best-estimate	Baseline-DSA [BL-AOO] crediting DL2 only	Conservative-DSA [CN-DBA] crediting DL3 only	Extended-DSA EX-DEC	Chapter 15 is generally compliant for AOOs, DBAs and DEC. For DBA's, in addition to having DL3 safety functions for containment, emergency core cooling and shutdown, BWRX-300 has DL4a functions for initiation of Reactor/Containment Isolation Valves and ICS, which are considered to go beyond the safety design requirements of REGDOC-2.5.2 for ECCS and Containment functions.

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3.6 REGDOC-2.4.2 *Probabilistic Safety Assessment for Nuclear Power Plants*

REGDOC-2.4.2 describes the requirements and guidance for a licensee to establish a program for the development and use of probabilistic safety assessment (PSA), as a means to manage radiological risks, and contribute to the safe design and operation of reactor facilities.

PSA is a quantitative analysis that considers both the likelihood and the consequences of various plant transients and accidents. The primary objectives of the PSA are to help with:

- identifying the sequences of events and their probabilities, which lead to challenges to fundamental safety functions, loss of integrity of key structures, release of radionuclides into the environment, and public health effects,
- developing a well-balanced Nuclear Power Plant (NPP) design by demonstrating that no particular design feature or postulated initiating event can make a disproportionately large contribution to the overall risk,
- providing assessments of the quantitative safety goals (the probabilities of occurrence for severe core damage states, and the assessments of the risks of radioactive releases to the environment),
- identifying facility vulnerabilities, and assessing modifications to systems or design improvements that reduce the likelihood of core damage and releases to the environment, and
- to provide insights into severe accident⁵ management.

Section 3 of REGDOC-2.4.2 provides detailed requirements for PSA, including the requirement to perform Level 1 and Level 2 PSA, to conduct the analysis under a management system and quality assurance program, to develop a PSA model that reflects the as-built design and operation of the NPP, to identify all potential site-specific initiating events and potential hazards (internal, external, and potential combination of hazards), use of assumptions, methodology and computer codes, and to include sensitivity and uncertainty analysis, and risk importance measures in the PSA.

Review of the PSA presented in the BWRX-300 PSAR Rev. 0 is in Section 6.4 of this report.

3.7 REGDOC-2.4.3 *Nuclear Criticality Safety*

REGDOC-2.4.3 *Nuclear Criticality Safety*, [R-5] sets out requirements for nuclear criticality safety and provides guidance on how these requirements can be met, with the expressed purpose of, “*preventing criticality accidents in the handling, storage, processing, and transportation of fissionable materials, and the long-term management of nuclear waste.*”

⁵ For licensing new NPPs, PSAs help identify complementary design features (internationally, these are also called additional safety features) for severe accidents, or actions that operators can take during severe accidents to reduce risk.

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Criticality safety requirements in REGDOC-2.4.3 are applicable to any reactor using enriched fuel, along with requirements for a Waste Management Program, utilizing CSA Standards N292.1 *Wet storage of irradiated fuel and other radioactive materials*, N292.2 *Interim Dry Storage of Irradiated Fuel*, and N292.3 *Management of Low and Intermediate-Level Radioactive Waste*.

For the BWRX-300 design using low enriched uranium (< 5% U-235) fuel, demonstration of criticality safety during the transport, handling, and storage of the fuel is expected to be part of the licensing basis.

An evaluation of the methodology for BWRX-300 Out-of-Core Criticality (OCC) Assessment is described in Section 6.5 of this report.

4.0 SAFETY IMPROVEMENTS IN THE BWRX-300 DESIGN

During the independent review of the BWRX, it was noted that the provisions for defence-in-depth and improvements in risk and safety are significant in the design. While not always directly compatible with Canadian regulations, these safety improvements are substantial. Many of these improvements were also noted by the US NRC during its review processes, in documents such as Licensing Topical Reports for BWRX-300 Reactivity Control [G-3], RPV Isolation and Overpressure Protection [G-7], and Containment Response [G-10]. A few selected high-impact improvements are noted below.

4.1 Seismic Margin

The BWRX-300 is qualified to withstand a 0.3-g earthquake and has a high confidence of a low probability of failure of its seismic success path for a 0.45-g earthquake. The likelihood of such a high seismic magnitude at the Darlington site from faults in eastern North America is extremely small. Given the large margins and robust seismic protection of critical systems, the design appears very robust in this regard.

4.2 Simplified Design

Simplification of the design leads to higher reliability from the reduction in complexity as well as the reduced number of components that are required to operate and can fail.

The BWRX-300 has significantly fewer structures, systems, components and credited operator actions than existing BWRs and other nuclear power reactors. This has reduced the number of internal postulated initiating events and event combinations, which reduces the complexity of the safety case, the number of contributors to risk and the minimum shift complement. It also would mean a reduction in the number of maintenance tasks performed in outages as well as online testing requirements which would simplify operational planning and execution, improve generation reliability, and improve overall safety.

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4.3 Passive Safety Systems

As nuclear reactors are evolving, they are moving away from active components such as pumps and valves. BWRX-300 is an example of evolution in BWR technology where most of the pumps and valves have been eliminated, with a significant reduction in piping, and the containment design has been simplified. There are three passive safety principles: gravity driven flow, pressurized reservoir, and natural circulation driven flows and heat transfer.

- The primary coolant circulation in the core is driven by gravity which generates a net buoyancy force in a tall chimney with low density fluid. There are no active components (no jet pumps) inside the RPV to drive coolant flow through the core and hence reliability is improved.
- Reactor scram by hydraulically driven control rods is moved by pressurized high-pressure fluid that is passively stored in hydraulic control units (HCU).
- Consequences of Large Loss of Coolant Accidents (LOCAs) have been largely mitigated by the RPV isolation concept wherein all major connections to the vessel have redundant isolation functions to limit blowdown discharge. Upon isolation, there is a large amount of liquid remaining in the core⁶ which assures fuel remains covered. Isolation condensers (ICs) are then deployed to act as a heat sink for the isolated RPV. Each of the three independent Isolation Condenser System (ICS) trains contain two independent actuation valves, one with throttling capability for controlled cooldowns via DL2, and the other on/off for DBA protection within DL3.

Rather than allow a reactor to drain and uncover fuel, BWRX-300 design employs an intuitively simpler and preferable approach to not uncover fuel by starting with more coolant in the RPV and limiting its loss during the accident, and continuously remove decay heat from the core through passive heat exchangers that do not rely on pumps and complex valve/piping configurations to drive flow (see below). It also reduces concerns associated with recovering and re-injecting large volumes of potentially radioactive water that would spill indefinitely from breaks that are not isolated (e.g., recovery strainer plugging issues).

- The reactor is connected to three isolation condenser systems (ICSs) that are connected to the RPV and are at reactor pressure during normal operation. The ICS heat exchanger parts reside in external pools above containment. The condensate return legs of ICS have valves that are normally closed and fail open. ICS has many functions, including aiding over pressurization protection, removing decay heat during certain normal outages and also for certain AOOs and DBAs, and supplies some coolant inventory (stored in ICS HXs and piping when poised). Based on RPV pressure and level setpoints or scram, the ICS return valves open, condensate flows into the chimney while steam from the RPV steam dome flows into ICS heat exchangers and transfers heat to water pools outside containment while condensing. ICS operates on passive principles: buoyancy-difference

⁶ Although the IPR team views inventory preservation as a strength of BWRX-300 design for reducing Emergency Core Cooling requirements and reducing safety system complexity, a key related finding of this review is that it does not eliminate the need for longer term coolant makeup capability, e.g., for unisolable small pipe breaks. This is noted elsewhere in this report.

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driven flow, and heat transfer by condensation on the tube side and natural circulation/boiling on the shell side.

The removal of decay heat from the core, an ECCS function, is achieved through the use of ICS in BWRX-300. The ICS, by its simplicity, proven design⁷ based on OPEX, passive operating principles and large heat removal capacity, is superior to the ECCS recirculation phase currently employed in most LWRs and CANDUs. It does not rely on a complex arrangement of pumps, valves, strainers, external heat exchangers for operation in a potentially radioactive environment, with significantly higher expected reliability.

- e) In the event of a break inside containment in un-isolated small pipes, the containment pressure and temperature will increase. The containment is protected against over pressurization and excessive temperature increase by the passive containment cooling system (PCCS). The PCCS operation is also based on gravity-driven flows to transfer heat to an external pool and requires no valves to actuate. The heat exchangers are in the containment with fluid intake and return in an external pool. The hot fluid moves to the external pool and is replaced by cold fluid through an intake in the external pool.

4.4 Inherent Safety Features

The BWRX-300 design has inherent safety features that should be viewed as an important strength. For example, the inherent reactivity characteristics (described in [G-5]) associated with fuel temperature and coolant voiding provide strong and prompt negative feedback to limit power increase during selected postulated BWRX-300 accidents where fuel temperature or coolant void fraction increases. This reduces speed requirements for the means of shutdown. It also reduces the consequences of Design Extension Condition (DEC) cases that include postulated failure/impairment of the scram (shutdown) function.

GE-H has appropriately credited inherent reactivity characteristics as part of the justification for an alternative approach for means of shutdown (Reference [G-14]). If other important inherent safety features are identified during BWRX-300 licensing, it is expected that they will be similarly highlighted in Chapter 15 of the PSAR or related licensing documentation. Refer to Section 6.0 below for other related considerations.

4.5 Protection Against Single Failures and Reliability Requirements

As described in the PSAR, many BWRX-300 systems are single-failure proof, and this evolutionary design appears to have adequate redundancy to meet the Single Failure Criterion for safety systems to function to mitigate DBA consequences. As well, BWRX-300 design is expected to comply with the Reliability requirements (REGDOC-2.6.1) to achieve the same objective of mitigation of DBAs with sufficient redundancy included in the design.

⁷ ICS is a proven design notwithstanding the IPR team's concerns for ICS potentially impeding natural circulation in the vessel and fuel cooling due to its condensate discharge location into the chimney. This is noted elsewhere in this report.

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4.6 Use of Digital Control

Another BWRX-300 advantage is the excellent use of distributed control and digital technologies, especially for licensing in Canada where reliable Digital Control Computers have been successfully used since 1971 (Pickering A NGS) for process systems and since 1990 (Darlington NGS) for safety systems.

4.7 Proven Fuel and Other Design Elements

The BWRX-300 fuel GNF2 has been used and proven in operating BWRs [G-9], and this aspect is expected to reduce the overall effort by OPG to develop, license, commission and operate BWRX-300. There is considerable operating experience and qualification of the fuel which makes it a reliable choice for the BWRX-300 design.

This observation is also extended to the design of the 57 control rods, also using the best features of previous BWR designs including Fine Motion CRD mechanism.

Natural circulation in the RPV has been previously used in the Dodewaard nuclear power plant (Netherlands).

Similar isolation condenser systems have been used in early BWRs as the emergency heat removal systems successfully for over 40 years. The genesis of the BWRX-300 IC is the Simplified Boiling Water Reactor IC design which was tested extensively for performance and other operating characteristics. The IC version used in BWRX-300 is similar to that in the Economic Simplified Boiling Water Reactor (ESBWR) with some differences including higher design pressure, hardware material selection to enhance the design, and condensate discharge location that should not affect heat removal characteristics of the IC.

5.0 MAIN FINDINGS

The IPR team review comments can be found in the following sections of this report:

- This section (Section 5.0) highlights the main findings of the independent peer review.
 - As previously noted, if the IPR reviewer concludes that the safety assessment includes a defensible “alternative approach” per REGDOC-2.5.2 Clause 11, this is noted in this section.
 - This section also provides a basis or rationale for the IPR team’s judgment that with further revisions in design and/or safety analysis, the PSAR should be able to address these review findings adequately.
- Section 6.0 discusses the detailed accident analysis review.
- Section 7.0 discusses general topics and OPEX from other BWR stations.

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- Appendix B includes supplementary review comments that complement the discussion in this Section and Section 6.0.

5.1 Additional information required in the PSAR

PSAR Rev. 0 [G-1] was reviewed against Canadian REGDOCs. Based on the IPR, there are items that require additional information. Specifically, Clause 8.5 of REGDOC-2.4.1 identifies requirements for deterministic safety analysis documentation. The information provided in Chapter 15 does not appear to be sufficient to meet the above Clause, especially Items 5, 6, 7, 8 and 9.

- Major items that require additional details include, but are not limited to:
 - Event Identification and Classification (See Section 5.4),
 - Selection of Bounding Events (See Section 5.4),
 - Justification of equivalent standards, alternative approaches and graded application of REGDOC-2.5.2 (See Section 5.2).

It is acknowledged that additional information may be present in other documents referenced in the PSAR, likely GE-H proprietary reports, and their review is considered outside of the scope of the IPR. See Section 2.2.1 on IPR scope limitations. While not meant to be an exhaustive listing, the need for additional clarification or information has been noted in this report including in the comments in Appendix B.

- Another area that requires more detail is Severe Accident Deterministic Safety Analysis (DSA). The design has incorporated important industry lessons to make sure the core damage frequency is extremely low, and some scenarios are practically eliminated. This means that the large release frequency also is extremely low without credit for a low conditional core damage probability (CCDP) and severe accident analysis. The preliminary design has included some features to reduce the CCDP. For example, the containment is inerted to prevent hydrogen burning in containment. However, the PSAR has no severe accident analysis results, and the design identifies DL4b equipment but most lack detailed descriptions. Preliminary severe accident analyses in a separate proprietary report were not reviewed. See Section 6.2.3.3 for more details.

While not an exhaustive listing, where additional clarification or information is required, it has been noted in this report.

5.2 Technical and Licensing Risks

A clause-by-clause comparison of REGDOC-2.5.2 requirements and guidance to the BWRX-300 design, if exists, was not reviewed by the IPR team. PSAR Rev. 0 has many references to REGDOC-2.5.2 sections with which the BWRX-300 complies. There is one case (Chapter 4) of an equivalent US standard to CSA N286.7, one case (Chapter 8) where a rationale is provided in the PSAR for not complying with REGDOC-2.5.2 (redundancy of electrical power),

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one case (Chapter 3) where another report (NEDC-33974P) is referenced to provide an alternative approach to REGDOC-2.5.2 (seismic categorization) and two cases (Chapters 3 and 9A) where the justification for an alternative approach to IAEA/REGDOC-2.5.2 (independence of defence lines) and fire protection have been left for a future revision to be confirmed as the design is finalized. Significant potential gaps in compliance with the requirements in applicable regulatory documents have neither an equivalent standard, a rationale in the PSAR, a report referenced by the PSAR nor explained as a future task upon completion of the detailed design. They are listed below and in Appendix B of this document, per PSAR Rev. 0. The major gaps in the design or design description in the PSAR which should be dispositioned include, but are not limited to, the following:

- a) There are no details on many of the complementary design features and safety-grade (DL4b) water injection which may be required for unforeseen events, or for inventory make-up for unisolable leaks or breaks (DBA) in the longer term. Chapter 15.6 provides only a high-level description of many important functions (such as CRD purge water injection). It is not clear if the DL4b injection function is functionally and physically separated from the systems intended for other DL functions, defense lines or whether it uses diverse and flexible equipment and portable components.
- b) For small, unisolable liquid line breaks (DBA), the PSAR identifies no provision for coolant make-up to the RPV for the first seven days when the DL4b CRD coolant injection can be credited, if it still has inventory left. The PSAR does not contain any assessment after 3 days, e.g., a coping study using best estimate analysis techniques, demonstrating acceptable fuel temperature or water level in the core. DL4b designated systems are meant to provide protection for extreme events, multiple (i.e., combined) events, or multiple failures. In the CN-DBA analysis, the collapsed water level falls below the Top of Active Fuel at about 55 hours. Without inventory make-up, the collapsed water level will continue to decrease likely leading to substantial fuel uncover and heat up within the first 7 days. For this DBA, PSAR provides limited information for only the first three days and does not provide sufficient evidence that the fuel temperature can be kept below normal operating temperature within the first 3 days and that the event does not progress to severe core damage before 7 days. In addition, for unisolable breaks within containment, if continuous water injection to RPV is required, water discharge into containment may pose containment integrity or equipment operability issues due to flooding. Furthermore, once the RPV and containment pressures equalize, there is a potential for non-condensable nitrogen gas to enter the RPV and impair the effectiveness of ICS heat removal capability.
- c) In the past, strict independence of process control and safety systems has been a Canadian regulatory requirement. Given that BWRX-300 uses the same control rods for reactor power and reactivity control during normal operation, and for DL2 and DL3 scram, and for DL4a rod run-in or ARI for accidents, BWRX-300 would have been non-compliant with past Canadian regulations. The current Canadian regulatory requirements (e.g., section 6.1⁸ of REGDOC-2.5.2) now allow sharing of process/safety

⁸ "The levels of defence in depth shall be independent to the extent practicable."

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- equipment (i.e., levels of defense). Additional discussion of this issue can be found in the next bullet and in Section 7.1 of this report.
- d) Similarly, because of the positive void reactivity coefficient inherent in CANDU reactors, Canadian reactor licensing has historically required two fully capable Shutdown Systems, e.g., see [R-9]. This is no longer a hard requirement in REGDOC-2.5.2, Section 8.4. In the interests of technology neutrality, REGDOC-2.5.2 requires two appropriately redundant and diverse means of shutdown one of which, designated as the “fast-acting” means of shutdown, needs sufficient redundancy. REGDOC-2.5.2 does not presume that every candidate design will have reactivity characteristics that require two fully effective and independent Shutdown Systems. As discussed in [G-14], the BWRX design combines an inherently safe core design with multiple appropriately redundant and diverse means of shutdown. These means of shutdown include automatic DL2 anticipatory trip functions to mitigate AOOs; automatic, triplicated, single-failure-proof DL3 hydraulic scram; automatic DL4a fast control rod insertion that uses independent sensors/logic or ARI; and a manual DL4b Boron Injection System (BIS) for extremely low probability BDBA sequences. Except for BIS, all of these means use the same 57 control rod blades. However, sharing of these in-core components is explicitly allowed in REGDOC-2.5.2 Clause 8.4. A strong case can be made that the BWRX means of shutdown may already comply with the relevant REGDOC-2.5.2 requirements and may in fact exceed some. Even if the BWRX design is deemed to be not fully compliant, the information provided in [G-14] makes a case for BWRX means of shutdown to be accepted to provide an equivalent level of safety using the “alternative approach” per REGDOC-2.5.2 Clause 11. Clause 8.4.2 of REGDOC-2.5.2 establishes numerical expectations for shutdown reliability as the basis for two shutdown means (e.g., the cumulative frequency of failure to shutdown should be less than 10^{-5} failures per demand). The results of reliability analysis calculations can be an important factor in resolving this issue. See the discussion regarding the requirements and guidance for means of shutdown in Sections 3.5 and 3.5.1.
- e) Both REGDOC-2.5.2 and REGDOC-2.4.1 require one or two diverse trip parameters (for each shutdown means) unless it can be shown that failure to trip will not lead to unacceptable consequences. The BWRX-300 design, which is considered a reactor with inherent safety, must demonstrate that the trip coverage requirements of REGDOC-2.5.2 and 2.4.1 (shown in Table 1 of this report) are met in the design or via a documented alternative approach to the REGDOCs utilizing Clause 11 of REGDOC-2.5.2 and/or Graded Approach in REGDOC-2.4.1. See the discussion regarding trip coverage requirements and guidance in Sections 3.5 and 3.5.1. Additionally, consideration should be given to credit the second trip parameter for each means of shutdown for events with positive pressure feedback.
- f) As discussed in Sections 6.4 and 7.5 of this report, the possibility of severe core damage leading to containment failure and large release (as happened at Fukushima) is a potential concern. For DEC, BDBA and severe accident sequences, BWRX-300 incorporates many, significant improvements for emergency cooling of the core as well as for containment emergency cooling and venting. With these improvements, plant states leading to significant release of radioactivity have been practically eliminated, as

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is required by Clause 7.3.4 of REGDOC-2.5.2.

5.3 Normal Operating Conditions Analysis

The analysis of the Normal Operating Conditions (NOC) documented in separate GE-H proprietary reports was not reviewed. Demonstration of the normal operating conditions will ensure that the reactor performs as designed. This issue is considered not only as a potential nuclear safety issue but also as an economic concern since convincing evidence is needed in PSAR to support safe operation of the reactor at the design power output of 300 MWe.

There should be a discussion and detailed results of overall core coolant flow, individual fuel channel flows, spatial neutron flux/power distribution, and rod positions as a function of fuel exposure (or for BOC, MOC, EOR), and load following. The Critical Power Ratio (CPR) distribution and safety and operating margins can then be determined with accuracy once coolant flow rates for each channel and spatial power distribution are known with accuracy. The codes used for these purposes need to be appropriately qualified with code accuracy determined for each parameter to provide confidence in the safety and operational margins.

Decay Ratio for Stability seems to be the only criterion used to evaluate stability. The decay ratio appears to be extremely sensitive to operating parameters, such as feedwater temperature, bringing into question⁹ whether the decay ratio less than 0.8 alone is a sufficient criterion for determining stability. Additional details are provided in Section 7.3.

5.4 Event Identification & Classification and Selection of Limiting Events

EI&C

The selection of limiting events in Chapter 15 of the PSAR is a reasonable reflection of experience with BWR DSA. However, a complete identification and classification of events should be included in the Pre-Operational Safety Analysis Report or the Final Safety Analysis Report (FSAR) using the systematic methodology as described by GE-H in PSAR Chapters 3 and 15, and in accordance with REGDOC-2.5.2 and REGDOC-2.4.1.

For example,

- A list of system failures is not provided to identify and select limiting events.
- Design specific initiating event frequencies are not available to confirm event classification. The lists of initiating events in the preliminary PSA and DSA differ, which results in no frequencies quoted for some DSA initiating events and no DSA for some PSA events.

⁹ T.H.J.J. van der Hagen, R. Zboray and W.J.M. de Kruijf, "Questioning the use of the decay ratio in BWR stability monitoring," Annals of Nuclear Energy, 27 (2000) 727-732.

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- Initiating event examples in Appendix B of REGDOC-2.4.1 are to be considered, or justification for exclusion is required. While additional detailed comments can be found in Appendix B, non-exhaustive examples include:
 - Loss of Reactivity Control and Loss of Power (both as AOO and DBA),
 - Feedwater Temperature Increase and RPV pressure decrease (both as AOO and DBA),
 - RPV Inventory Increase and RPV Inventory Decrease (both as AOO and DBA),
 - FW temperature decrease that is just insufficient to reach the DL2-27 setpoint for AOO analysis, and
 - Blinding breaks (break discharge is insufficient to actuate the leak detection setpoint for isolation).
- Complex DEC's arising from PSA other than Turbine Trip have not been identified.
- External hazard analysis and event classification needs to be completed along with PSAR Sections 3.3 and 3.4.
- DBAs related to reactivity and power distribution anomalies only consist of fuel loading events, and there is no AOO or DBA for reactivity and power anomalies, e.g., due to inadvertent control rod movements. Either additional analyses covering inadvertent rod movements (with no DL2 rod blocking functions credited) are needed in the DBA or additional justification for their exclusion is needed. Higher frequency reactivity or power perturbation events need to be considered as AOOs to ensure DL2 functions can mitigate these events.

Selection of Limiting Events

Associated with the incomplete Event Identification and Classification (see Section 5.4), the justification of the limiting (bounding) events is inadequate. There are several events that require additional supporting information or analyses to confirm that they are indeed bounded by the existing analyses. For examples, (also see Appendix B comments about PSAR Sections 15.2 and 15.5):

- Feedwater temperature increase from FTC malfunction (AOO),
- RPV pressure decrease from RPC malfunction (AOO),
- RPV inventory increase from RLC malfunction (AOO),
- RPV inventory decrease from RLC malfunction (AOO),
- Reactivity and/or power increase from control malfunction for the full range of initial reactor powers (AOO and DBA). For example, as an AOO, analysis should be performed with the rod withdrawals within the rod blocking limits (DL-2),
- Break of piping that extends below the TAF in the RPV (AOO and DBA),
- Medium LOCA – Non-isolable (DEC, Table 15.6-7),

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- Control rod ejection (practically eliminated),
- External events such as seismic, flood, tornado, cold weather, etc. (AOO, DBA and DEC),
- Load following (analysis of normal operation, combined with AOO, DBA and DEC),
- Startup (analysis of normal operation),
- Hot shutdown (analysis of normal operation),
- Cold shutdown (analysis of normal operation), and
- Severe accidents (DEC).

While 15.2.3 of the PSAR provides some rationale for grouping of events, a summary chart which maps the events analyzed in the PSAR to those events expected per REGDOC-2.4.1 should be provided.

5.5 Code Applicability, Accuracy, and Uncertainties

While a detailed review of code validation and applicability and accuracy (e.g., using external documents listed in PSAR Table 1.10-1 of [G-1]) was beyond the scope of the IPR, the IPR team recommends that additional information be provided in the PSAR on code accuracy and applicability to meet the requirements and guidance in Clause 4.4.5 and 8.5 of REGDOC-2.4.1, which also requires compliance with CSA N286.7. Code applicability needs to be justified for each category of initiating events individually rather than by AOO, DBA and DEC classifications, and code accuracy should be quantified specific to each event category and each Figure-of-Merit (FOM). While Rev 0 of the PSAR includes some limited discussion of TRACG and its qualification, the biases (applicable to both AOO and DBA analysis) and uncertainties (applicable to DBAs only) for each event are not quantified. Evaluation of GE-H proprietary code qualification reports was not included as a part of the scope of this Review. Some of this work may exist for a subset of events from previous regulatory submissions to the NRC related to CSAU, however a complete assessment over all analyzed events is needed. Further, after the quantification of the event-specific uncertainties, the analysis then needs to account for the code bias and uncertainty in predicting FOMs for that specific event. Based on previous experience, code applicability and accuracy work, and capturing its results in safety analysis, requires a significant effort and these activities should be well underway to be completed for the FSAR.

Uncertainty Analysis

Safety margins to PSA safety goals and the acceptance criteria for DSA of most AOOs, DBAs and DECAs appear to be large for the BWRX-300 design. These apparent large margins may reduce once 1) more appropriate acceptance criteria are adopted for AOOs (such as fuel and equipment remaining fit-for-service following more frequent events), 2) conduct of analysis for DBAs is re-evaluated (such as assumed failure of DL-2 functions that may reduce event consequences; see Table 3 and Appendix B Comment 36), and 3) appropriate trip-parameter coverage is provided for DBAs. REGDOC-2.4.1 requires that DBA safety analysis

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conservatively accounts for both prediction uncertainties (e.g., hydraulic loss factors, CHF, two-phase interfacial phenomena...) and operational variations (shutdown depth, plant initial conditions, calibration effect, instrument drift and instrument uncertainties, etc. which the PSAR accounts for by setpoint/delay analytical limits).

It is recommended that the PSAR clearly identify the FOMs for each AOO and DBA, and include code prediction uncertainty, as well as applicable operational uncertainties, for each FOM for DBAs. The IPR team also notes that more discussion and assurance is needed for each CN-DBA analysis as in many cases the application of conservatism to only minCPR and local power may be insufficient, and for the case where a larger subset of parameter uncertainties was included (e.g., load rejection) the details and magnitudes of the uncertainties are not provided.

5.6 Demonstration of New BWRX-300 Features

Isolation Condenser System (ICS) Condensate Return to Chimney

In BWRX-300, the ICS condensate is returned to the chimney while in operating BWR designs equipped with ICS, condensate return is into the downcomer. This is considered a significant novelty relative to existing BWR designs, and hence requires additional support to validate its effectiveness. The ICS, the primary heat removal system for many design basis accidents, is a key safety feature in the BWRX-300 design which lacks jet pumps and solely relies on buoyancy forces for core cooling. ICS operation while the reactor is at power will reduce the coolant flow rate through the core by reducing the density differential between the downcomer and chimney, and hence the driving force for natural circulation flow and fuel cooling. There is no OPEX for this ICS design configuration and the PSAR does not provide sufficient basis for demonstration of effectiveness through experiments or code validation. The 3D and two-phase flow phenomena in the chimney and the core (e.g., countercurrent flow/flooding in the core, condensation in the chimney, interfacial shear phenomena in the chimney) during ICS operation is complex, both during the initial operation by causing condensation of steam in the chimney and the resulting impact on the flow field, pressure, and reactivity response, and later during quasi-steady state operation with reduced core flow rate through the core. This review also acknowledges potential benefits of this design choice, primarily in reducing the number of nozzle connections to the RPV for SDC and Boron Injection systems. This configuration might also provide benefit in overpressure protection and reactivity response during the initial discharge of cold condensate water stored in the return line for some accident scenarios. Overall, the IPR team regards this aspect of the ICS design as "First of a Kind" because of the condensate discharge location, especially given that BWRX-300 lacks jet pumps for forced circulation of coolant through the core.

The CN-DBA analysis provided in Chapter 15, e.g., for spurious operation ICS while at power, does not provide sufficient information or references to available code validation basis in the PSAR. While such information may be available through other sources as discussed with GE-H during the course of this review, adequate discussion and references to support the analysis should be provided in the PSAR. Furthermore, for this DBA, the analysis assumes normally operating pressure and inventory control systems to be frozen as is which leads to depressurization and a subsequent reduction in reactor power, flow reversal in the core, and reactor scram due to low RPV pressure. The analysis needs to be updated to reflect other

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comments provided in this report regarding the conduct of safety analysis (Section 5.8 below and Appendix B) and needs to be supported by demonstrable evidence of adequate code qualification for the ICS operation at high power. As well, the code qualification basis for the complex two-phase flow phenomena, e.g., prediction of steam condensation in the chimney, the resulting depressurization effect and its impact on core flow rate and reactivity response, is not provided in the PSAR for those accident scenarios that benefit from initial ICS operation in reducing RPV pressure and core reactivity.

It should be noted that the ICS discharge location is not deemed to be a concern after the reactor scram. Even though the coolant flow path through the core is complex, the fuel should be cooled adequately as long as the water level in the chimney remains above the top of active fuel.

Reliability of the New RPV Isolation Valves

The BWRX-300 has a new feature: isolating the reactor vessel to protect against excessive coolant inventory loss for LOCAs by placing two integral RPV isolation valves in series on each large RPV connection. This feature does not eliminate potential LOCAs but greatly mitigates their consequences. The PSAR should include an additional description about the isolation valves: their control system, power supply and failure modes since any common cause failure mode would invalidate many conclusions in the safety case. Furthermore, any leak, seal failure or casing crack in the inner most isolation valve would cause a break or leak that cannot be isolated. While failures of the RPV itself are precluded, it is not clear if the integrity of the valve packings, stem and seals are of the same reliability. These issues may be addressed in the PSA, but the information was not available in the PSAR.

Steel Bricks

The use of Steel Bricks™ for containment is a First of a Kind Application (FOAK) and demonstration of its suitability provided external to the PSAR was not reviewed. Additional discussion is in Section 6.2.3.2 of this report.

5.7 Acceptance Criteria

Derivation and Appropriateness of Acceptance Criteria

The Acceptance Criteria need to satisfy both safety concerns and economic concerns (e.g., for AOOs there must be no damage to fuel, structures and/or pressure boundary). However, the acceptance criteria in the PSAR defined in Tables 15.3-1 for AOOs and Table 15.3-2 for DBAs are not always consistent with REGDOC-2.4.1 and REGDOC-2.5.2 requirements and guidance.

The allowable acceptance criteria per the REGDOC-2.4.1, specifically Clause 4.3.2, Clause 4.3.4 and Tables B.1 and B.2, are provided in Table 3 of this report and must be consistently set in the PSAR. The following items are specifically noted and further elaborated in Table 3 and the detailed comments in Appendix B:

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- a) AOO acceptance criteria are also a significant economic concern as initiating events in the AOO frequency range should not result in significant economic damages. Appropriate acceptance criteria for AOO would be avoidance of fuel dryout and Service level B limits (equipment remain fit-for-service). As an example, *the fuel shall remain suitable for continued use after AOOs* (REGDOC-2.5.2, Clause 8.1.1), while Table 15.3-1 in PSAR Rev 0 only requires no fuel failure due to certain failure mechanisms. As another example, Table 15.3-1 in PSAR Rev 0 only requires containment integrity to be maintained, while Table B.1 of REGDOC-2.4.1 requires containment to be fit-for-service with ASME Service Level B limits not exceeded.
- b) For DBAs, Service Level C is generally required for pressure retaining systems. While fuel failures are preferably avoided, CNSC allows for exceptions. Nevertheless, as an example, while Table B.2 of REGDOC-2.4.1 requires no fuel centerline melting, no corresponding requirement exists in Table 15.3-2 of PSAR. In fact, "Loss of fuel rod mechanical integrity will not occur due to fuel melting" is listed as an AOO acceptance criterion in Table 15.3-1. Similarly, prompt criticality should be avoided during DBAs as per Table B.2 of REGDOC-2.4.1 and Clause 6.1.1 of REGDOC-2.5.2 Rev 2 (draft). However, the information in the PSAR does not clarify whether prompt criticality has been avoided, e.g., for postulated initiating events like Turbine Trip/Load Rejection that leads to positive reactivity insertion.

5.8 Conduct of Deterministic Safety Analysis

Crediting DL2 actions for DBAs

As briefly noted in Table 3, CN-DBA analyses need to be conducted to demonstrate that DL3 (and possibly DL4a if they are credited to mitigate DBAs) safety functions alone are sufficient to mitigate DBAs, without the aid of any beneficial DL2 safety function. See REGDOC-2.4.1 Clause 4.4.4, Item 3. This means that DL2 functions need to be credited in CN-DBA analyses if they make the consequences of DBA scenarios worse. Instead, the CN-DBA analyses in PSAR assume all DL2 functions fail as is for DBAs. Certain normally operating DL2 functions may exacerbate the event consequences or have a potential to change the event progression in a significant manner, e.g., assuming reactor pressure control and reactor level control being frozen as is in the CN-DBA analysis of spurious operation of ICs. Specific examples can be found in the detailed comments in Appendix B.

Justification of Conservative Assumptions for DBA

The selection of conservative assumptions for each event, reactor power, exposure, safe operating envelope limits and acceptance criterion are not adequately defined and justified. Each DBA also needs a precise description of all conservatisms applied to ensure the analysis remains bounding including a description of the assumptions on instrumentation timing and uncertainties, system performance, and rod insertion characteristics along with any modelling conservatisms (e.g., minCPR, assembly power, core power etc.).

For example, in the generator Load Rejection/Turbine Trip CN-DBA analysis, the PSAR includes allowances for uncertainties while the analyses for other DBAs with increase in reactor pressure state "no additional conservatism is applied given the significant margin to

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the acceptance criteria". CN-DBA analysis must meet each acceptance criterion while accounting for operational variations, prediction uncertainties, and measurement uncertainties associated with operational parameters. Furthermore, it is not clear if operating assumptions (trip detection system timing, rod insertion characteristics) would provide more assurance of the overall conservatism rather than apply a conservatism to the minCPR only.

6.0 ACCIDENT ANALYSIS REVIEW

6.1 Bounding Accidents for Shutdown Function

This section and Appendix A discuss the BWRX-300 safety analysis results presented in Chapter 15 and highlight the unique features and behaviour relative to typical Canadian safety analyses. The BWRX-300 safety analyses involve AOOs, and assessment of the protection provided by DL2 systems, AOOs that progress to DBA due to a failure of one or more DL2 protection systems or postulated initiating events that give rise to DBAs in and of themselves, and AOOs/DBAs that progress to DEC due to failure in either DL2 and/or DL3 systems.

Several common areas for improvement were identified which may be duplicate of comments made elsewhere in this report and are summarized as:

1. There should be a clear identification of the conservatism applied in each analysis including those related to operational parameters (such as trip timing, initial margin to trip, rod insertion times, or system initial and boundary conditions) as well as modelling conservatisms (such as Critical Heat Flux (CHF) model, frictional loss coefficients, two-phase friction multiplier). For many analyses, the reported margins in the minCPR are quite large because the timing of the DL2/DL3/DL4a actions is calculated at near best-estimate conditions, and the first trip parameter is credited for CN-DBAs. If the analysis was conducted accounting for modelling uncertainties, measurement uncertainties and operational variations and a secondary trip parameter, if required, or the second means of shutdown, timing of DL2/DL3/DL4a actions may be different, and hence the apparent minCPR margin may not be as high as indicated.
2. In some accident analyses, evidence is not provided that the high power (100%FP) case is bounding. For example, there are fewer poised control rods and the margin to neutronic trips is larger at lower power levels and hence shutdown for accidents requiring a DL3 trip on high neutron or high thermal power may be delayed. Therefore, the void reactivity effects may counteract any benefits from lower initial fuel enthalpy at that power. The main text should clarify and justify the bounding nature of all the assumptions applied in the analysis.
3. Code modelling uncertainties, measurement uncertainties associated with operating parameters and DL2/DL3 initiating signals, and operational variations need to be quantified and accounted for in analysis, especially CN-DBA analysis as needed. While some of this information is captured in Table 15.5-5, it is not clear if the numbers provided in the tables on setpoint and timing account for all instrument uncertainties (using ISA 67.04 or equivalent) and actuation uncertainties (e.g., control rod gate times). Furthermore, while a reference is provided for the uncertainties used in Section 15.5.4.2.1

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for the Generator Load Rejection or Turbine Trip DBA, a summary table on the magnitude of the uncertainties is needed. In many DBAs the only uncertainties applied are to bias the CPR downward and the local power upward, but neither the values used in the approach are provided nor is sufficient justification provided to ensure such an approach bounds all uncertainties.

4. The results are generally presented only for the most bounding fuel exposure, however for completeness a summary of key events or differences as a function of exposure, or transient predictions for key FOM should be provided for each exposure to demonstrate the sensitivity and the bounding nature of the selected condition.
5. A very brief description of the accident transients is provided; however, a thorough discussion of the phenomena, key changes in trends (e.g., core flow reversals), secondary peaks in some phenomena/parameters and/or oscillations should be provided in the write-ups of each section. This would aid the reader in understanding the event trajectory and the evolution of the important phenomena during the transient. Graphs in the PSAR should have better resolution to adequately display rapid and/or brief oscillations. A discussion should be included to explain important transient inflections, especially unexpected ones. For example, in Figure 15.5-5 of the PSAR, there is no obvious initial power increase despite the immediate reduction of feedwater temperature and delayed (by 5 seconds) control rod insertion. Some increase in power in the first 5 seconds of this transient would be expected. If there is a power increase that is obscured by coarse time scaling, then the resolution should be improved per the above. There should also be a clear description of DL1/2 functions for controlling power level during normal operation, AOOs and DBAs. Plots of CPR should be provided where power and flow are changing, especially where the power to flow ratio is increasing.
6. For DBAs, for each shutdown means, an effective backup trip parameter should be provided if available, or a justification for single-parameter trip coverage should be provided (i.e., for reactor with inherent safety and direct trip parameter). See the discussion in Section 3.5.
7. Identify the accident that sets the operating limits and timing requirements for the DL3 trips and rod movement times. Describe how these will be complied with.

6.2 LOCA analyses

BWRX-300 design features used to mitigate containment overpressure scenarios include the normally operating cooling Air Handling Units (AHUs) and the Passive Containment Cooling System (PCCS), in addition to RPV isolation valves and a hardened vent line. In order to preclude pressurization due to the accumulation of combustible gases and the potential for hydrogen explosions, the BWRX-300 containment is equipped with a Containment Inerting System (CIS) which fills containment with nitrogen gas during normal operation. Temperature control is via the AHUs during normal operation and PCCS during accidents.

The accident scenarios that most challenge the containment are the LOCAs. In these scenarios, there is discharge from the pressurized RPV into containment, leading to

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pressurization and temperature increase in containment, in addition to potential releases of radioactivity into containment.

6.2.1 DBA LOCAs

The DBA LOCAs are subject to the following containment acceptance criteria:

- Per Section 15.5.4.6 of the PSAR, LOCA scenarios were developed to demonstrate that the fuel and containment integrity acceptance criteria are met for 72 hours using only passive heat removal systems.
- The peak accident pressure is less than the containment design pressure of 0.414 MPa(g) and reduces to less than 50% of the calculated peak pressure for the most limiting LOCA, within 24 hours.
- The calculated containment shell temperature does not exceed the design temperature of 165.6°C.
- The containment atmosphere remains sufficiently mixed such that deflagration or detonation thresholds are not exceeded.
- The dose limit for DBAs is 20mSv.

Section 1.3 of NEDC-33922P-A R3 [G-10] additionally specifies that “containment venting or leakage shall not be credited for at least 72 hours to demonstrate that the above acceptance criteria are met during a design basis event or accident”.

A review of the accident scenarios was performed with a focus on the LOCA events and containment performance. The PSAR divides LOCAs into Section 15.5.4.6 which addresses LOCAs inside containment and 15.5.9.2 which addresses LOCAs outside containment. The LOCA events (and other events in the PSAR) do not feature any combustible gas generation. Therefore, the DBA LOCAs are evaluated against the acceptance criteria, excluding criterion #4 above, which is always met.

Design Features that limit the consequences of LOCAs include:

- RPV isolation valves on large lines (MS, FW, ICS) that close within 10 seconds of the event.
- MS pipes have a flow limiter placed close to the RPV. Breaks on the MS pipes are postulated to occur downstream of the flow limiter.

The main review findings are presented below.

Unless otherwise noted, general comments below pertain to all LOCA scenarios, whether inside or outside containment.

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- The assumptions and steady state conditions in the PSAR need to be augmented to provide more information regarding the inputs to the analysis and to demonstrate the conservatism of the selected input values. For example, Table 15.5-1 'Key Initial Conservative LOCA Evaluations' provides a limited set of parameters and although it states whether the upper or lower range is selected, it does not show the normal ranges. Furthermore, it is unclear whether containment leakage has been accounted for or not. Per the discussion above, leakage is not to be credited in order to maximize containment overpressure but there is no discussion on leakage in Section 15.5.4.6. All important and relevant assumptions should be tabulated with rationale as to why they are conservative or appropriate for use. It is noted that Section 15.5.1.1.2 of the PSAR provides a description of medium margin events and references to [G-10] for additional analysis details. However, the PSAR would benefit from more explicit documentation of conservatisms rather than a reference to another document.
- The description of the scenario should clearly identify the failed component for each acceptance criterion as per the Single Failure Criterion and provide a rationale for its selection. It appears that in many events, one ICS train is assumed to be the failed component for all acceptance criteria including those related to containment integrity.
- Given that the ICS pools are of different size (ICS C being the smallest of the three), it is unclear whether crediting ICS A and B pools is conservative.
- Acceptance criterion #2 above states that containment pressure reduces to less than 50% of the calculated peak pressure for the most limiting LOCA, within 24 hours. GE-H is requested to clarify whether this is 24 hours after the start of the event and to provide the rationale for this acceptance criterion.

6.2.1.1 DBA Small Break

Small breaks inside containment are those with a break area of less than or equal to 19-mm in diameter and are not isolated. Additional comments specific to the Small Breaks Inside Containment [Section 15.5.4.6.4] are as follows:

- In the SBLOCA scenario, the failed component for the SFC is not identified.
- Reverse flow from containment to the RPV once the pressures in the RPV and containment equalize may introduce noncondensable gases into the RPV and impair heat transfer in the tube side of ICS. It is important to demonstrate with confidence that containment pressure still reduces faster than the RPV pressure to preclude noncondensable gases seeping into the RPV and impair ICS effectiveness. Alternatively, reduction in heat transfer removal characteristics of ICS should be modeled in analysis.

6.2.1.2 DBA Breaks Outside Containment

LOCAs outside containment are treated in Section 15.5.9.2 of the PSAR and state that the break discharge for these scenarios is bounded by the LOCAs inside containment. As such,

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the same containment response from LOCAs inside containment are referenced in Section 15.5.9.2 of the PSAR. Two scenarios are modelled for the releases: 1) the maximum equilibrium iodine concentration during full power operation (0.2 $\mu\text{Ci/gm}$ Dose Equivalent I-131) and 2) the iodine concentration for an assumed pre-accident spike (4.0 $\mu\text{Ci/gm}$ Dose Equivalent I-131). Fuel failures during the DBA are not predicted and therefore, only the activity normally contained in the reactor coolant prior to the break is modelled. In all LOCA scenarios presented in the PSAR, the doses are very low and within the public dose limits.

The breaks outside containment conservatively assume instantaneous ground level release with no building holdup is modelled. The release is modelled over a period of 10 minutes after the event with a total of 99.9% transport of the Turbine Building airspace.

The main comments are presented below:

1. Dose calculations: The discussion on calculation of dose is very high level and it is recommended that additional detail be included in the PSAR to describe the assumptions and methodology employed. The release flow rate is calculated in the PSAR but there are only few high-level statements in Section 15.5.1.2 regarding the codes used.
2. The release is modelled over a period of 10 min, without a documented rationale for using 10-min.
3. Section 15.5.9.2.3 "Large Isolation Condenser Pipe Breaks Outside Containment" states that loads, pressure, and temperatures outside containment, as well as radiological consequences should be evaluated. Once the design and configuration of the lines is complete, the results of these assessments should be included in the Safety Analysis Report.
4. Small breaks outside containment are analyzed in Section 15.5.9.2.5. The first scenario analyzed is a small ICS line break. It is unclear how this small break is detected within 5 seconds and isolated within 10s, similar to a large ICS line break. Small breaks are not isolable per Section 15.2.4.6 and are assumed to remain non-isolated for 72 hours for conservative analysis sequences. The current analysis is performed assuming isolation within 10 seconds, which is not consistent with small break assumptions. PSAR subsections for event description, coolant mass release, release duration and no holdup release to the environment flow rates, and results need to be revisited.
5. The small Instrument Line Break analysis in Section 15.5.9.2.5 states that the line is not isolable. However, Tables 15.5.4.3A and B only provide release information up to 5 hours, instead of 72 hours.
 - a. The instrument line break analysis description refers to various instances of break flow into containment. However, this scenario is a break outside containment. It is unclear how primary coolant flows into containment for this scenario.
 - b. For the instrument line break analysis, coolant mass release calculation is stated to be based on a 10 second ICS isolation time. The release should be 72 hours for

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an unisolable small break, and it should not be related to an ICS line break, as this is a separate scenario.

- c. PSAR subsections for coolant mass release, release duration and no holdup release to the environment flow rates, and results need to be revisited.

6.2.2 DEC LOCAs

For Design Extension Conditions (DECs) or Severe Accidents (SAs), the acceptance criteria are discussed in Section 3.5.6.1.1 and Section 15.1.4 of the PSAR:

- In the Deterministic Safety Analysis, AOs and DBAs with postulated failures of DL2 and DL3 function can be analyzed in EX-DSA. For those events, DECs without core damage default to DBA acceptance criteria as screening criteria to evaluate core damage.
- The ultimate pressure capacity of containment ensures the structural integrity and leak tightness meets the requirements of CNSC REGDOC-2.5.2 for DEC events.
- Robustness Against Combustible Gas Pressure Load Containment can withstand the DEC loads resulting from combustion of gases per CNSC REGDOC-2.5.2.
- Per the PSAR, consistent with REGDOC-2.5.2 Section 8.6.12, containment should be a reliable leak-tight barrier for a minimum of 24 hours following the onset of core damage and continue to provide a barrier against the uncontrolled release of fission products following the initial 24-hour period.

There are no LOCA DEC sequences in the PSAR. As mentioned previously, all LOCAs were treated as DBAs in the PSAR. Section 15.2.4.6 of the PSAR states that “Although the frequency of the largest breaks may be lower than 1E-5/year, all sizes of pipe breaks resulting from various PIEs are conservatively analyzed as DBAs. Further, the largest breaks are instantaneous double-ended guillotine ruptures of the large pipes.” Therefore, it is anticipated that as the PSAR matures, there may be some changes to the classification of events.

As there are no DEC LOCA events in the PSAR, only high-level comments are provided below.

1. DEC LOCA events, such as large LOCAs with failed isolation valves, are to be included in the fault list and analyses included. This needs to be evaluated once the details are included in the POSAR.
2. It is unclear whether the analysis demonstrating compliance to acceptance criteria #3 has been performed. There are no scenarios in the PSAR where hydrogen burn, deflagration or detonation is considered. This is due to the nitrogen inerting of containment. Failures of the Containment Inerting System (CIS) are to be precluded or operationally mitigated (e.g., by reactor shutdown) to ensure that the nitrogen inerting remains intact during operation.

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3. DEC sequences with Molten Corium Concrete Interaction (MCCI) need to be precluded or analyzed, and details regarding the corium shield covering the basemat are required.
4. Similarly, the inclusion of PARS needs to be established or excluded based on a technical rationale. Although identified as a DL4b component in Table 15A-1, there is no discussion of these units or their function in the analysis as there is no analysis involving hydrogen.

6.2.3 Other Containment Topics

6.2.3.1 Containment Isolation

One of the main functions of containment is to isolate and retain fission products in the event of a loss of RPV integrity or line break inside containment. Based on a review of the PSAR, there are a variety of containment penetrations including instrumentation, electrical and mechanical.

Appendix B of CSA N290.3-16 [R-7] lists requirements for containment piping barrier requirements for new builds. REGDOC-2.5.2 has similar requirements. CSA N290.3 prescribes that:

1. Two barriers are required when the piping penetration connects directly to containment and is not part of a qualified closed system outside containment. If the pipe is open to containment 1 h per year or more, two automatic isolation barriers in series shall be provided. If less than 1 h per year, manual, automatic or a combination of manual and automatic barriers shall be used. An example would be the Containment Inerting System Injection line.
2. Two barriers are required, one inside and one outside containment, for piping that connects to the reactor coolant system. For a pipe with barriers open for 1 h per year or more, two automatic isolation barriers shall be provided and fail closed. An example would be the Main Steam System piping.
3. Closed piping systems require a single isolation barrier located outside containment as close as practicable to the containment structure unless the system is provided with continuous leak monitoring and meets the Class 1 or 2 containment requirements of CSA N285.0/N285.6. An example would be the PCCS.

High-level review comments on containment isolation are presented below.

1. Based on the information presented in the PSAR, additional information is required to determine the location of the isolation valves and the containment penetrations. Clause 8.1 of CSA N290.3-16 states "The containment boundary, including the location of all containment penetrations and the associated extensions to the containment, shall be clearly identified and documented". A figure that clearly indicates the containment penetrations and isolation mechanisms should be included in the PSAR, similar to Figure 6.2-3 but for all significant containment penetrations.

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Available flow diagrams would also be beneficial in defining the containment boundary locations.

2. The CSA N290.3-16 isolation requirements listed above appear to be met for many of the major systems penetrating containment. However, as per the above comment, additional information is required to clearly illustrate this. In some instances, it remains unclear whether systems adhere to the requirements in CSA N290.3 both in terms of the location and number of isolation devices, as well as their functionality (manual or automatic).
3. The ability of containment to isolate in a BDBA should be demonstrated to comply with CSA N290.3-16 Clause 9.2.1.4.
4. Per Clause 9.2.2.1 of CSA N290.3-16, "all piping penetrations shall be provided with isolation barriers in accordance with this Standard. Alternative isolation barriers may be used where justification is provided by the designer". It is noted that at the time of writing this report, GE-H is pursuing an alternative approach to containment isolation for the following five systems [G-14]:
 - a. ICS
 - b. BIS
 - c. CRD System
 - d. CIS
 - e. EFS

Per Reference [G-14], the containment isolation valve arrangement was designed in accordance with U.S. NRC regulations in 10 Code of Federal Regulations (CFR) Part 50, Appendix B, General Design Criteria. Reference [G-14] documents GE-H's rationale for the use of the alternate approach and how equivalence is being sought out. The CIS system may require further scrutiny due to its role in post-accident pressure control.

6.2.3.2 Containment Construction and Design Features

The containment structure is a Steel-Plate Composite Containment Vessel (SCCV). It is integral and surrounded by the Reactor Building which is made of Steel Bricks™. Steel Bricks™ are used to form the structural basemat (which is shared with the SCCV), cylindrical exterior wall, interior walls, fuel and cooling pools, refuel floor and roof of the Reactor Bay. Steel Bricks™ are enhanced steel-plate composite structural elements. Steel Bricks™ are used in this FOAK application to a reactor containment structure.

Based on a review of the PSAR, the following main comments are made related to the containment structure:

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1. Robustness of Steel Bricks™, and the effects of irradiation on Steel Bricks™ in particular, need further demonstration. It is suggested that the PSAR provide additional information on the qualification of Steel Bricks™ and the results of full-scale and partial scale experiments that were slated to have been completed in 2021 (accident thermal and pressure loading, seismic loading and thermal loading, impactive loading, accelerated corrosion).
2. The normal operating values for containment are not provided. For AOOs that do not result in energy release into containment or loss of containment heat removal, the quantitative acceptance criteria are said to be the same as in normal operation but there are no quantitative values provided for normal operation in Table 15.3-1 of the PSAR.
3. Chapter 6 of the PSAR provides details on the containment design. The PSAR should elaborate on features such as the rupture disk and containment vent line. Demonstration of these features should be included in the PSAR analysis.
4. It is unclear if there are design features available for filtered venting aside from the containment vent line. There are references within the PSAR to Appendix 15B for further information on containment filtered venting, but currently Appendix 15B only describes the BIS system in some detail and filtered venting is not addressed.

6.2.3.3 Severe Accident Design Features and Analysis

The PSAR does not present any severe accident analyses where there is significant core damage and collapse.

General comments on containment design for a severe accident include:

1. Features designed for DEC orbdba scenarios are mentioned but their description is incomplete. Features such as the containment filtered venting system, Passive Autocatalytic Recombiner System (PARS), and quench tank require more description.
2. The PSAR mentions a Containment Corium Shield to prevent Molten Corium Core Interaction. Demonstration that the core melt event is practically eliminated or mitigated via other measures is required.
3. Per CSA N290.3-16 Clause 9.1.3, "The selection of the containment concrete material for new builds shall take into account the effects of severe accidents involving interaction between a melted reactor core and concrete," including the effects of combustible gases, aerosols, and corrosive conditions.

A similar statement appears in REGDOC-2.5.2 [R-2] in Clause 5.3.4.1. The PSAR does not have any scenarios in which combustible gases are evaluated, based on nitrogen inerting of containment. Use of coatings on exterior surfaces and use of metallic insulation is mentioned in the PSAR but is not described in detail. Additional information would augment the PSAR statements that the requirements of CSA N290.3-16 and REGDOC-2.5.2 are met.

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6.3 Bounding Accidents for Dose

Section 15 of the PSAR provides a high-level description of the methodology used and select assumptions for radionuclide release and dose calculations.

Comments on the radiological consequences of the LOCA outside containment scenarios were presented in Section 6.2.1.2 and are not repeated here. Additional comments can be found in Appendix B.

6.3.1 Fuel Load Error

In addition to the LOCAs outside containment previously discussed, there are other events that lead to releases of radionuclides.

These include:

1. Fuel Loading Error Event (FLE) in Section 15.5.4.3.1 of the PSAR – Mislocated Fuel Bundle, and
2. Inadvertent Single Control Rod Withdrawal at Power (ICRW) in Section 15.5.5.3.2 of the PSAR.

The FLE event provides dose results in Table 15.7.3-2, but the ICRW event does not provide dose results and refers to the PSA.

- The FLE dose calculation description in the PSAR provides less detail than what is provided in the LOCA sections. Inventories, flow rates, X/Q values up to 30 days and filtering details are lacking. Comments in Appendix B on Section 15.5.4.3.1 of the PSAR describe a lack of information/results.
- The fission product inventory is multiplied by 3.5 (1.4x2.5) to account for variations in fission product inventory and bundle power. More details regarding the basis for this factor are required.
- The X/Q value in the table of results, Table 15.7.3-2, is only provided for the 0-2 hours timeframe. Additional clarification is required regarding X/Q values used for the 30-day duration.

6.3.2 Fuel Handling Accidents

Fuel Handling accidents are discussed in Section 15.5.8 of the PSAR. The fuel handling is assumed to occur a minimum of 24 hours after shutdown. Therefore, a decay time of 24 hours is applied in the analysis. The event considers 60 isotopes which are the dominant contributors to immersion and inhalation doses from airborne activity. Particulate isotopes are retained by water pools, and noble gases and gaseous forms of iodine are assumed to be available to escape the water in the reactor cavity pool or fuel pool. Credit is taken for pool scrubbing per US NRC Regulatory Guide 1.183. Once the radioactivity is released from the reactor cavity pool, it is assumed to mix with the free air volume in the refuelling outage floor

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and crane area. Leakage from this area is assumed to occur at a flowrate that transports the entire release to the environment in 2 hours. The rationale for this time duration should be provided.

6.4 Probabilistic Safety Analysis

Section 15.6 of the PSAR provides the PSA scope and methodology. It also includes a summary of PSA results and insights from the PSA documented in [G-12]. Detailed numerical results of the PSA are presented in section 15.7 of the PSAR.

In addition to Sections 15.6 and 15.7, PSAR includes two appendices (Appendix 15A and Appendix 15B) which are relevant to the PSA, and which were reviewed by the IPR team. This section of the IPR report will summarize team findings related to section 15.6, section 15.7, Appendix 15A and Appendix 15B of PSAR Revision 0. Those findings are as follows:

PSA Scope, Methodology and Results

The PSA scope and methodology presented in Section 15.6 are consistent with Canadian requirements [R-4] and past OPG PSA practice. The numerical analysis results in Section 15.7 for BWRX-300 Severe Core Damage Frequency (SCDF) and Large Release Frequency (LRF) meet Canadian Requirements with a wide margin.

The PSA provided in Section 15.6 of PSAR Rev. 0 is a PSA summary rather than a “complete” PSA. However, the IPR team concludes that it satisfies CNSC requirements outlined in Clause 4.4.1 of [R-1].

Emergency Makeup

Concern for the apparent lack of makeup water for events such as pipe breaks that cannot be isolated has been addressed in Section 15.6 of PSAR for PSA purposes. Specifically, Section 15.6 includes a discussion of the injection function of the CRD Hydraulic Subsystem. This important inventory makeup capability is described in CRD design information as well (e.g., Section 4.6.2.2). Given the important safety function provided by the CRD Hydraulic Subsystem, the IPR team suggests that PSAR Section 15.6 be further augmented by inclusion of a summary of relevant hydraulic capacity and hydraulic component redundancy information and include appropriate deterministic safety analysis covering the first 7 days when the CRD water injection can be credited. The injection function is an important factor for SCDF and LRF minimization. Inclusion of capacity and redundancy information would assist assessment of the numerical risk reduction afforded by this safety function.

Containment Overpressure and Venting

As discussed in Section 7.5 of this report, severe accident analysis and corresponding results are required to address potential severe core damage leading to containment failure and large release (as happened at Fukushima), with a focus on containment emergency cooling and venting provisions for DEC, BDBA and severe accident events.

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Based on the information presented in 15.6.1.10 of PSAR, it is apparent that the Passive Containment Cooling System (PCCS) cannot prevent containment challenge (due to overheating) for all accidents; for some events, venting is required to maintain containment integrity. Section 15.6.1.10 of PSAR also discusses venting and states that a hardened vent is included in the design and that this vent is filtered. However, Sections 6.3.5 and 6.3.6 of PSAR state that the need for a filter has yet to be confirmed.

Venting provisions (including filtration, if any) need to be finalized and unambiguously described in the PSAR. The PSAR credits pool scrubbing in the analysis. However, other sections of the PSAR are ambiguous about whether another filter will be installed. If a filter is included in the design, the PSAR description should provide salient design details such as flow capacity, pressure rating, filtration efficiency etc.

Design Extension Conditions – Appendices 15A and 15B

As indicated above, Appendices 15A and 15B are complementary to each other and complementary to the PSA. These appendices deal with Design Extensions Conditions (DEC) with particular emphasis on identifying accident sequences and conditions that can be regarded as “practically eliminated”. This is an important part of compliance with REGDOC-2.5.2 [R-2] requirements for “defence in depth” (Section 6.1) and DEC (Section 7.3.4). Taken together, Appendices 15A/15B identify accident sequences that have been “practically eliminated”; they highlight important “complementary design features” that help provide “practical elimination” for these sequences; and they supply the logic to explain how “practical elimination” has been achieved for these sequences.

Although Appendices 15A and 15B provide necessary and valuable information, the IPR team concludes that they would benefit from the following suggested refinements:

- Discussions in the two appendices are complementary. Readability would be improved if the two appendices were combined and integrated into a single appendix titled “Design Extension Conditions” or “Beyond Design Basis Accidents”.
- The title of Appendix 15A is incorrect. The current title refers to “Reference Source Term for Conditions That Are Practically Eliminated”. Conditions that are “practically eliminated” require no further attention in the plant design, so there is no need to define a reference source term. The REGDOC-2.5.2 requirement is to define a reference source term for Design Extension Conditions, i.e., conditions that have **not** been “practically eliminated”.
- Regardless of the Appendix 15A title, that appendix currently provides no source term. This omission will have to be corrected in the POSAR.
- Appendix 15B does an excellent job at describing the Boron Injection System (BIS) complementary feature. However, it should also provide similar information for the complementary systems described elsewhere in 15.6 (e.g., control rod water injection, ultimate pressure relief).

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6.5 Methodology for Out-of-Core Criticality Safety Analysis

The PSAR discusses the OOC in Section 15.5.9.3. Most of this section focuses on the assessment of dose consequences at the site boundary of an out of core criticality event. The introductory paragraph discusses the choice of the representative accident based on a separate OOC demonstration document. It identifies the HDFS racks, and the RAJ-II transport inner container as the aspects to consider.

The text indicates that the HDFS racks were found to be below the USL for normal and credible abnormal conditions and states that the RAJ-II container will be used as a basis for the representative accident in the dose assessment. In the dose assessment of this representative accident, the physical and design barriers that prevent an OOC occurrence are described. It is not clear, however, that the event sequence involving the RAJ-II container has been assessed to be non-credible. It is recommended that this discussion be revised for clarity to show that all credible events remain subcritical as required per REGDOC-2.4.3, Clause 2.3.2.2.

The remainder of Section 15.5.9.3 in PSAR describes a representative out-of-core criticality accident scenario in accordance with requirements in Clause 16.4 of REGDOC-2.4.3. This requirement is for a “representative OOC accident scenario [to be] investigated to show that the dose consequence at the site boundary does not exceed generic criteria to trigger a public evacuation.”

The defined accident is on the reload floor of the reactor building with an unsafe number of RAJ-II inner containers stacked too close together and subjected to a catastrophic flood. The spent fuel pool storage racks are based on a lattice with optimal configurations and spacing controlled by engineered controls, whereas the accident scenario involving the RAJ-II ICs relies partly on administrative limits that are less reliable. Therefore, an accident involving the RAJ-II container configuration is more likely and using it as the defined representative accident is reasonable.

The subsequent dose assessment from the above representative accident should follow Clause 16.4.1 of REGDOC-2.4.3, which specifies using an estimated fission yield from the accident to determine the appropriate source term. The number of fissions in the proposed accident is estimated by using a comparative accident from the literature. The defined comparative accident is based on Reference 15.5-14 of the PSAR. Note that REGDOC-2.4.3 quotes an NRC guide, which in turn refers to the same Reference 15.5-14 of the PSAR, so this is acceptable. The CNSC defined representative accident in Clause 16.4.1 of REGDOC-2.4.3 is 1E19 fissions unless otherwise justified. The assessment finds a comparative accident using the same general type of material as the real system (UO₂ fuel rods moderated by light water). This gives 3E18 and one power excursion of < 180 s. The document does not provide a justification for the total number of fissions of 4E19 listed in Table 15.5-46, which is apparently used as the basis for the dose analysis. If a scaling factor was used, it should be mentioned. Note that the assumed total number of fissions is higher than the CNSC requires for dose calculations and is reasonable.

The calculation methodology of the dose resulting from the representative accident discussed above is described in PSAR Section 15.5.9.3. This includes high-level descriptions of the

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computational models, the source term determination, and the dose rate determination process. This high-level description is reasonable and meets technical expectations.

The Darlington Site Environmental Impact Statement (EIS) [O-4] Executive Summary states that fuel for the proposed new reactor is enriched between 1 and 5 wt% ²³⁵U. This remains true of the BWRX-300. The report also states that OOC events would not be credible with appropriate design and control procedures in place. However, modelling of a hypothetical OOC event has shown that workers in the vicinity would be subject to substantial risk but the public would not (based on shielding and distance). The results presented in the PSAR show that with the planned shielding, the representative dose is well below evacuation criteria and the conclusions of the EIS remain valid.

6.5.1 Out of Core Criticality Safety Assessment

A supporting report [G-8] was also reviewed as it provided an overview of the expected program management and technical practices that will be applied for Out of Core Criticality Safety. The introduction to the report states that its purpose is “to provide information on GE-H’s understanding of the CNSC expectations and regulatory requirements in the programs pertaining to out of core criticality safety assessments. It also confirms that the BWRX-300 out of core criticality program, as it is evolving, is meeting the CNSC expectations in REGDOC-2.5.2 “Design of Reactor Facilities: Nuclear Power Plants” [R-2] and REGDOC-2.4.3 “Nuclear Criticality Safety” [R-5] for provisions for out-of-core criticality safety.” It is understood that the OOC safety analysis had not been finalised at the time of the IPR, but this document provided insight into the approach. Overall, this document describes the process and main highlights of the OOC analysis well with reference to a more detailed report. A few areas for improvement are noted in the following paragraphs.

Section 2.1.1 (Process Management) establishes Upper Subcritical Limits (USLs) in accordance with Clause 2.3.2.2 of REGDOC-2.4.3 [R-5]. This section quotes two different USLs but does not explain why they are different. There should be consistency of approach or further discussion.

Section 3 of [G-8] describes the decision to not include a criticality alarm system. This section indicates that the detailed analysis performed in Rev 0 of [G-8] demonstrates that “inadvertent criticality will not occur and that a criticality alarm system is not necessary for the BWRX-300”. REGDOC-2.4.3 [R-5] does not give guidance on how the criterion of inadvertent criticality not occurring is to be assessed, but as “credible” is defined as once in a million years, this would generally require a very robust argument. The discussion presented in [G-8] does not meet this threshold. Other supporting documents may offer more information but were not reviewed.

Section 4.1 of [G-8] describing the OOC safety aspects of the BWRX-300 fuel storage design contains the statement: “The fuel storage area shall contain a criticality alarm compliant with REGDOC-2.4.3 Section 3.0 as discussed in Section 3.0 of [G-8].” However, as described in the previous paragraph, Section 3.0 of [G-8] indicates that no criticality alarm is required. This means that Sections 3.0 and 4.1 of [G-8] are not consistent regarding the requirement for a criticality alarm and this discrepancy should be resolved.

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7.0 SELECT GENERAL TOPICS FOR DISCUSSION

7.1 Independence and Separation of Safety Systems

Historically, Canadian safety/licensing regulations have included requirements related to independence and separation of process systems from safety systems, and independence and separation of safety systems. The positive void reactivity behaviour of CANDU reactors led to an explicit Canadian requirement for two independent and fully effective Shutdown Systems for CANDU reactors, and an independent reactor regulating system. The serious accident that occurred on the NRX experimental reactor in 1952 led to important lessons related to separation, diversity and independence of control and safety systems, shutdown system depth, and human factors.

Regulations in all jurisdictions acknowledge the importance of independence and separation of mitigating defences for reactor safety. However, specific regulatory approaches for achieving independence and separation can differ. Different operating experience and different designs for reactors that have been licensed in other jurisdictions have led to different approaches and different requirements for independence and separation. For example, REGDOC-2.5.2 does not require two fully capable Shutdown Systems for reactor designs with inherent safety (e.g., with negative fuel temperature reactivity feedback and negative void reactivity feedback as in BWRX-300 design); instead, it requires, two means of shutdown with more stringent requirements for the fast-acting means of shutdown. For another example, REGDOC-2.5.2 allows sharing of process and safety equipment if accompanied by appropriate safeguards against common cause failures.

The CNSC has now updated Canadian regulations to make them “technology neutral” achieving general consistency with the IAEA standards and, thereby, to explicitly allow design approaches that have been proven in other jurisdictions, even if those approaches differ from past Canadian practice. That is the case for independence and separation of safety systems. Per Clause 8.4 of REGDOC-2.5.2, there is no longer a requirement for two fully effective Shutdown Systems. Instead, two separate, independent, and diverse shutdown means are required, with the requisite redundancy provided in the fast-acting shutdown means. Similarly, per Clauses 7.6.1 and 7.6.5 of REGDOC-2.5.2, there is no longer a requirement for strict separation of process and safety equipment. Instead, REGDOC 2.5.2 establishes requirements for shutdown reliability and resistance to common cause failures. However, it does not presume that past Canadian practice in these areas is the only way to achieve the required level of safety.

The IPR team concludes that the BWRX-300 design has the capability to comply with Clause 8.4 of REGDOC 2.5.2 requirements for two separate, independent, and diverse means of shutting down the reactor. This capability needs to be demonstrated through additional CN-DBA safety analysis, since PSAR Chapter 15 demonstrates effectiveness of only the DL3 shutdown means for mitigating DBAs. The two separate, independent, and diverse shutdown means are considered by the IPR team to be the DL3 means of shutdown (the “*fast-acting*” shutdown means) and the DL4a (the *second*) means of shutdown. While sharing the same control rods within the core, their ex-core components are not shared, they are independently instrumented and controlled, and their motive forces of inserting the control rods into the core are diverse. In addition to the DL3 and DL4a shutdown means, PSAR shows many postulated

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initiating events can be mitigated by the DL2 shutdown function which also uses the same control rods. For REGDOC-2.5.2 compliance purposes, the IPR team does not consider the DL4b Boron Injection System (BIS) as an effective shutdown means as it lacks automatic control features and dedicated instrumentation but provides an additional means to shutdown the reactor for DEC's.

As noted above and in Section 3.5 of this report, additional demonstration of adequacy of current BWRX-300 design features for shutdown means and their reliability is recommended, e.g., through additional CN-DBA safety analysis for crediting only the DL4a shutdown means, if required or through the existing EX-DEC analyses included in Chapter 15 depending on the combined initiating event frequency and the probability of failure of DL3 scram, in addition to the CN-DBA safety analysis crediting only the DL3 shutdown means, since REGDOC-2.5.2/REGDOC-2.4.1 require demonstration of effectiveness of two shutdown means for mitigation of DBAs. Adoption of the "alternative approach" provisions in Clause 11 of REGDOC-2.5.2 may be used to address residual issues.

Additional discussion on independence of the two shutdown means is provided in Sections 3.5, 5.2 and 7.4 of this report. The following discussion provides additional information regarding IPR team findings related to separation of process and safety functions.

The BWRX-300 design includes a number of safety components and provisions that are common to multiple levels of defence-in-depth (and common to process and safety systems). Examples include:

- Control rods are common to all levels of defence in depth (DiD);
- Some elements of hydraulic scram are common to DL2 and DL3/DL4a; and
- Reactor pressure vessel coolant makeup, e.g., for some small breaks after 3 or 7 days, is common to all defence levels.

The common components used by safety and process systems are generic to successfully operating BWRs. The BWRX-300 rod design is very similar to the design used in many BWRs that have safely operated for many years with this arrangement. BWRX-300 design brings in additional accident detection and rod insertion mechanisms. Reference [G-3] provides an excellent discussion of BWRX-300 reactivity control including a detailed discussion of neutron absorbing rod design features that have been incorporated to ensure that rod safety performance cannot be hampered by their use for control. While sharing of the control rods is a potential area of concern, some sharing is allowed in Clause 8.4 of REGDOC-2.5.2. The BWRX-300 design uses the same neutron absorbing rods for reactor control and various means of reactor trip (scram). The IPR team concludes that this sharing has been implemented in accordance with separation requirements stipulated in REGDOC 2.5.2 Clauses 7.6.1 and 7.6.5. It will be necessary to further justify the sharing of components. To confirm adequacy of existing shutdown means of BWRX-300, IPR team also recommends quantitative reliability calculations to demonstrate that shutdown failure frequency of less than 1E-5 per demand is achieved as noted in Clause 8.4.2 of REGDOC-2.5.2.

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7.2 Alternative Approach and Graded Approach

As discussed above, REGDOC-2.5.2 and REGDOC-2.4.1 allow for alternative approaches that provide an equivalent level of safety and Graded Approach, respectively.

The IPR team has identified specific instances where BWRX-300 approaches may differ from REGDOC requirements but are supported by past BWR practices that are well-established and which have been approved by regulators in other strongly regulated jurisdictions. This, in conjunction with the large improvements in risk metrics, provides a strong basis for the safety of the BWRX-300 design.

It is important to note that the CNSC is taking a number of steps to assist their review (and approval) of alternative approaches that are new to Canada. Reference [M-2] describes some of these steps. Reference [R-8] provides a specific example of the CNSC collaborating with the US-NRC to harmonise CNSC/NRC requirements for BWRX-300.

As discussed in Section 3.4.1 of this report, GE-H has started to develop and document “alternative approach” justifications for BWRX-300. The first step in that direction ([G-14]) addresses Safety System independence, means of shutdown and containment isolation. It is expected that [G-14] will be used as a template for additional “alternative approaches” as required and that all such approaches will be appropriately included in the POSAR or FSAR.

Examples of an “alternative compliance” or a “graded approach” in the PSAR could include:

1. Independence of Safety Systems (see Section 7.1 and Reference [G-14]),
2. Two Independent Shutdown Means (see Reference [G-14]),
3. One or Two-trip parameter coverage for each Shutdown Means depending on inherent reactor characteristics,
4. Containment Isolation (see Section 6.2.3.1 and Reference [G-14]), and
5. ICS as partial fulfilment of ECCS.

As discussed in Sections 3.4.1, 5.2 and 7.2 above, there is no current regulatory requirement for two independent Shutdown Systems for BWRX-300. The two BWRX-300 means of shutdown for DBAs, considered to be the DL3 scram and DL4a fast control rod insertion by the IPR team, may already be fully compliant with current REGDOC-2.5.2 and REGDOC-2.4.1 requirements. If this proves to be the case, there may no longer be a need to consider BWRX-300 means of shutdown as a non-compliant approach requiring Alternative Approach justification per REGDOC-2.5.2 Clause 11.

7.3 BWRX-300 Stability

For the purposes of this report, the discussion of “core stability” includes reactor power instabilities, thermal hydraulic (flow) instabilities, coupled power/flow instabilities and startup

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instability. This section will summarize the IPR Team's review of these instabilities and the potential impact of these instabilities on BWRX-300 safety.

Power and flow instabilities are not new to Canadian nuclear power plant licensing. Although exceedingly infrequent, events have occurred where CANDU units have had to shut down when control systems were unable to adequately limit spatial power swings. Similarly, safety analysis for CANDU units discusses the potential for so-called "figure-of-eight" flow oscillations under certain design basis accident conditions.

The first step in the IPR Team assessment of BWRX-300 stability was to assess GE-H stability design plans relative to REGDOC-2.5.2 stability requirements.

Sections 4.4.1 and 4.8 of the PSAR outline the BWRX-300 design bases for meeting REGDOC-2.5.2 stability requirements. Section 4.8.3 describes three modes of oscillations for Type 2 instability. The first two modes are core-wide and region-wide oscillations and have been addressed in Section 15.5.2.4. The third mode is about single channel oscillations where the CPR could decrease due to a change in thermal hydraulic conditions even at unchanged bundle power as a result of a parallel channel effect. This has not been addressed in the PSAR.

Important aspects of the BWRX-300 stability design approach can be summarized as follows:

1. BWRX-300 is designed so that coupled neutronic and thermal hydraulic power oscillations are not possible throughout the entire normal operating region,
2. BWRX-300 provides appropriate defense lines to prevent oscillations from becoming unacceptably large under AOO and DBA conditions, and
3. As a backup, BWRX-300 implements a system to detect and suppress instabilities, should they occur.

If properly implemented, this design approach should be acceptable for operational purposes and for Canadian licensing.

The second step for the IPR Team was to review the PSAR analysis to determine if the analysis results are consistent with the stated design approach. In the PSAR, general stability analysis information is provided in Section 15.5.2.4. Instability during normal operation and AOO, DBA and DEC events are shown in PSAR Sections 15.5.2.4 and 15.5.3 (AOO), 15.5.4 (DBA) and 15.5.5 (DEC).

Available analysis results in PSAR Section 15.5 indicate that GE-H stability design goals have been generally met in BWRX-300. Power swings are negligible for the analyzed AOO events. For DBA and DEC events, some accident scenarios (e.g., turbine trip) exhibit significant power swings that are indicative of potential stability issues. However, the indicated magnitude of these swings is limited sufficiently to preclude fuel and/or plant damage. The safety analysis may have to be updated to address some broader analysis comments captured in the rest of this review report.

The third and final step in the IPR Team review of stability was to complete a review of OPEX related to power oscillations. This OPEX review is discussed in Section 7.6 below. This

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review concluded that power oscillations were observed in BWRs prior to ~1999, but that a number of corrective actions were taken that have been successful in preventing adverse oscillation events for forced circulation BWRs since then.

Based on the preceding (stability design, stability analysis and stability OPEX), the IPR team concludes that stability is unlikely to emerge as a significant operational risk for BWRX-300. However, as indicated above, the PSAR analysis needs to be completed for normal operation as well as updated to reflect comments captured in this review report. In addition, the IPR team has identified several areas where the PSAR should be revised to improve clarity and/or to strengthen the stability safety case. Some examples include:

- It is suggested that PSAR Figure 4.4.2 (Power / Flow Map) be revised to show flows and powers for selected AOO and accident scenarios (especially limiting ones). The basis for this figure should also be provided, such as accounting for code prediction uncertainties associated with natural circulation.
- Similar to the comparison of power/flow maps shown for other natural circulation BWRs, they should be included in the PSAR to show the margin to stability boundary.
- It is suggested that the safety analysis clearly identify the sensitivity of stability margin to important input assumptions such as inlet subcooling, trip delays, rod drive delays, parallel channel effects and balance of plant feedback. Use of decay ratio (DR) as the sole acceptance criterion should be justified considering its strong dependence on inlet subcooling.
- An improved description of the analytical test is needed for PSAR Section 15.5.2.4, e.g., how the velocities were perturbed in different channels for demonstration of the preclusion of region-wide oscillations.
- The PSAR stability description should provide a discussion and a reference that confirms adequate validation of the TRACG stability analysis models for BWRX. This discussion should include suitable uncertainty/accuracy estimates for TRACG results. It is noted that TRACG has been validated for ESBWR and traditional BWRs.
- The PSAR should include analyses of the start-up procedure as this is a natural circulation plant.
- Balance of plant (BOP) could affect the oscillation due to feedback effect such as in loss of feedwater heater transient. GE-H to confirm whether the plant model includes BOP.

7.4 Anticipated Transients Without Scram (ATWS)

If an operating reactor fails to shutdown (scram) after an initiating event that requires a trip, core disassembly can result from core overpower and/or overheating. In Canadian PSA parlance, the result is “Fuel Damage Category 1” (FDC1), which is the worst possible core damage classification. Energy release to containment for such an event would be large and is difficult to quantify. Therefore, the common Canadian approach is to assume that the energy release would lead to consequential containment failure and a “Release Category 1” (RC1) large release from the plant; RC1 is the worst possible release classification.

Although such an event sequence is very unlikely, the large potential consequences have resulted in the topic receiving intense scrutiny for many years and in many jurisdictions. In Canada, the issue is usually referred to as “Failure to Shutdown” and led to the adoption of two fully independent and fast-acting Shutdown Systems each with high reliability as a design

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requirement for CANDU reactors and hence failure to shutdown or ATWS is not included as a DBA for CANDUs (see reference [R-9]). The Canadian requirement for two fully capable Shutdown Systems for CANDU reactors was largely driven by CANDU reactivity feedback characteristics, particularly the positive void coefficient. In the US, such events are called "Anticipated Transients without Scram" (ATWS) and have been debated for over 50 years. The first detailed US study on the topic (WASH-1270) was released in 1973. This study has been revisited many times in the ensuing years. The reports EPRI NP-251 [M-3] and NUREG-1780 [U-1] provide a good summary of relevant history.

Unlike Canada, the US decided that two independent and fully capable shutdown systems were not required. This conclusion extends to BWRX-300. There are technological differences between CANDU and US LWRs that account for the different shutdown approaches. A key difference is that US LWRs (including BWRX-300) exhibit negative void reactivity and negative fuel temperature feedback.

In recent years, many key CNSC regulatory documents (REGDOCs) have been revised to make them "technology neutral" and, thereby, facilitate licensing of non-CANDU designs. As a result, [R-9] has been superseded by new REGDOCs (e.g., REGDOC-2.5.2) that explicitly allow alternative shutdown approaches that are safe and appropriate for BWRs, but which would not have been compliant with previous rules that were developed specifically for CANDUs [G-4]. In both Canada and the US, the key issue has been how reliable must shutdown function be to reduce the probability of core disassembly due to lack of shutdown to an acceptably low figure. CANDU reactors have a positive void reactivity coefficient which led to a hard requirement for two fully effective, fast acting Shutdown Systems for CANDUs. In the US, negative void feedback and negative fuel temperature feedback made it possible to avoid having to add a second, fast-acting fully effective shutdown system in light water reactors (including BWRs).

In addition to the DL3 shutdown means, the BWRX-300 design has design features that ensure effective and reliable shutdown means: a strong DL2 shutdown function for mitigating AOOs (and for DBAs, but appropriately not credited in CN-DBA analysis), provision of an independent DL4a shutdown means, and the manual DL4b (BIS). The DL2, DL3 and DL4a provide redundant and highly reliable means to automatically insert the control rods into the core. Reference [G-5] provides a good summary of the safety case for the shutdown approach used in BWRX-300.

The IPR team has reviewed the case presented in [G-5]. Using [G-5] and other relevant information, Reference [G-14] is a good first step towards formulating an "alternative approach" case for BWRX-300 means of shutdown -- although the BWRX-300 design may already be compliant with the REGDOC-2.5.2 requirements and guidance without a need for use of an alternative approach as discussed elsewhere in this report (e.g., Section 3.5). The safety case presented in [G-14] was prepared in accordance with Clause 11 of REGDOC-2.5.2 and has been discussed with the CNSC. The safety case presented in [G-14], augmented as discussed in this report, can be used to show that the BWRX-300 design complies fully with the underlying safety intent of the means of shutdown requirements of REGDOC-2.5.2, namely the frequency of failure to shutdown on demand is to be less than 1E-5.

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7.5 Operating Experience Review: GE Reactors

Numerous BWRs have safely operated in multiple jurisdictions for many years. This operating experience (OPEX) has created a large database of useful information. The following is a non-exhaustive list of major items resulting from an OPEX review, likely to be the subject of discussion as BWRX-300 licensing moves forward:

- Browns Ferry fire (1975),
- Browns Ferry partial scram failure (1980),
- Severe core damage at Fukushima Daiichi (2011), and
- Containment failure and large release at Fukushima Daiichi (2011).

Browns Ferry Fire

In 1975, a serious fire at Browns Ferry disabled multiple safety barriers, leading to the near uncovering of the reactor core. There was no core damage or radioactivity release. However, the event was a significant “near miss” and led to numerous changes for fire protection in US nuclear plants. There are many good descriptions of this event. Reference [U-2] provides a concise summary of the event and focuses on the important topic of the changes that were made to upgrade US nuclear plant fire protection to prevent event recurrence. The fire protection improvements arising from the Browns Ferry fire have been incorporated in the design of BWRX-300. The IPR team suggests that GE-H be asked to provide a concise Executive Summary of the event and the follow-up improvements that were made (and which are included in BWRX-300).

Browns Ferry Partial Scram Failure

As alluded to in Section 7.4 above, Browns Ferry experienced a partial scram failure in 1980. As discussed in References [G-4] and [G-5], the primary contributing factor for this failure (insufficient scram discharge volume) has been eliminated from the BWRX-300 design [G-5].

Power Oscillations (Multiple Units)

Power oscillations were observed on multiple units in the period 1982 – 1999. References [M-4] and [G-6] discuss some specific events and actions taken to address the observed oscillations. Table 1 of Reference [M-5] provides an excellent summary of relevant OPEX associated with these power oscillations. As evidenced by the lack of adverse events since 1999, the corrective actions that were taken to address these oscillations are considered effective at least for the operating fleet of BWRs that employ jet pumps. BWRX-300 has features that differ from most of the operating BWRs, namely lack of jet pumps and reliance on natural circulation, and the ICS discharge location. The IPR team concluded that this issue was important enough to warrant further review. That review is summarized in Section 7.3 above.

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Severe Core Damage at Fukushima Daiichi

The multi-reactor accident that occurred at Fukushima Daiichi in 2011 is an obvious and important piece of BWR OPEX.

For BWRX-300, the Isolation Condenser System (ICS) is the key line of defense against core damage following station blackouts and other common mode events such as the earthquake and tsunami that hit Fukushima Daiichi. Since Fukushima Daiichi Unit 1 also included an ICS, the IPR team has reviewed the BWRX-300 design and concluded that this design includes significant ICS improvements that preclude a repeat of the Fukushima ICS failure.

The IPR team suggests that GE-H be asked to provide a concise Executive Summary to demonstrate ICS effectiveness and reliability. This summary should include (but should not be limited to) the following points:

- The Fukushima Daiichi Unit 1 ICS correctly and automatically deployed by opening ICS isolation valves following detection of the initiating earthquake that occurred off Japan's coast in 2011.
- Electrical power at Fukushima Daiichi did not fail immediately following the earthquake. While power was still available, Fukushima plant operators closed ICS isolation valves to prevent sudden overcooling of the RPV, taking the IC system out of service. The BWRX-300 design includes new features such as the addition of a parallel ICS condensate return valve with throttling capability (to control normal cooldown), while ensuring both ICS isolation valves in parallel are fully open under accident conditions.
- Power was subsequently lost when the tsunami triggered by the earthquake hit the shore. With power gone, Fukushima Daiichi Unit 1 operating staff could no longer re-open the ICS isolation valves that had been closed earlier when power was available. This led to a rapid progression to severe core damage in Unit 1.
- A number of operational changes have been made within the BWR fleet to prevent the recurrence of what occurred in Fukushima Daiichi Unit 1. In addition to these changes, the BWRX-300 design includes an additional ICS isolation valve that fails open on loss of power. This parallel fail-safe valve precludes the recurrence of the ICS isolation that occurred at Fukushima Daiichi Unit 1.

Containment Failure and Large Release at Fukushima Daiichi

As discussed in the preceding section, during the 2011 accident at Fukushima Daiichi Unit 1, the untimely isolation of ICS contributed to severe core damage in that unit.

Inability to cool the badly damaged Unit 1 core then led to unacceptably high temperature and pressure in the Unit 1 Primary Containment Volume. This, in turn, led to containment failure (likely at the containment upper flange) and a large release of radioactivity to the environment.

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As discussed in the preceding section, BWRX-300 ICS improvements have significantly reduced the likelihood of ICS failure and consequent severe core damage. Since a large release cannot occur without severe core damage, ICS improvements can be regarded as large release prevention as well as severe core damage prevention. For accidents that progress to severe core damage, release can be mitigated by operation of containment cooling and venting provisions. The BWRX-300 containment design includes a Passive Containment Cooling System (PCCS) and hardened venting capability (including water scrubbing). As discussed above in section 6.4, the Level 2 PSA does not fully discuss the release mitigation provided by these features. To address the possibility of containment failure and large release (as happened at Fukushima), GE-H should provide a concise executive summary of steps that have been taken in BWRX-300 to “practically eliminate” the potential for a large release.

7.6 Operating Experience Review: Other Boiling Light Water Reactors (Non-GE)

In addition to the GE-H BWR fleet, other/similar boiling light water reactors have operated in the UK [Winfrith Steam-Generating Heavy Water Reactor (SGHWR)], Japan [Fugen Advanced Test Reactor (ATR)] and Canada [Gentilly-1 CANDU Boiling Light Water (CANDU-BLW) Reactor]. The IPR team reviewed OPEX from these plants as well.

The IPR team review of this non-GE-H OPEX did not reveal any significant new information with direct impact on BWRX-300 licensing. However, this review did shed some useful light on an important aspect of the BWRX-300 design.

As discussed in Chapter 2 of Reference [M-6], the earliest (and most successful) of the non-GE-H BWRs was the Winfrith SGHWR. The SGHWR was a vertical, pressure tube, heavy water moderated, boiling light water reactor. It produced 100 MWe and was built in the United Kingdom. This reactor operated as a commercial prototype and produced reliable power to the national grid system from 1968 to 1990, when it reached the end of its design life. Because light water was used as coolant, slightly enriched uranium was used to offset the resulting neutron absorption. Enrichment contributed to the reactor having a negative void coefficient, which provided stabilizing negative feedback to coolant boiling and void formation in the core.

Gentilly-1 was a Canadian design that was similar to and was modelled on the SGHWR. It was designated as the CANDU Boiling Water Reactor or CANDU-BWR and also known as the CANDU Boiling Light Water Reactor or CANDU BLW. It had a capacity of 250 MWe and produced first power in 1971. Unlike the SGHWR, Gentilly-1 was not a successful design and operated for only 183 full power days. Because the reactor used natural uranium and light water coolant, it had a very strong positive void coefficient that made the reactor difficult to control. Rather than invest in expensive modifications to improve spatial control, it was decided to shut the plant down (see Chapter 2 of [M-6]).

The final pressure tube, heavy water moderated, light water cooled BWR was the Fugen plant. Fugen was a domestic Japanese design. The reactor was started in 1979 and operated successfully until 2003. Similar to the SGHWR (and different from Gentilly-1), the choice of reactor fuel provided a very small void coefficient that supported stable spatial control. Unlike the SGHWR, Fugen used a full Mixed Oxide (MOX) fuel core and used the

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plutonium content of this fuel to deliberately “fine tune” void reactivity in the range from very slightly negative to very slightly positive (Reference [M-7]).

As indicated above, the IPR team found nothing in the OPEX for these three reactors that has any significant direct bearing on BWRX-300 licensing. However, the OPEX review highlighted an important lesson about void reactivity for water cooled reactors and stable spatial control of reactor power. Specifically, it is important for the core void reactivity coefficient to be either negative or only very slightly positive. Strongly positive void reactivity produces significant spatial control problems. BWRX-300 exhibits negative void reactivity and can be safely operated in a stable manner, as is the case for past reactors in the GE-H BWR fleet.

8.0 CONCLUSIONS AND RECOMMENDATIONS

The BWRX-300 design embodies engineered, passive and inherent safety characteristics that provide excellent safety. These include multiple means of reactor shutdown, inherently safe features (strongly neutronic feedback from coolant voiding and fuel temperature), and an excellent operating track record for BWRs. Advanced features in BWRX-300 include passively driven core flows to eliminate the need for core internal recirculation jet pumps, and significant simplification in the number of systems, components, pipes, and valves. The IPR team also notes that the Defence Line 2 (DL2) systems in BWRX-300 are more comprehensive than comparable safety features in the CANDU. In addition, the Isolation Condenser System (ICS) and Reactor Pressure Vessel (RPV) isolation provide a unique means for long term core cooling in the event of a Loss of Coolant Accident (LOCA) as compared to other reactor designs which require complex engineered Emergency Core Cooling (ECC) measures. Finally, the BWRX-300 passive containment cooling design is expected to improve the reliability of containment for many accidents. This IPR report highlights many of these positive features in Section 4.0.

The following points summarize the IPR team’s conclusions in areas of importance to shutdown means:

- REGDOC-2.5.2 Clause 8.4 requires two means of shutdown but does not presume that every candidate design will have reactivity characteristics that require two fully effective and independent Shutdown Systems. The BWRX design combines an inherently safe reactor design with appropriately redundant and diverse engineered means of shutdown that include a DL2, a DL3, a DL4a and a DL4b (BIS) functions.
- Except for the BIS, these shutdown means use the same 57 control rod blades. Given that BWRX-300 shares the same control rods for normal reactor power control and accident mitigation, BWRX-300 would have been non-compliant with past Canadian regulations, even though current Canadian requirements (e.g., Clauses 7.6.1 and 7.6.5 of REGDOC-2.5.2) now allow limited sharing of process/safety equipment between levels of defense and among safety systems. Sharing of these in-core components is explicitly allowed in REGDOC-2.5.2 Clause 8.4. A strong case can be made that the BWRX means of shutdown described in the PSAR may already be compliant with the relevant requirements in REGDOC-2.5.2. Table 2 in Section 3.5.1 of this report summarizes potential compliance gaps associated with this issue.

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- Both REGDOC-2.5.2 and REGDOC-2.4.1 require one or two diverse trip parameters for each shutdown means. The inherently safe BWRX-300 design must demonstrate that the trip coverage requirements and guidance shown in Table 1 and Table 2 of this report are met in the design or via a documented alternative approach to the REGDOCs utilizing Clause 11 of REGDOC-2.5.2 and/or Graded Approach in REGDOC-2.4.1.

The IPR team recommends the following regarding shutdown means:

- Declare DL4a as the second shutdown means for mitigation of DBAs – but without any design change or classification. DL4b BIS provides additional coverage for extremely low frequency BDBA scenarios but is not one of the two shutdown means credited to mitigate DBAs. Document justification that DL4a as is meets REGDOC-2.5.2 requirements for the non-fast-acting (second) shutdown means,
- Provide justification in Chapter 15 that BWRX-300 is an inherently safe reactor,
- Perform additional CN-DBA safety analysis, or reorganize the existing EX-DEC analysis in Chapter 15, crediting DL4a shutdown means only for mitigation of DBAs,
- The CN-DBA analysis in Chapter 15 should include a documented justification for the first trip parameter to be a direct trip parameter for both DL3 and DL4a, or otherwise is carried out in such a manner that the first trip signal for each of the two shutdown means is not credited to shutdown the reactor,
- Consideration should be given to credit only the second trip parameter for each shutdown means for events with positive pressure feedback that involve simultaneous positive reactivity insertion and impaired heat removal, and
- To address any potential residual licensing concerns related to independence and separation of the two shutdown means, perform and document reliability analysis to demonstrate that the failure to shutdown on demand is less than 1E-5.

Provided that the above gaps and recommendations are satisfactorily addressed (primarily through documentation updates and additional safety analysis), the IPR Team views that BWRX-300 means of shutdown not only meet but also exceed the REGDOC-2.5.2 and REGDOC-2.4.1 shutdown requirements and guidance for inherently safe reactors.

PSAR Rev. 0 provides no details on many of the complementary design features and safety-grade (DL4b) water injection which may be required for unforeseen events, or for inventory make-up for unisolable leaks or small loss of coolant accidents (DBA) in the longer term. Chapter 15.6 of the PSAR provides only a high-level description of many important functions (such as CRD purge water injection). It is not clear if the DL4b injection function is functionally and physically separated from the systems intended for other DL functions, defense lines or whether it uses diverse and flexible equipment and portable components.

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An important gap identified by the team relates to small pipe breaks with loss of preferred power and the need for emergency coolant makeup. The analysis in the PSAR concludes that fuel is adequately cooled for the first 3 days of the event. However, the analysis terminates with significantly reduced (collapsed) water level below the Top of Active Fuel in the downcomer. The analysis should be extended to summarize the steps needed to bring the reactor to an acceptable end state; restoration of water level in the RPV and leak isolation. For DEC sequences, this includes the PSAR discussing the potential for severe core damage occurring between 72 hours and 7 days (after which CRD makeup can be credited). The later and concluding stages of the accident progression need to be better described in the PSAR. In addition, for breaks within containment, if continuous water injection to the RPV is required, continuous water discharge through the break into containment may pose containment integrity challenges or equipment operability issues due to flooding in the containment. Furthermore, once the RPV and containment pressures equalize, there is a potential for non-condensable nitrogen gas to enter the RPV and impair the effectiveness of ICS heat removal capability.

FOAK design features of the BWRX-300, and hence concerns, include the use of Steel Bricks™, reactor pressure vessel (RPV) isolation valves, and ICS condensate discharge location into the chimney. The IPR Team concludes that additional analysis and qualification testing will be required to confirm these FOAK features as acceptable design approaches.

Although RPV isolation is identified here as an area of concern, the IPR team has also identified it as a design strength. Current Canadian nuclear plant designs include high-capacity ECCS pumps or accumulators to restore cooling to uncovered/uncooled fuel after postulated pipe breaks. The alternative approach in BWRX-300 is to start with a large water inventory in the RPV and to preserve adequate water in the core using RPV isolation valves that close following postulated failure of piping. It is intuitively preferable to not uncover fuel during postulated accidents by starting with more coolant and limiting its loss. This approach avoids the difficulty of rewetting hot surfaces after inventory loss and fuel heat up; it also reduces concerns associated with recovering, circulating outside containment for removing decay heat and re-injecting large volumes of water that would spill indefinitely from breaks that are not isolated (e.g., recovery strainer plugging issues).

For DEC, BDBA and severe accident sequences, BWRX-300 incorporates many, significant improvements for emergency cooling of the core as well as for containment emergency cooling and venting. With these improvements, plant states leading to significant release of radioactivity, have been practically eliminated, as is required by Clause 7.3.4 of REGDOC-2.5.2.

The IPR team also identified throughout this report numerous gaps, areas of further work and recommendations mostly in the safety analysis and documentation. These are documented in Section 5.0, Table 2, Table 3 and Appendix B, and include:

- Deterministic safety analyses in Chapter 15 of PSAR do not provide sufficient details for an FSAR,

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- Derived acceptance criteria for AOOs and DBAs need to be compliant with REGDOC-2.4.1 requirements and guidance;
- The analysis of the Normal Operating Conditions (NOC) is lacking, and Decay Ratio seems to be the only criterion used to evaluate stability without sufficient justification;
- Sufficient details and rationale are not provided for Event Identification & Classification;
 - Initiating event frequencies are not available to confirm event classification;
 - Initiating event examples in Appendix B of REGDOC-2.4.1 are to be considered, or justification for exclusion is required;
 - There is no AOO or DBA analysis for reactivity and power anomalies, e.g., due to inadvertent control rod movements;
- The justification of selection of bounding events for analysis is inadequate;
- Additional information needs to be provided in the PSAR on code accuracy and applicability to meet the regulatory requirements and guidance;
- The FOMs for each AOO and DBA should be clearly identified, and include code prediction uncertainty, as well as applicable operational uncertainties, for each FOM for DBAs;
- Code applicability needs to be justified for each category of initiating events individually rather than by AOO, DBA and DEC categories, and code accuracy should be quantified specific to each event category and each Figure-of-Merit (FOM);
- CN-DBA analyses need to be conducted to demonstrate that DL3 (and possibly DL4a if they are credited to mitigate DBAs for shutdown) safety functions alone are sufficient to mitigate DBAs, without the aid of any beneficial DL2 safety function; and
- The selection of conservative assumptions for each event, reactor power, exposure, safe operating envelope limits and acceptance criterion are not adequately defined and justified.

The above safety analysis and documentation gaps may have been addressed already in documents external to the PSAR. The IPR team recommends that they are addressed in a Pre-Operational Safety Analysis Report (POSAR) or Final Safety Analysis Report (FSAR) to demonstrate full compliance with the regulatory requirements and guidance for deterministic safety analysis.

Overall, the IPR team concludes that PSAR Rev. 0 provides sufficient detail to allow meaningful assessment of BWRX-300 safety features to support a license-to-construct application. Further, the team concludes that GE-H has successfully adapted established BWR design approaches to meet the new Canadian licensing requirements. The plant safety

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features described and analyzed in PSAR Rev. 0 demonstrate that the BWRX-300 design is safe.

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- [G-10] GE Hitachi Nuclear Energy, Licensing Topical Report, "BWRX-300 Containment Evaluation Method", NEDC-33922P-A, Revision 3, June 2022.
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Appendix A Accident Analysis Review

A summary of the IPR findings on the accident analysis in Chapter 15 is provided in Section 6.1 of this report. The following text provides a detailed accident-by-accident review of the Safety Analysis documented in Chapter 15 of the PSAR based on a pre-Rev 0 version of the Safety Report. Hence some of the comments below are no longer applicable to the issued Rev 0 of the PSAR. The comments are included here for completeness but may not be relevant due to changes made in the PSAR Rev 0 release.

A.1.0 TURBINE TRIP / LOAD REJECTION

CANDU power plants have a positive void coefficient of reactivity. As a result, power in current OPG plants can increase rapidly following postulated Loss-of-Coolant Accidents (LOCAs) due to coolant voiding, while adversely affecting the core cooling simultaneously. This reactor characteristic is the primary driver for having two shutdown systems in current CANDUs. In addition, CANDU accident analysis for LOCA events defines performance requirements for key reactivity control measures (e.g., shutdown system speed and reactivity depth).

The BWR behaviour is quite different in this regard due to the negative void coefficient of reactivity: increased coolant voiding introduces negative reactivity. For BWRs, the dominant reactivity phenomenon that leads to a potential power increase is void reduction rather than void formation. Since events such as turbine trip, load rejection, or spurious isolation of the RPV, can quickly collapse void in a BWR, and impairs fuel cooling simultaneously, these events feature prominently in BWR safety analysis. This leads to accident phenomenology that is dissimilar to those in CANDU analysis, but overall, a LBLOCA in CANDU is similar to a Turbine Trip in a BWR as they would result in a similar behaviour; a positive reactivity insertion and simultaneous impairment to fuel cooling.

The IPR team has reviewed BWRX-300 turbine trip safety analysis.

For the AOO TT case in [G-1], anticipatory scram (on load rejection or turbine trip) promptly terminates the event without any significant increase in reactor power. This is a result of the fast detection and action of the DL2 instrumentation and rod insertion and viewed positively by the IPR team.

For the DBA case presented in section 15.5.4.2.1, assumed impairment of DL2 power reduction systems, consistent with the REGDOC-2.4.1 requirement for DBAs not benefiting from DL2 functions, leads to a significant power excursion (~ 500%FP). For these cases, inherent reactivity feedback effects and DL3 reactivity control measures (e.g., DL3 scram prevent core damage. The safety report also includes a “Complex” turbine trip assessment (Section 15.5.5.2.2) where the DEC analysis proceeds similar to the AOO case, except half of the rods fail to scram and run-in fails to move the uninserted rods. In the event of this complex sequence the reactor reaches a new steady state condition, (Figures 15.5-130 and 15.5-133) and operators will either take action to insert the remaining rods, or will manually

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inject boron. The PSAR also mentions an option of the operators reducing feedwater flow, which reduces water level which reduces core flow, which decreases power by negative void reactivity until the steam flow matches the feedwater flow, which is not analyzed and implies its power decrease is limited by avoidance of RPV isolation and ICS initiation on both low level L2 and sustained low feedwater flow.

Although the BWRX-300 turbine trip analysis is new for Canada, it is neither new nor unique for BWR plants in general. Based on Reference [M-1], the power pulse seen in the BWRX-300 analysis are fairly typical for BWR plants (which are numerous, and which have been operating safely for many years). Another important point is that BWR analysis results for turbine trip events have been well validated by the tests and benchmarking activities described in Reference [M-1], albeit without direct evaluation of any potential differences caused by the natural circulation flows inside the BWRX, and for the chosen ICS discharge location that can affect both the neutronic response of the core and fuel cooling by impairing core flow driven by buoyancy forces of the DBA case.

The IPR team also recognizes that there is also a large difference in the frequency of Turbine Trip or Load Rejection events and LBLOCA events. It is not practical to carry out an integrated LBLOCA test nor is there OPEX, while Turbine Trip events are fairly frequent and likely have OPEX data related to system performance. This would make OPEX from existing BWRs quite useful to increase confidence in the prediction of consequences of a Turbine Trip. The OPEX can be used either directly in an assessment or through code validation, especially if there are events in the OPEX where the Turbine Bypass Valve failed to open during a Turbine Trip or Load Rejection. The IPR team searched for such OPEX through open sources, but could not find any that is directly relevant other than the limited tests available from the Peach Bottom-2 reactor. GE-H should consider referring to OPEX to improve confidence in system performance for the AOO case.

Based on the above, the IPR team suggests a number of changes to improve the clarity of the turbine trip safety analysis presented in Reference [G-1]. Suggested improvements to the analysis include the following:

1. The analysis should clearly specify trip/scram delays, requirements on rod insertion timing, and the conservatisms/uncertainties included in the assumed system behaviour (I.e., it is unclear if the analytical limits information in Tables 15.5-3 to 15.5-5, for example, accounts for all instrument uncertainties and timing delays). The analysis needs to demonstrate that inherent reactivity effects and backup reactivity control features safely limit power for credible variations of response time.
2. The analysis should clearly specify the allowance/magnitude made for void reactivity uncertainty, initial void fraction in the core (and bias applied for conservatism) and associated prediction uncertainty and reference the experiments and or reports used to derive the void reactivity, void fraction, and error allowances. While Section 15.5.4.2.1 provides a high-level summary, detailed numbers should be provided in a table.

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3. The analysis should confirm that results are valid for reduced initial reactor power and regardless of core irradiation (i.e., BOC, MOC, EOR). Ideally results for each exposure, or a table summary, should be provided for comparison.
4. The time scales of analysis summary graphs should be adjusted to better capture rapid changes at the start of transient cases. E g., both the short term (~5s) and longer term (500s) graphs should be provided in Figures 15.5-52 and 15.5-57 since the power and fuel clad temperature excursions and other responses that immediately follow the IE cannot be read on the time scale provided in Chapter 15).
5. The transient plots of the CPR, centreline temperature and cladding strain also are needed for the DBA case to confirm their relative margins.

A.2.0 LOSS OF COOLANT ACCIDENT (LOCA)

As noted previously, the large break LOCA analysis for BWRs differs significantly from CANDU analyses, with the primary difference being that the voiding during a large break LOCA tends to reduce power in BWR. Therefore, the requirements in BWRs tend to be on core refill (for traditional BWRs), core isolation (in BWRX-300) and phenomena related to containment and dose. These are some of the bounding initiating events to show the effectiveness of DL3 [Sections 15.2.4.6, 15.5.4.6 and 15.5.9.2] and DL4a [Section 15.2.4.6] to cope with:

1. uncovering of the core and dose (unisolable small breaks [15.5.9.2.5]),
2. the increases in containment temperature and pressure (unisolable small breaks inside containment [Section 15.5.4.6.4] and large breaks inside containment [Section 15.5.4.6.1, 15.5.4.6.2 and 15.5.4.6.3]) and
3. RPV isolation of large breaks outside containment and dose [Section 15.5.9.2.1, 15.5.9.2.2 and 15.5.9.2.3].

All these events are analysed with conservative assumptions as DBA, regardless of frequencies of occurrence. Reference [G-10] also includes baseline and uncertainty analyses. Section 15.2.4.7 of the PSAR provides a break-specific summary of the DL3 and DL4a initiation parameters for scram/control rod drive and ICS as well as for isolation of pipe breaks and containment.

Review comments on the containment function during LOCAs are provided in Section 6.2.1 of this report.

A.2.1 DBA Large Main Steam Line Break Inside Containment

While at full power, DL3 functions scram the reactor on high containment pressure (1 s), open one train of ICS return valves (1 s) on high containment pressure and isolate the RPV within 10 s on high containment pressure following a DBA consisting of a large main steam line break inside containment, Common Cause Failure (CCF) of DL2 and failure of one ICS train

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(to account for the SFC) with a conservative assumption of consequential loss of preferred power (feedwater pump trip, Turbine Stop Valve (TSV) closure and containment isolation valves open to discharge steam from both ends of the break). The assumptions used in the Loss of Preferred Power (LOPP) case is intended to bound the case of preferred power available. Operator action replenishes the ICS pool after 3 days. The peak cladding temperature margin remains large throughout the transient. The peak containment pressure (419 kPa) and temperature (133°C) margins are judged in the analysis to be medium and hence initial condition and modelling parameters are biased conservatively to demonstrate the margin to the acceptance criteria.

Review comments for this event are as follows:

1. A figure (preferable), or reference, with the configuration of the pipes and valves is needed within this section to clearly indicate the flow paths and break locations.
2. Confirm whether the trip parameter instrumentation uncertainties and timings are based on an ISA 67.04 instrument uncertainty standard, or alternate.

A.2.2 DBA Large Feedwater Pipe Break Inside Containment

While at full power, DL3 systems scram the reactor on high containment pressure (1 s), open two trains of ICS return valves (1 s) on high containment pressure and isolate the RPV within 10 s on high containment pressure following a DBA consisting of a large feedwater line break inside containment, CCF of DL2 and failure of one ICS train (to account for the SFC). Operator action replenishes the ICS pool after a week. The peak cladding and containment temperature margins remain large throughout the transient.

Review comments for this event are as follows:

1. A summary table of the important parameters and the conservatisms applied in the analysis from a system or containment standpoint would be beneficial.
2. The rationale to exclude a DEC case for this event should be provided or a DEC analysis of this event added.

A.2.3 DBA Small Break

While at full power, DL3 systems scram the reactor on low RPV pressure (12 s), isolate the RPV and open two trains of ICS return valves on low RPV level (60-65 s) following a DBA consisting of a small LOCA, CCF of DL2 and failure of one ICS train to account for the SFC (i.e. 2 ICS trains credited) with a conservative assumption of consequential loss of preferred power (feedwater pump trip, TCV as is or TBV open).

Review comments for this event are as follows:

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1. The transient CPR, centreline temperature and cladding strain are also needed for the DBA case to confirm their relative margins.
2. The description of the small break sizes and locations (instrument lines vs. ICS small breaks) that can lead to degradation of ICS performance if such a situation can arise is spread out and hard to follow.
3. The rationale to exclude a DEC case for this event should be provided or a DEC analysis of this event added.

A.2.4 DBA Breaks Outside Containment

The thermal hydraulics for breaks outside containment are bounded by the conservative assumptions of breaks inside containment and changes in actuation parameters from high containment pressure to pipe break indications. The dose margins are large.

Review comments for this event are as follows:

1. Either provide explicit documentation on an alternative regulatory approach or confirm whether there is an effective backup/second DL3 scram parameter since regulatory documents may require such an assessment.
2. Add descriptions of the flow limiters, their exact location, and performance specifications or explicitly cross reference the figures elsewhere in the safety report.
3. Provide a detailed description of the DL3 and DL4a ICS leakage detection systems. Confirm what instruments are used and their uncertainties, and which systems are used to activate the DL3 and/or DL4a shutdown.
4. Identify the valves used to isolate ICS breaks. Explain how the DL2 and DL3 valves on the ICS return line are isolated given they are operated separately by DL3 and DL2.
5. The rationale to exclude a DEC case for this event should be provided or a DEC analysis of this event added.

A.3.0 LOSS OF CONDENSER VACUUM

The IPR team has reviewed BWRX-300 loss of condenser vacuum safety analysis presented in Sections 15.5.3.2.3 and 15.5.5.2.3 of the PSAR. As discussed in Section 15.2.4.2 of the PSAR, turbine trip and loss of condenser vacuum produce similar plant responses. Therefore, the discussion in Section A.1.0 above applies to the loss of condenser vacuum events as well.

A.4.0 LOSS OF PREFERRED POWER

This is one of the bounding initiating events to show the effectiveness of DL2 [Section 15.5.3.2.4], DL3 [Section 15.5.4.2.2] or DL4a [Section 15.5.5.6] to cope with the positive

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reactivity from an increase in RPV pressure resulting from a reduction in flow. The loss of preferred power is similar to Loss of Offsite Power (LOOP) events and hence scrutiny of these accident sequences will be high given the lessons learned from the Fukushima accident. The BWRX-300 design is well suited to handle such events given the availability of a passive ICS system with both the capability for throttling (via DL2) to ensure a controlled cooldown during an AOO, and a fully open valve (via DL3) for more severe events.

A.4.1 AOO

While at BOC full power, a DL2 action scrams the reactor (control rod fraction increases to 0.90) on generator load rejection, opens the Turbine Bypass Valve (TBV) on high RPV pressure and closes the TBV and opens an ICS train return valve on high main condenser pressure to decrease the RPV pressure and remove decay heat following an AOO loss of preferred power. Operator action replenishes the ICS pool in a timely fashion. .

Review comments for this event are as follows:

1. Provide some discussion on the timing of the low bus voltage signal, any OPEX or testing information, and its robustness/reliability for any total or partial loss of preferred power or reference the design documentation.

A.4.2 DBA

While at End of Rated (EOR) full power, DL3 scrams the reactor (control rod fraction increases from 0 to 0.9) on high neutron flux, performs RPV isolation and opens the ICS train B return valve on high RPV pressure to decrease the RPV pressure and remove decay heat following a DBA loss of preferred power (assuming a CCF of DL2, a single failure of ICS train A). Operator action replenishes the ICS pool in a timely fashion. Core oscillations decay and the peak cladding temperature margin remains large throughout the transient.

Review comments for this event are as follows:

1. Outline the conservative assumptions applied to operating conditions, trip parameters/timing, and analysis models and provide the timing of events in summary tables for BOC/MOC/EOR if possible, since the timings will be a function of the exposure. Confirm that the calculation of such values is consistent with ISA 67.04 (or equivalent) where appropriate.
2. The second ICS is credited to actuate but it is not clear whether it is fully open (or throttled). Confirm that the condensate return valve is fully open as per the DL3 action.
3. The figure resolution (i.e., x-axis scaling) needs improvement to show the peak reactor power at 153% full power and the flow oscillations. Consider plotting both the short-term and long-term responses separately so that the details of the event are visible.
4. Provide a description in the section on why there are two pressure and Peak Cladding Temperature (PCT) peaks.

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5. The transient CPR, centreline temperature and cladding strain also are needed for the DBA case to confirm their relative margins.
6. Explain the cause of the sustained low core inlet flow and reversals if they occur.

A.4.3 DEC

While at EOR full power, DL4a actions close the TCVs and open the TBV on generator load rejection signal (0.25 s), run in the control rods (control rod fraction increases from 0 to 0.9 in 70 s) and open an ICS train A return valve on high flux (5 s) and close the TBV on loss of preferred power (6 s) to decrease the RPV pressure and remove decay heat following a DEC loss of preferred power (assuming CCF of DL3). Operator action replenishes the ICS. Core flow oscillations decay once the reactor is shut down and the peak cladding temperature margin remains large throughout the transient.

Review comments for this event are as follows:

1. The figure needs better resolution (i.e., x-axis scaling) to show the peak reactor power at 250% full power and the flow oscillations.
2. Describe why there are two pressure and PCT peaks.
3. The transient CPR, centreline temperature and cladding strain also are needed for the DEC case to confirm their relative margins.
4. Explain the cause of the core flow reversals, if any.
5. Given the potential for rapid and frequent oscillations provide a reference to the fuel qualification documentation supporting integrity for this case.

A.5.0 LOSS OF FEEDWATER HEATING

This is the bounding initiating event to show the effectiveness of DL2 [Section 15.5.3.1.1] or DL3 [Section 15.5.4.1.1] to cope with the positive reactivity from a reduction in coolant temperature.

A.5.1 AOO

While at MOC full power, direct low feedwater temperature DL2 functions bring the reactor to steady operation near 80% full power following an AOO loss of one feedwater heater. The CPR margin remains large throughout the transient. No significant safety concerns arise and the reactor returns to a steady-state.

IPR recommends additional analysis at a feedwater temperature decrease just less than the SCRR I DL2-27 setpoint ($<16.6^{\circ}\text{C}$), to confirm that the plant DL1 or the remaining DL2 functions can compensate such a blinding temperature decrease for this AOO.

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A.5.2 DBA

While at MOC full power, DL3 functions scram the reactor (control fraction increases from 0.05 to 0.90) on high simulated thermal power (97 s), isolate the main steam on low RPV pressure (111 s), isolate the feedwater on high RPV level (302 s) and open an ICS train B return valve (>400 s) on high RPV pressure following a DBA consisting of loss of all feedwater heating (CCF of DL2 and SFC of ICS train A). Operator action replenishes the ICS pool after a week. The peak cladding temperature margin remains large throughout the transient.

Review comments for this event are as follows:

1. The transient CPR, centreline temperature and cladding strain also are needed for the DBA case to confirm their relative margins. Showing only sheath temperatures for cases without dryout doesn't add much to the presentation since in such a case there will always be only small changes and high margins on sheath temperature. However, if the CPR is shown to encroach on dryout condition, then one could speculate that for small changes in the system or uncertainties dryout could occur and hence a temperature excursion could take place. Hence providing the CPR trends provides an assurance as to the actual margin to fuel sheath temperature increases.
2. Identification of the conservative assumptions used in the DBA analysis is needed.
3. Extend the analysis to show the increase in RPV pressure following RPV isolation as well as the effectiveness of ICS.
4. The rationale to exclude a DEC case for this event should be provided or a DEC analysis of this event added.

A.6.0 FEEDWATER FLOW INCREASE

This is one of the bounding initiating events to show the effectiveness of DL3 [Section 15.5.4.4.1] to cope with an increase in RPV coolant inventory added to the downcomer.

A.6.1 DBA

While at MOC full power, DL3 functions isolate the feedwater on high RPV level (25 s), scram the reactor (control rod fraction increases from 0.05 to 0.90) on high simulated thermal power (30 s), isolate the main steam on low RPV pressure (90 s), and open an ICS train B return valve (>400 s) on high RPV pressure following a DBA consisting of both feedwater pumps increase to maximum flow, CCF of DL2 and SFC of ICS train A. Operator action replenishes the ICS after 3 days. The peak cladding temperature margin remains large throughout the transient.

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Review comments for this event are as follows:

1. Identification of the conservative modelling assumptions used in the DBA analysis is needed especially for neutronic trips initiated by void collapse and the rod insertion characteristics.
2. Add additional plots with scales showing short term details since the plot scaling does not show the first 5s where the power excursion takes place.
3. The transient CPR, centreline temperature and cladding strain also are needed to confirm their relative margins.
4. The rationale to exclude a DEC case for this event should be provided or a DEC analysis of this event added.

A.7.0 INADVERTENT ISOLATION CONDENSER SYSTEM (ICS) INITIATION

This is one of the bounding initiating events to show the effectiveness of DL2 [Section 15.5.3.4.1] or DL3 [Section 15.5.4.4.2] to cope with an increase in RPV coolant inventory added to the RPV chimney above the reactor core.

A.7.1 AOO

While at EOR full power, the DL2 functions Reactor Pressure Control (RPC) and Reactor Level Control (RLC) bring the reactor to steady operation near full power with reduced steam and feedwater flows and a reduced feedwater temperature following an AOO inadvertent isolation condenser initiation – one train. The CPR margin remains large throughout the transient. Power oscillations remain small and eventually dissipate.

Review comments for this event are as follows:

1. The discussion should be enhanced to include the sensitivity of power oscillations to reactor power and or trip timing should be discussed for the AOO one ICS train case. Describe the cause of these oscillations. Specify whether these are loop wide oscillations (core, chimney, and downcomer).
2. The core inlet flow initially increases in the predicted results. This should be explained.
3. This AOO one ICS train case should be reanalysed as a DBA without credit for DL2.

A.7.2 DBA

While at MOC full power, inadvertent isolation condenser initiation – all trains, and CCF of DL2, would cause a reduction in core flow and thus an increase in core void – thereby lowering reactor power. DL3 functions scram the reactor (control rod fraction increases to 0.90) and isolate the main steam on low RPV pressure (50 s) and isolate the feedwater on high RPV level (70 s). Operator action replenishes the ICS in a timely manner. The peak

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cladding temperature margin remains large throughout the transient despite three instances of core inlet flow reversal.

Review comments for this event are as follows:

1. The transient CPR, centreline temperature and cladding strain also are needed to confirm their relative margins.
2. Explain in the report the sudden peak in steam flow near 40 s and why the downcomer liquid level and core flow peak at 120 s.
3. The capability of the code for simulating the large liquid ingress into the chimney and the complex mixing behaviour needs to be demonstrated. The validation basis for this event and the evidence should be summarized or referenced here, including:
 - a. Identify whether there are 3D changes in the flow distribution in the chimney that could cause the CPR in some channels to be reduced lower than that in the average assembly.
 - b. Explain why the core inlet flow initially decreases unlike the increase in the AOO.
 - c. The injected flow into the chimney persists for 50s until the low pressure setpoint is reached. Thus, prediction of the phenomena becomes increasingly important given the duration of the event.
 - d. Identify any near or full-scale experimental data to support the code predictions in this complex situation.

There are no significant concerns if the margin to CPR is adequately large, and the code predictions are conservative with respect to the complex 3D phenomena. However, demonstration of this facet has not been provided.

A.8.0 RPV PRESSURE CONTROL DOWNSCALE

This is one of the bounding initiating events to show the effectiveness of DL3 [Section 15.5.4.2.3] to cope with the positive pressure reactivity from isolation of the steam lines.

A.8.1 DBA

While at EOR full power, DL3 functions scram the reactor (control rod fraction increases from 0 to 0.9) on high neutron flux (0.8 s), isolate the feedwater on high RPV level (167 s) and open an ICS train B return valve (155 s) on high RPV pressure following a DBA closure of TCV and TBV, CCF of DL2 and SFC of ICS train A. Operator action replenishes the ICS tank water after an appropriate period of time. The RPV overpressure and peak cladding temperature margins remain large throughout the transient.

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Review comments for this event are as follows:

1. Outline the conservative assumptions applied and provide the timing of events in summary tables for BOC/MOC/EOR if possible.
2. Outline the conservative assumptions applied to the trip timing calculations since the trip registers very fast (within the first 0.75 s). Confirm calculations are consistent with ISA 67.04 (or alternate) where appropriate.
3. The transient CPR, centreline temperature and cladding strain also are needed for the DBA case to confirm their relative margins.
4. Analysis of the margins for overpressure, fuel centreline melting, and CPR are needed for a backup/second DL3 scram parameter.
5. The second ICS is credited to actuate but it is not clear whether it is fully open (or throttled). Confirm that the return valve is fully open.
6. The figure resolution needs improvement to show the peak reactor power at 154% full power and RPV dome pressure at 8.17 MPa.
7. Explain why there are two pressure and PCT peaks.
8. Explain why the core flow increases and peaks at 140 s.
9. Extend the analysis to show the increase in RPV pressure following RPV isolation as well as the effectiveness of ICS.
10. The rationale to exclude a DEC case for this event should be provided or a DEC analysis of this event added.

A.9.0 INADVERTENT RPV ISOLATION

This is one of the bounding initiating events to show the effectiveness of DL2 [Section 15.5.3.2.2] or DL3 [Section 15.5.4.2.4] or DL4a [Section 15.5.5.2] to cope with the positive pressure reactivity from isolation of the steam lines.

A.9.1 AOO

While at EOR full power, DL2 functions scram the reactor (control rod fraction increases from 0 to 0.9) on main steam isolation valve position closed (0.5 s), conservatively close the other main steam isolation valve on high steam line flow (3 s) and open an ICS train return valve on high RPV pressure (not modelled) to decrease the RPV pressure and remove decay heat following an AOO closure of one main steam line isolation valve. The RPV overpressure and CPR margins remain large throughout the transient. Operator action replenishes the ICS after a week.

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Review comments for this event are as follows:

1. Table 15.5-4 should include best estimate rod insertion timing for AOO cases as well as conservative timing for DBA cases.
2. Explain in the text the short-term negative void reactivity vs. void fraction and the CPR transient behaviour.

A.9.2 DBA

While at EOR full power, DL3 functions scram the reactor (control rod fraction increases from 0 to 0.9) on high neutron flux (1.09 s) and open the ICS train B return valve on high RPV pressure (122 s) following a DBA CCF closure of all main steam and feedwater RPV isolation valves, CCF of DL2 and SFC of ICS train A. Operator action replenishes the ICS after a week. The RPV overpressure (8.73 MPa) and peak cladding temperature margins remain large throughout the transient.

Review comments for this event are as follows:

1. Outline the conservative assumptions applied and provide the timing of events in summary tables for BOC/MOC/EOR if possible.
2. Outline the conservative assumptions applied to the trip timing calculations since the trip registers very fast (within the first 1.06 s). Confirm the assumptions conform to ISA 67.04 or equivalent.
3. The transient CPR, centreline temperature and cladding strain also are needed for the DBA case to confirm their relative margins.
4. The second ICS is credited to actuate but it is not clear whether it is fully open (or throttled). Confirm that the return valve is fully open.
5. The figure resolution needs improvement to show the peak reactor power at 158% full power and RPV dome pressure at 8.61 MPa.
6. Explain why there are two pressure and PCT peaks.

A.9.3 DEC

While at EOR full power, DL4a functions run in the control rods (control rod fraction increases from 0 to 0.9 in 70 s), open all ICS trains' return valves and trip the FW pumps on the main steam isolation valve position closed (1 s) following a DEC CCF closure of all main steam isolation valves and CCF of DL3. Operator action replenishes the ICS when needed. Core flow oscillations decay once the reactor is shut down and the peak cladding temperature margin remains large throughout the transient.

Review comments for this event are as follows:

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1. Provide the uncertainty of the increase in RPV pressurization rate to 11.60 MPa coinciding with a large reduction in ICS flow at 30 s and completion of the control rod run in at 60 s.
2. The figure resolution needs improvement to show the peak reactor power at 150% full power and the flow oscillations.
3. The transient CPR, centreline temperature and cladding strain also are needed for the DEC case to confirm their relative margins.
4. Explain why void collapse takes 60 s.
5. Explain the increase in PCT at 60 s vs. the decrease in average fuel temperature.
6. Explain the cause of the core flow reversal at about 90 s.

A.10.0 LOSS OF FEEDWATER FLOW

This is the bounding AOO initiating event to show the effectiveness of DL2 [Section 15.5.3.3.1], DL3 [Section 15.5.4.5.1] or DL4a [Section 15.5.5.8] to cope with a decrease in coolant inventory.

A.10.1 AOO

While at EOR full power, DL2 functions start the standby feedwater pump (10 s) and reactivity effects bring the reactor to steady operation near 80% full power following an AOO loss of one feedwater pump. Reactor power oscillations are small, and the CPR margin remains large throughout the transient.

A.10.2 DBA

While at MOC full power, the reactor level and core flow are reduced generating more void, which lowers reactor power following a DBA consisting of a CCF closure of all RPV feedwater isolation valves and CCF of DL2. DL3 functions scram the reactor (control rod fraction increases from 0.05 to 0.90) on low RPV level (L3, 25 s), isolate the main steam on low RPV pressure (85 s) and initiate ICS on low RPV level (L2, 120 s). Operator action replenishes the ICS when needed. The peak cladding temperature margin remains large throughout the transient.

Review comments for this event are as follows:

1. The transient CPR, centreline temperature and cladding strain also are needed to confirm their relative margins.
2. Identify the conservative assumptions or uncertainties in CPR as well as conservatisms in operating conditions assumed.

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3. Assessment or analysis of the impact of a single-failure in the ICS system is needed, although it is unlikely that such an SFC will cause any appreciable impact.
4. Explain the cause of the core flow reversal.

A.10.3 DEC

While at MOC full power, the reactor level and core flow are reduced generating more void, which lowers reactor power following a DEC consisting of a CCF closure of all RPV feedwater isolation valves and CCF of DL3. DL4a functions perform rod run-in and isolate the main steam on sustained low feedwater flow (65 s) and initiate ICS on high RPV pressure. Operator action replenishes the ICS after a week. The peak cladding temperature margin remains large throughout the transient.

Review comments for this event are as follows:

1. The transient CPR, centreline temperature and cladding strain also are needed to confirm their relative margins.
2. Explain the cause of the core flow reversal.
3. Explain the void reactivity minimum and core average void fraction maximum at 140 s.

A.11.0 RPV PRESSURE CONTROL OPEN

This is the bounding non-LOCA DBA initiating event to show the effectiveness of DL3 [Section 15.5.4.5.2] to cope with a decrease in coolant inventory.

A.11.1 DBA

While at MOC full power, the RPV pressure and power decrease accompanied by the increase of the core void fraction and flow following a DBA opening of the TCVs and TBVs, CCF of DL2 and SFC of ICS train A. DL3 functions isolate the feedwater on high RPV level (225 s), scram the reactor (control rod fraction increases from 0.05 to 0.90) and isolate the main steam lines on low RPV pressure (70 s) and open an ICS train B return valve (>500 s) on high RPV pressure. Operator action replenishes the ICS after a week. The peak RPV pressure and cladding temperature margins remain large throughout the transient but are still increasing at the end of the simulation (500 s).

Review comments for this event are as follows:

1. Outline the conservative assumptions applied and provide the timing of events in summary tables for BOC/MOC/EOR if possible.
2. The transient CPR also is needed for the DBA case to confirm the relative margin.

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3. Explain what would happen if the RLC continues to function normally rather than freezing as is.
4. Explain the reason for the reversals and sustained low value of core flow after 100 s.
5. Extend the analysis to show the increase in PCT and RPV pressure following RPV isolation as well as the effectiveness of ICS beyond 500s.
6. The rationale to exclude a DEC case for this event should be provided or a DEC analysis of this event added.

A.12.0 OTHER ACCIDENTS

Appendix B also includes comments on the remaining bounding events:

1. All Control Rod Withdrawal [Section 15.5.7.1] – This is a proposed reactivity event wherein due to a CCF in the control system all rods are removed from the core and the rod blocking algorithms are assumed to not function. The reactor Scrams on DL3 simulated thermal power. After the trip, the systems isolate the RPV and initiate ICS response.
2. Inadvertent Control Rod Withdrawal [Section 15.5.5.7.2] – analysis is performed for a single Control Rod (CR) removed from the core with a failure of DL2 systems and rod-blocking. Some fuel elements enter the boiling transition. While the PCT temperatures are within the DBA limits, local fuel failures may occur but are limited in location and number.
3. Fuel Loading Errors [Section 15.5.4.3] – Fuel loading errors have a low probability of occurrence and may be detected through procedural checks or during reactor start-up and noticeable core power shape anomalies. If undetected, multiple misplaced fuel assemblies may exceed boiling transition in the conservative analysis and some fission products may be released from the fuel. In this event, operators will become aware of the failed fuel and locate the affected assemblies. Analysis shows that the dose acceptance criteria should be met even in the bounding case.

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Appendix B Detailed IPR Team Comments

This Appendix provides supplementary comments to those in the main body of this report. In some instances, these comments elaborate on those already made in the main body of the report and cross-references to the appropriate main body section have been identified. The comment numbering is not sequential, as some comments made during the review of PSAR Rev. A were subsequently removed once the information in PSAR Rev. 0 was reviewed by the IPR team (see Section 2.3 for details).

Item #	PSAR Rev. 0 Reference	Excerpt from PSAR Rev. 0	IPR Comment on PSAR Rev. 0	Related Main Report Section
1	1.7.13, Table 6.3.2-1, 9A.4.2.1.1, 9A.4.2.6	The system also maintains a slightly positive pressure in containment to prevent air (oxygen) in-leakage into the inerted spaces from the Reactor Building.	REGDOC-2.5.2 Section 8.6.1 Additional information lists CSA N290.3 which requires the following: "13.1 The containment atmosphere shall be maintained at subatmospheric pressure in normal operation to limit uncontrolled leakages to the environment." BWRX-300 containment is kept slightly above atmospheric pressure. Recommend adding a description of how the RB completely encloses the primary containment system and controls the release to the environment during normal operation to confirm the following understanding. The BWRX RB envelopes the primary containment (1.7.6) and the RB HVAC (9A.5.1.1.1 9A.5.1.1.4 and 9A.5.1.3) maintains a negative pressure (except the inner ICS pool, SCA and battery rooms) and vents to the continuous exhaust air plenum/plant vent stack. Therefore, the releases from the primary containment are filtered (9A.5.1.1.4 and 9A.5.1.3) and monitored (20.8.7) by the Process Radiation Monitoring System (11.5.1.2).	Standalone comment
2	1.8, 7.3 (Tables 7.3-1, -3, -4), 9A.1.2.3.5, 15.6.1.3.6	Intra-system dependencies and intersystem dependencies including functional, human, phenomenological, and CCFs that influence system unavailability or the system contribution to accident sequence frequencies are identified.	Clarify the validation included in each defence level, such as alarms and rod blocks, which prevent common failures arising from the mode switch being connected to multiple defence levels and whether the mode switch has a CCF on all defence lines.	6.4
3	3, 4 and 6		REGDOC-2.6.1 (Reliability Programs) and REGDOC-2.5.2, e.g., Clause 5.6, have Reliability requirements for DL-3 safety functions, e.g., "The safety systems and their support systems shall be designed to ensure that the probability of a safety system failure on demand from all causes is lower than 1.0E-3". It is not clear where such requirements are addressed for all DL3 functions in Chapter 3, 4 or 6. Clarify that Section 7.3.1.3.2 that claims high reliability of 1E-4 failure on demand covers only the instrumentation for C10 and that reliability analysis will confirm the probability of the entire safety system (I&C for actuation and end devices for execution) failure on demand from all causes is lower than 1E-3.	6.4
5	Table 3.2-1, 7.1.2, 7.2.1, 7.3.2, 7.3.3, Tables 7.1-1, 7.3-6 and -7	The DL4a (i.e., Safety Category 2) and DL2 (i.e., Safety Category 3) DCIS functions are implemented in different subsystems of C20. The DL4a DCIS functions are implemented with SC2 equipment. The majority of the C20 SC2 equipment is located in its own fire barrier room in the Control Building with the remainder located in compartmentalized fire barrier rooms in the Reactor Building. The DL2 functions are implemented with SC3 equipment. The C20 SC3 equipment is located in two separate fire barrier rooms in the Control Building.	Identify any CCF in system C20 between the functions shown as green and blue (Figure 7.2-1) and confirm PSA accounts for CCF. Explain how signals sent/shared from DL4a to DL2 are not credited with mitigation for the same postulated initiating event. Also explain the rationale for 1MSRIVC and LOCV events being analyzed both as AOO and DEC, but not as DBA.	6.4

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6	7.3.3.4, Table 7.3-2, 15.6, 15B	A preliminary list of the Type D, E, and F accident monitoring variable is provided in Table 7.3-2.	Add descriptions of all the DL4b equipment. Clarify whether Type D, E and F accident monitoring variables are DL4b.	6.2.3.3
7	3.3 (e.g., 3.3.1.1.2)	Instead of using UHRS representing the mean estimate of the seismic hazard as mandated by ASCE/SEI 43, the bounding FIRS, PBSRS and PBIRS are conservatively developed using the 1E-4 and 1E- 5 MAPE UHRS representing UB estimates of the seismic hazard. These UB UHRS are developed as described in Subsection 3.3.1.1.1 to account for the epistemic uncertainties related to the site inputs and simplified SRA methodology. The resulting horizontal Ground Motion Response Spectra (GMRS) are further adjusted to meet the minimum required response spectra requirement using the generic spectrum in CSA N289.3, Clause 4.3.2 anchored at the minimum peak ground acceleration value of 0.1g.	REGDOC-2.4.1 event classification scheme for IEs is frequency based and does not distinguish internal events (Chapter 15) from External Hazards (Chapter 3.3). External events are hence characterized as DBA's if their frequencies fall between 1E-5/y and 1E-2/y. This event classification scheme is not always consistent with some existing CSA Standards, e.g., CSA Standard for Seismic, which defines Design Basis Seismic Event with a frequency of 1E-4/y for new plants. REGDOC-2.5.2 refers to both REGDOC-2.4.1 and to Canadian National Standards or Equivalent (Clause 5.13) for event classification. It is not clear from reading PSAR how external events are classified and whether they are classified in strict accordance with REGDOC-2.4.1 or Clause 7.13 of REGDOC-2.5.2. For DBA seismic events (DBE), as per ASCE/SEI Standard 43-05 and NRC RG 1.208, the target value of 1E-05 is used for the mean annual probability of exceedance (frequency 1E-05/year) of the onset of significant inelastic deformation by scaling the site specific mean 1E-04/year uniform hazard response spectra by a design factor. Clarify that this is consistent with the REGDOC-2.4.1 classification of DBA for common cause events having frequencies down to 1E-5/year. This comment is not limited to seismic events only, but applicable to all external hazards. In Chapter 3 or Chapter 15, no DSA or dose calculations are provided for any of the external event DBAs assuming equipment not qualified fails as a consequence of the DBA.	Table 3 , 5.4
8	3.4.2	SC1 SSC and SC2/SC3 SSC credited with flood event mitigation in the fault evaluation are protected against internal flooding.	There is insufficient information to review flood protection. Flood assumptions are in the external PSA Summary report. It is understood that a flood hazard assessment is in progress and no DSA is anticipated.	5.4
9	7.1.1, 15.2, 15.5.5.2.2, Tables 15.6-7 & 15.7.9-1	DL4a is used to mitigate PIEs that are not mitigated by DL2 (...) and provides a second credited defense line for such PIEs (after DL3). DL4a is designed to work in tandem with DL2 to ensure all AOOs and DBAs resulting from a single failure are mitigated by two defense lines among DL2, DL3, and DL4a. Accordingly, DL4a is not required to be independent and diverse from DL2. DL4a can be used, along with unaffected DL2 functions, to mitigate a PIE as part of the same event sequence (...). All AOOs and DBAs resulting from a single failure are required to be mitigated by DL3 and separately by DL2, DL4a, or a combination of DL2 and DL4a. DL2 and DL4a are not credited to mitigate the same PIEs independently of each other and are therefore not required to be independent from each other. The postulated initiating event is the same as for the LR-TT AOO event. Additionally, the event assumes that half of the control rods with the highest rod worth fail to scram and the CRDM run-in fails to insert the rods that failed to scram. No other failures are assumed. Medium LOCA – Isolable (MLOCA-I): 3.1E-05/yr	The rationale for Bounding DEC Without Core Damage Event Selection in 15.2 lacks detail for level 1 PSA complex sequences. For examples, half the rods are mechanically stuck for the complex sequence load rejection or turbine trip, and there are only AOO and DBA and no DEC for loss of feedwater heating while there are AOO, DBA and DEC for Loss of Preferred Power. The PSA claims no severe core damage for non-isolable MLOCA-N and LLOCA-N as well as RPV ruptures above the TAF; these LOCA events are not identified as DEC and their analysis and long term RPV water makeup are not analyzed with the isolable LLOCA-I in Chapter 15. <div></div>	5.4 (5.7, A.1.0, A.5.2, 6.2.2), 6.4

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		Large LOCA – Isolable (LLOCA-I) 3.6E-06/yr Medium LOCA – Non-isolable (MLOCA-N) 1.6E-06/yr Large LOCA – Non-isolable (LLOCA-N) 6.1E-7/yr At Power CDF 1.1E-08/yr		
11	7.3, Tables 15.5-5, 15.5-49, 15.5-50		These tables identify the DL2/3/4 functions at a high level. The IEC 61888 setpoints associated with the DL functions, and associated uncertainty allowances for measurement and prediction uncertainty are not available to review. As an example, DL3-16 is for ICS initiation, and relies upon Line Break Indication (MSL, FWL, CUW, ICS). Provide the setpoint for line break indication for each line, how are the line breaks diagnosed and differentiated from normal flow rates and under what conditions they are conditioned in or out.	5.5
12	4, 5, 7		The PSAR does not include a description of the neutron detectors and gamma thermometers instrumentation, e.g., whether they are prompt to changes in neutron or gamma flux. Identify the type of detectors used and their characteristics, and whether this is generic BWR instrumentation for this fuel design. If the detector characteristics are different than those assumed in Chapter 15, the safety analysis needs to be updated accordingly.	5.5
13	4, 15.5		The general approach of safety analyses is first to establish applicability of the computer codes (e.g., TRACG and GOHIC) for different BWRX transients. That is, to identify important phenomena for each transient and to show that code has models for important phenomena. The second step is to quantify accuracy in prediction of important phenomena for each transient by establishing relevant test matrix for each transient, ensuring scaling and relevancy of tests to the transient conditions, and comparing the code predictions with the test data. The relevant requirements and guidance for code applicability, code accuracy and safety analysis documentation are provided in CSA N286.7 (Clause 10), REGDOC-2.4.1 (Clause 4.4.5 and Item 6 in Clause 8.5). These code qualification steps have not been reviewed since they are documented in GE-H proprietary reports rather than the PSAR.	5.5
14	4, 15.5		Code predictions are affected by the time steps, convergence criteria, model selections and spatial discretization. This is described in Chapter 3 for the seismic design. For deterministic safety analysis, they are documented in GE-H proprietary reports, which have not been reviewed. Chapter 15 does not provide justification for use of safety analysis codes for each application.	5.5
16	4.8.3	Type 1 instabilities experienced during startup do not result in a reactivity/power response. Type 1 oscillations are characterized by initiation of vapor production in the chimney region leading to a reduction in hydrostatic head in the chimney and a resultant core flow increase, which, in turn, could cause voids to collapse in the chimney. The BWRX-300 reactor goes through an unstable phase during startup. This type of oscillation is unavoidable in a natural circulation reactor because the unstable power/flow region must be crossed prior to establishing a steady two-phase voided region in the chimney; however, the magnitude of the flow oscillations is typically very small. As Type 1 oscillations do not result in a change in core moderator	During Startup of the reactor, the reactor is filled with liquid and rods are slowly withdrawn. During this operation, the reactor may be in the region of Type 1 instability. Simulations need to be performed to confirm that this is managed by heating up the coolant in the vessel nearly uniformly in forced circulation mode by running the SDC pumps, while bypassing SDC heat exchangers, and then pressurizing to 6 MPa prior to reactor power exceeding 2.5%FP. During this phase, there may be geysering where vapor bubbles may condense in subcooled liquid in the chimney leading to a flow increase followed by a flow decrease as cold water gets into the core. Explain how geysering is managed.	5.3, 7.3

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		density, there is no power response and, as a result, the Fuel Cladding Integrity Safety Limit is maintained.		
18	Figure 4.4-2	BWRX-300 Power / Flow Map	<p>One infers from this figure that BWRX-300 design is operating at its maximum flow rate that is achievable, and at its limit for natural circulation flow rate through the core. That is, any further increases in reactor power does not result in an increase in coolant flow rate and might instead cause a decrease. Core coolant flow rate at full power, its distribution among channels and associated prediction uncertainties are a key input to CPR calculations (Section 4.4.1.1).</p> <p>Accurate core flow rate predictions during normal operation at power rely on accurate prediction of driving force (e.g., the void fraction in the chimney) and accurate prediction of hydraulic resistance (e.g., two-phase flow resistance multiplier in the core), as acknowledged in Sections 4.4.1.4 and 4.4.1.5. A high-level description of the method for predicting void fraction, pressure drop and flow distribution among channels is provided in Sections 4.4.2.3, 4.4.2.4 and 4.4.2.5 but the PSAR does not provide sufficient details for code prediction accuracy and prediction uncertainties. Convincing evidence that the code predictions for flow rate are accurate, based on code validation against properly scaled experiments, is needed to confirm that BWRX-300 can achieve the predicted nominal flow rate through the core, as well as its distribution among channels, during steady-state full power operation.</p> <p>Similarly, the PSAR lacks evidence for accuracy of void fraction predictions in the core during normal operating conditions. This is important because initial void fraction and its distribution is important for initiating events that involve a positive reactivity insertion due to void collapse.</p> <p>Adequate code validation and consistency of validation with the safety analysis is particularly important for natural circulation BWRX-300 since the flow rate and void fraction is expected to be tightly coupled, especially with ICS discharge location in the chimney region. Note that a review of the relevant GE-H proprietary reports (e.g., those referenced in Sections 4.3 and 4.4) is beyond the scope of this review.</p>	5.3

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19	4.2, 4.4	This methodology is applicable to the analysis of the fuel rod response for all transient events and has been qualified for performing fuel analysis for BWRX using US NQA-1 standard, which is equivalent to Canadian standard CSA N286.7. PRIME analyses are performed for the following conditions...The analytical methods described in this section are qualified according to the provisions of American Society of Mechanical Engineers (ASME) NQA-1 which are equivalent to the provisions of CSA N286.7 (Reference 4.4-1).	<p>The information presented in the PSAR alone is not sufficient to evaluate a first-of-a-kind reactor design that relies on natural circulation for fuel cooling. Various safety analysis and design assist codes (TRACG, PANAC11, PRIME, etc.) are used to calculate design margins and margins to DSA acceptance criteria. Codes need to comply with CSA N286.7 for each specific application (e.g., CN-DBA analysis for an IE) and need to provide sufficient justification for Code Applicability and Accuracy thru validation against relevant tests or existing in-reactor OPEX from other BWRs. The information provided in Section 15.5.1 and Appendix 3G for code qualification is at a high level (describing only the process of identification of important phenomena and code qualification including code accuracy in general terms) and is not sufficient to determine adequacy of code applicability and accuracy statements for each specific application. The references for code qualification and use (e.g., References 15.5-1, 15.5-3 and 15.5-7) are GE-H proprietary reports and their review is not included within the scope of IPR review.</p> <p>The following is an example to illustrate the code qualification concerns. Coolant flow rate distribution in the core, through each fuel channel, will be based on hydraulic resistance and its distribution along the channel including the orifice at the bottom, and void fraction/two-phase multiplier for each channel, as there is close coupling between bundle power, hydraulic loss coefficients and void fraction. Depending on the void fraction in the chimney and downcomer water level, flow rates and pressure drop across the core will vary and will be subject to prediction uncertainties. Use of codes' best estimates predictions, without accounting for operational variations and uncertainties and without quantifying and accounting for modeling uncertainties (bias and standard deviation) for predictions of Figure of Merit parameters erodes confidence in the safety analysis margins claimed in the PSAR. Reactor physics predictions will also be affected due to strong coupling between thermalhydraulics and reactor physics due to high negative void coefficient and Doppler effect, especially due to design basis accidents that collapse void in the core. This comment is applicable for Normal Operation (e.g., LHGR, MCPR, initial void fraction in the core and in the chimney) in addition to AOOs and DBA's.</p>	5.5
20	4.3.1.3, 4.4.1.1.2	The MCPR99.9% ensures that 99.9 % of the fuel rods in the core are not susceptible to boiling transition when considering the nuclear core design, plant system uncertainties, manufacturing uncertainties, and calculational uncertainties.	<p>This statement is supported by a BWRX-300 proprietary report which has not been reviewed by the IPR team. There should be no fuel element in boiling transition during normal operation, and for AOO's [REGDOC-2.5.2 Clause 8.1.1], and ideally even for most DBA's. If this MCPR99.9% criterion assures that no fuel element will be subject to boiling transition, after accounting for operational variations and modelling uncertainties, during normal operation and for AOO's at a 99.9% probability (or confidence level), the IPR team finds compliance with this criterion to meet the safety requirements for normal operation and AOOs with sufficient margins.</p> <p>Nevertheless, the quoted statement in the PSAR is open to misinterpretation, i.e., that 0.1% of the fuel rods is susceptible to boiling transition, even for during normal operation.</p>	5.7
21	4.4.2.5, 4.4.3.2, Figures in Section 15.5 showing Core Inlet Flow and Core Average Void Fraction	The bypass flow methodology is described in NEDE-32176P (Reference 4.3-4), Subsection 7.5.1.	Clarify whether the bypass flow, water rods and flow between the octagonal core and the shroud are included in the safety analysis core flows and void fraction shown in the figures. Clarify the directions of the bypass/water rod flows and flow between the octagonal core and the shroud during transients and accidents.	5.3

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22	4.6.1	The CRD system provides the primary means of reactivity control during normal, abnormal and accident conditions. The system design basis includes two diverse motive forces for the CRD insertion (scram) using high pressure water from the HCUs, and control rod insertion using the FMCRD motor. Incorporated into the design are positioning and protective features that prevent inadvertent withdrawal, drop, and ejection of the control rod due to a component break or other malfunction. Complete system functionality is described below in Subsection 4.6.2.	<p>REGDOC-2.5.2 has independence, separation and reliability requirements for shutdown function. BWRX-300 uses the same Control Rods both for reactivity/power regulation function (DL1 and DL2) and for shutdown means (DL3 and DL4a). For example, reactivity worth and rate of hydraulic scram and fast run-in is lower because some control rods are already partially inserted into the core prior to the accident.</p> <p>CSA N290.1 Clause 4.1.5 have requirements for separation of a Shutdown system (DL3, DL4a) and process systems (DL1 and DL2). REGDOC-2.5.2 also have various independence requirements between the levels of defence (e.g., Clauses 4.3.1 & 6.2), and between the two independent shutdown means. IPR team considers the two diverse motive forces of inserting the control rods into the core, i.e., hydraulic and electric motor, alone to be not sufficient to meet the independence requirements between the two shutdown means (DL3 scram and DL4a fast motor run-in) for mitigation of DBAs. The IPR Team recommends a defensible "alternative approach" to close this potential gap, through crediting DL3 and DL4a shutdown functions as the two shutdown means and demonstrating that the existing design meets the overall reliability requirements for shutdown function.</p>	5.2 c), d), 7.1, 7.2, 7.4 and Comment 23
23	3.1.6.3	Exceptions to rules of independence are described, assessed, and justified. If equipment supports functions in more than one defense line, there is an increased focus on their reliability in the application of DL1 compared to a design feature credited in only one defense line.	<p>REGDOC-2.4.1 and REGDOC-2.5.2 require two Independent Shutdown means. BWRX-300 has one DL3 shutdown means (hydraulic scram), and one DL4a shutdown means (ARI or fast motor run-in). While Boron Injection System (BIS - Ch 4.5.2) is a means to add negative reactivity, it has no dedicated, independent instrumentation, and is not automated. Hence, it is not considered a bona fide Shutdown means.</p> <p>Notwithstanding, BWRX-300 design provides strong DL2 & DL4a reactivity control/shutdown functions that are largely independent from each other and from the triplicated DL3 scram function. Overall reliability for reactor shutdown for DBA's needs to be such that failure rate cannot be higher than 1E-5 failure per demand (REGDOC-2.5.1 Clause 8.4.2). A defensible "alternative approach" supported by quantitative reliability calculations should be sufficient to meet the shutdown safety function requirements in REGDOC-2.5.2.</p>	5.2d), 7.1, 7.2, 7.4 and Comment 22
24	4.6.2	<p>The CRD system performs the following functions: 1. Control...in response to control signals from RC&IS...14...scram in response to signals from the ATS. 15...SCRR1 in response to signals from the ATS...18...scram upon receiving a scram signal from the SC1 I&C system. 19...scram upon receiving the scram signal from the DPS...21...fast run-in of the FMCRDs upon receiving a signal from the FMCRD controllers with input from the ERICPs...</p> <p>The CRD hydraulic subsystem provides clean, demineralized water that is used to charge the scram accumulators and purge water flow to the FMCRDs during normal operation. The CRD hydraulic subsystem is also the source of pressurized water for purging the SDC pumps and filling the Nuclear Boiler System (NBS) reactor water level reference leg instrument lines.</p>	<p>REGDOC-2.5.2 has requirements for independence between systems performing different levels of defense (e.g., Clause 4.3.1 and 6.1). The control rods in BWRX-300, as an example, are used for DL-1, DL-2, DL-3 and DL-4a functions. As another example, the CRD hydraulic subsystem (a DL3 function) is considered to provide pressurized water for non-DL3 purposes, e.g., purging the SDC pumps and filling the reactor water level reference leg instrument lines for various DL functions, and coolant water makeup (DL4b).</p> <p>Related to reference leg lines filled by the CRD system, an additional concern is whether the RPV Level instrumentation used by process and safety systems during accidents will continue to work if the CRD hydraulic system runs out of inventory during postulated accidents.</p>	5.2 c)

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27	15.B.1.8	Dedicated piping and collection for sodium pentaborate solution is provided where drainage occurs. The collection vessel is a portable stainless steel drum. A low containment surrounds the system preventing the spread of sodium pentaborate leaks.	Confirm that being a manual standby system BIS leaks cannot dissolve the RPV (e.g. as per Davis-Besse, etc. OPEX) as a result of testing, etc.	7.5
28	15.5.5.2.2, Table 15A-1, 15B.1.1	...the event assumes that half of the control rods with the highest rod worth fail to scram and the CRDM run-in fails to insert the rods that failed to scram...A severe and unlikely condition where more than 50% of the control rods fail to insert leading to an accident with increasing severity that challenges the reactor and containment system is practically eliminated...The BIS provides a means of achieving cold subcriticality by mixing a neutron absorber with the primary coolant...	The rationale for the complex DEC using less than 50% of the control rods is in an external proprietary report and was not reviewed.	A.1.0
29	4.6.2.2.1, Tables 4.6-2 and 15B-1, 15B.1.3, 15.6.1.10	<p>The scram accumulator stores sufficient energy to fully insert two control rods at any anticipated reactor pressure. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below...Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the MCR. The alarm would prompt operator action to repressurize the nitrogen bottle using an external supply of nitrogen gas.</p> <p>A single triplex injection pump capable of 2.27 m3/hr (10 gpm) pumping against the maximum reactor pressure of 12.41 MPaG (1800 psig).</p> <p>The ultimate pressure regulation function of the RPV provides emergency pressure relief in the event of a severe pressure transient.</p>	Provide some confirmation that for overpressure protection when the RPV pressure exceeds the SCRAM accumulator pressure and BIS pump head that DL2 ATS SCRAM, DL3 SCRAM, DL4a DPS ARI SCRAM and DL4b BIS would not be available. Explain how the scram and BIS functions operate at high RPV pressure approaching Level C conditions in the vessel. The overpressure protection report (NEDC-33910P) was not reviewed.	7.4

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30	Figures 4.3-1, 5.1-1 and 5.1-4, 5.1, 5.1.4.1, 5.3.1.3.1, 15.5.4.6.2, 15.5.5.3.1	<p>Innovative features related to BWRX-300 are the partitionless RPV chimney region flow characteristics (see Subsection 5.4.3); RIV qualification (see Section 5.10) and the RPV water level measurement in the core region (see Subsection 5.4.9).</p> <p>The water level in the reactor vessel is measured by the temperature compensated dP devices (calibrated for specific RPV pressure and temperature conditions) ... Normal operations (DL1) for RPV water level is controlled by reactor level control system.... Water level measurement by differential pressure is also used for transient/AOO (DL2) and DBA (DL3) monitoring... Because there are not any operator actions associated with water level between TAF and TAF+4m, there are no water level measurements available in this range.</p> <p>Gamma Thermometers (GTs) will be utilized for the fuel zone water level measurements...The water level inside of the core will show adequate coverage for core cooling without the need for an extra system in the downcomer or on the RPV Outside Diameter (OD). It is generally considered that the level inside of the core provides the best information.</p> <p>The indicated water level stabilizes above the actual collapsed downcomer level. This is because the wide range level is off scale when the actual level falls below the lower tap and no longer indicates level.</p> <p>Sensed level increases due to continuing FW flow and flashing in the downcomer.</p>	<p>Clarify that the Figure 4.3-1 level sensors in the fuel or chimney region are not the RLC, DL2 and DL3 level measurement differential pressure variable leg taps, which are in the downcomer and their transmitters are outside the primary containment providing WR reactor water levels more than 4 m above the TAF. Details on the measurement of WR reactor water levels and interpretation (re. collapsed liquid level, potential vapour in the instrument lines) should be provided for DL1, DL2 and DL3 and the setpoints for L9, L8, L5, L3 and L2.</p>	A.2.2
32	5.3.4.5.3	Coolant temperature is maintained above the design minimum temperature with the head bolts tensioned.	Add a reference(s) that explains the rationale(s) for the minimum coolant temperature for bolts tensioned.	Standalone comment
34	5.12.3	The RPV head vent subsystem includes piping internal to the RPV head, two flange-mounted in series RIVs, and piping to the MSL or Quench Tank, both located inside PCS.	It is unclear if the Quench Tank is a potential design feature or an included design feature. In the PSAR, it is said to be connected to the RPV head for venting. However, in NEDC 33922P REV2 Figure 2-3, it is shown to be connected to the ICS.	6.2.3.3
35	6	Containment	REGDOC-2.5.2 Clause 8.6.1: "The containment shall include at least the following subsystems: [...] 3. equipment required to ... reduce the concentration of free radioactive material within the containment envelope". Clarify how and if BWRX-300 meets this requirement.	6.2.3

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36	6.2, 15.5.4.4.2	This power-uprated ESBWR IC version is used in the BWRX-300. Other modifications in the BWRX-300 IC design include design pressure (higher for BWRX-300), return into the chimney instead of the downcomer (refer to Section 6.2.6) and hardware material selections made to enhance the design.	<p>REGDOC-2.5.2 Clause 6.8 requires analysis of spurious operation of safety systems such as ECCS (ICS for BWRX-300). For ICS spuriously initiating together with a successful DL2 or DL1 action for controlling RPV pressure and level (e.g., through Turbine Control Valve operation), up to 12% of reactor power may be removed by the ICS possibly under a new steady-state condition during the accident. If reactor power is also controlled by DL1 or DL2 reactor power and reactivity control actions at near Full Power, this could lead to continued near Full Power operation with substantially lower core coolant flow rate due to lower natural circulation driving force because ICS discharges into the chimney. This could impair effective fuel cooling.</p> <p>The analysis of spurious ICS initiation in 15.5.4.4.2 needs to include sensitivity cases where DL1 and DL2 systems are credited for controlling RPV pressure, level and reactor power, if the consequences can be worse. This analysis was performed at MOC with some control rods inserted but Chapter 15 does not mention the Plant Automation System nor the Automatic Power Regulator; would the Automatic Power Regulator be expected to withdraw the Control Rods slowly to maintain the reactor at full power?</p> <p>Also the safety analysis codes need to be validated for such a complex configuration preferably using integral effects tests, and code prediction uncertainties need to be accounted for in the analysis. These may be documented in a BWRX-300 proprietary report, which was not reviewed. Also, see the next comment.</p>	5.6, A.7.2
37	6.2.3	The ICS is placed in operation ... the subcooled condensate ... enters the RPV chimney interior providing additional inventory while also quenching steam and lowering pressure at the exit of the reactor core.	BWRX-300 lacks forced recirculation pumps and relies completely on natural circulation to cool the fuel, with the gravitational driving force arising from coolant density differences between the downcomer and the core/chimney. Adding cold ICS return condensate water into the chimney region impairs natural circulation flow through the core, potentially impairing fuel cooling. The ICS in existing BWRs return condensate water to the downcomer - whether the BWRX-300 configuration will be effective cannot be determined from OPEX for existing reactors which is considered not applicable to BWRX-300. Impairing or stopping natural circulation at lower reactor powers when ICS is in operation might result in counter-current flow from the chimney into the fuel region. Experimental evidence from appropriately scaled tests is necessary to ensure adequate fuel cooling during DBA's, to prevent flooding and ensure core stability. Need accurate modelling/experiments and quantitative validation of counter-current flow and parallel channel flows in complex geometries. Such tests and code qualification activities may be documented in BWRX-300 proprietary reports, which were not reviewed by the IPR team.	5.6
38	6, 6.2.2, 6.2.3	Isolation Condenser System (ICS), which functions as the BWRX-300 Emergency Core Cooling System (ECCS)...ICS performs reactor coolant inventory addition and decay heat removal functions in response to...Provide water inventory for a minimum of seven days for ICS decay heat removal, even when assuming one ICS train is unavailable	ICS serves as the ECCS, among other process and safety functions (pressure control, decay heat removal during non-accident conditions, etc). ICS is a substitute for ECCS but lacks emergency coolant injection capability with no dedicated water reservoir pumps/valves for injection. In fact, other than the condensate water in the return lines during normal operation, it does not have any water inventory. For small break LOCA's, where isolation function is not provided, and for other events where RPV is isolated, there is no safety-grade system that can inject coolant inventory to the RPV when needed, such as if the isolation valves are leaking or for small liquid line DBA breaks for 7-days. Only after 7-days, the operator has an available DL4b function (using the CRD hydraulic system) to inject water into the RPV provided that there is sufficient coolant left in the CRD system.	Executive Summary, 5.2, Comment 116

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39	6.2	The heat rejection process is continued beyond seven days if the ICS inner pool inventory is replenished from an outside source. The ICS pools are located above ground and are not pressurized. Clean makeup water can be added directly to the ICS cubicles using diverse and readily available sources	Identify the contingency equipment for an event requiring ICS makeup in freezing weather in the PSA.	6.1
40	6.2.3, 15.5		ICS condensates return line is connected to the chimney. How does it affect core flow and power? Is there any counter-current flow at the core exit? Are there any tests with ICS draining in the chimney?	5.6
41	6.2.3.4	Once the valves fully open, they stay fully opened until they are reclosed intentionally by the operator.	If reclosure of the ICS after inadvertent opening is possible/allowable, then it should be analyzed.	A.7.0
43	6.3, 15.5		Details on PCCS modelling are provided in a GE-H proprietary report, which was not reviewed. Confirm whether the heat transfer coefficient on the tubes is reduced based on their grouped configuration.	6.2
45	6.3.6, 9A.4.2.3, 15.6.1.10, 15.6.1.11, Table 15B-1	<p>Containment Overpressure Vent Flow path The containment overpressure vent flow path is used in case of a severe accident where containment failure by overpressure is threatened. The containment overpressure vent flow path consists of a rupture disc, a locked closed manual bypass valve, pressure indicator, isolation valve, check valve, sparger, and associated piping. The overpressure vent flow path connects to the containment exhaust flow path and terminates in the Reactor Equipment Pool. By relieving pressure into the Reactor Equipment Pool, water scrubbing is expected to occur, thereby reducing the contaminants. Additionally, the Reactor Equipment Pool is located inside the Reactor Building such that further holdup of any potential released contaminants is provided.</p> <p>Containment Filtered Venting...The release is filtered to reduce the amount of fission products.</p> <p>The vent is equipped with a filter.</p>	It is noted that there is no physical filter on the containment vent flow path and that the filtration is achieved via pool scrubbing. There are several instances in the PSAR which refer to a filter on the vent line and these instances need to clarify that the only filtration is via pool scrubbing.	6.2.3.2

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46	5.7.6.1, 5.11.5, 7.3.1.3.2	The valve motive power is typically stored nitrogen in accumulators independent of other nitrogen uses. The components of the RCPB, ... are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed in accordance with generally recognized codes and standards, and under an approved quality assurance program with approved control of records. In addition, means are provided to detect and identify the location of the source of reactor coolant leakage, ..., for components of the RCPB. ... Individual valve leak rate testing for diagnostic and rework/refurbishment is based on the design performance criteria in (TBD).	Describe / prove reliability of the RPV isolation valves, "commensurate with the importance of the safety functions to be performed".	5.6
49	7.3.1.3.3	When a bypass used, the 2-out-of-3 logic essentially operates as the equivalent of 2-out-of-2 to trip	The trip parameter instrumentation can be taken out of service for repair or maintenance. REGDOC-2.5.2 Clause 7.6.2 requires each safety group to perform the required safety functions under the worst permissible systems configuration, taking into account such considerations as maintenance, testing, inspection and repair, and equipment outage. This is typically achieved by forcing the channel into a tripped state either automatically or by the operator and the trip logic is reduced to one-out-of-two from the normal two-out-of-three logic. Provide justification that 2-out-of-2 to trip logic meets the Single Failure Criterion requirements.	Standalone comment
50	6.2.7.3, 15.2.4.6.3, Tables 15.5-49 and 50	<p>Each ICS train is equipped with two flow detection elements in the form of 90-degree bends that have impulse lines with three parallel Differential Pressure Transmitters (DPTs). One flow detection device is located in the steam supply line, and another located in the condensate return line. These DPTs detect flow within the system. If the ICS train is in standby and flow is detected, a line break is indicated and the control system for this line initiates RIV closure for this train only.</p> <p>DL3-27, -28, -29 ICS RIV closure of the broken ICS train line break indication in the respective ICS train within 1 second for breaks larger than 19 mm (0.75 in) in diameter</p> <p>There is either a DL2 or DL4a function for all credited DL3 functions in Table 15.5-49, except the isolation condenser pipe breaks. Because the heat removal by the ICs is a higher class safety function than isolation by a DL4a function, no DL4a associated function exists. An unisolated isolation condenser pipe break is similar to an unisolated MS pipe break and is evaluated separately subject to different acceptance criteria.</p>	<p>Analyze the DEC unisolated MS (and FW) pipe break.</p> <p>For the DBA ICS break, the minimum detectable leak should be confirmed to be a small LOCA considering there is an orifice just downstream of the RPV in steam. The dose calculation should be revised to include the maximum undetectable break.</p>	5.6

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52	15.5.1	The PIRT is used together with the TRACG documentation to systematically demonstrate the applicability of TRACG models and the qualification of the TRACG model to predict the phenomena. Defining the nodalization and evaluation of the effects of scale are included. In addition to code applicability and qualification, the PIRT is also used as the basis to perform quantitative uncertainty analysis of transient scenarios, if needed.	The analysis in Chapter 15 lacks sufficient detail in both the text, tables and figures to be able to follow the analysis and to assess whether it meets code applicability, code accuracy (prediction bias and uncertainty for Figure of Merit parameters), event sequence, event consequences, timing and credit for safety functions, and whether the analysis meets the acceptance criteria. This includes the LBLOCA discussion and analyses of some other events are absent altogether. The required information might be available in GE-H proprietary code qualification reports which were not reviewed by the IPR team.	5.5
55	Figure 15.5-15, 15.5-21, etc.	Control Rod Insertion Fraction	Clarify the definition of the 90% insertion shown in the figures as the fraction of active fuel adjacent to control rods when all the control rods are in the fully inserted position.	A.4.1, etc.
57	Appendix 15B, Table 15B-1	There are several complementary design features provided ensuring that DEC's are either practically eliminated or extremely unlikely to occur. Complementary design features mitigating functions are provided in Table 15B-[1].	It is unclear what complementary design features have been included in the design. PARS (aside from that in the ICS), CRD coolant makeup, Core Catchers, etc. are mentioned but it is unclear whether they have been implemented in the design. For example, Chapter 15 Appendix 15B on features for stabilization of molten core are not yet complete.	5.1
60	15.2.4.1	There is no fault group for increase in core coolant temperature for the following reasons: 1. The Feedwater (FW) temperature is near the highest temperature that is feasible during normal operation. 2. Any increase in FW temperature would increase the core void fraction and reduce core power due to the decrease in void reactivity. 3. An increase in FW temperature does not result in an increase in core temperature because the core is boiling and remains at saturated conditions.	FW temperature increase may lead to higher void fraction, higher hydraulic losses due to increased two-phase flow losses in the core and hence lower coolant flow rate. It also erodes dryout margins directly by higher steam quality in the bundles or indirectly by reducing mass flow rate and pose a potential hazard. Analysis for such events is missing. Justification provided in the PSAR for not including this event for analysis is not sufficient. Note that for this event, (DBA) analysis should credit reactor power control as it should make dryout margins smaller. CN-DBA analysis should credit DL1 and DL2 actions only if they lead to more limiting consequences.	5.4
61	15.2.4.2	There is no fault group for decrease in reactor pressure because a pressure decrease reduces reactivity. The decrease in reactor pressure is caused by either: • Reactor power decrease that is covered by the reactivity at power distribution anomalies fault group • Reactor coolant inventory is lost that is covered by the decrease in reactor coolant inventory fault groups.	RPV pressure decrease, as a stand-alone PIE, may lead to higher void fraction, higher hydraulic losses and lower overall coolant flow rate through the core, and also erode dryout margins directly and indirectly, and pose a hazard. Analysis of such events is missing. Justification provided in PSAR 15.2.4.2 for not including this event for analysis is not sufficient. Note that for this event, (DBA) analysis should credit reactor power control as it would make dryout margins smaller. For CN-DBA, any automatic action whether DL1 or DL2, or operator action should be credited in the analysis only if that makes the event consequences worse. It is also not clear from reading the PSAR how reactor power level control works (DL1 or DL2) for events that lead to mild negative reactivity insertion, such as a sustained reduction in RPV pressure leading to more void fraction in the core and hence negative reactivity insertion. Specifically, does the control system respond to withdraw a control rod to keep the reactor power level steady at 100%FP? Or do they remain in their initial state and reactor power simply decrease to a new steady-state lower level? Any automatic action whether DL-1 or DL2, or operator action, e.g., that increases reactor power, should be credited in the analysis if that makes the event consequences worse.	5.4

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62	15.2.4.4	<p>These events may occur from a FW Inventory Increase. Inadvertent ICS initiation scenarios do not fit well into any fault group and are included in this increase in inventory event category. Inadvertent ICS initiation results in much less Inventory Increase than a FW increase. However, the dynamics of ICS flow into the chimney is different than an increase in FW flow. These scenarios are selected for analysis...There is one BL-AOO event, Inadvertent Isolation Condenser Initiation – One Train, and it is selected for evaluation...Two CN-DBA events are selected:</p> <ul style="list-style-type: none">• Inadvertent injection of all ICS trains (bounds one train) (Inadvertent Isolation Condenser Initiation – All Trains (CCF-IICI))• Feedwater Flow Increase - All Pumps (CCF-FWFI). This event bounds the CN-DBA event for increase in flow of one FW pump.	<p>The only AOO analyzed leading to RPV inventory increase is due to spurious operation of the ICS. The analyzed event does not cover the events that would be as a result of RPV level control malfunction, as an example. These two events would have different consequences; a DL-2 RPV level control malfunction increases core flow rate and increases reactivity and fuel cooling, while ICS operation decreases core flow, and decreases reactivity and decreases fuel cooling. Need to expand the scope of AOO analysis to cover other causes that lead to RPV inventory or provide sufficient convincing evidence for not doing so.</p>	5.4
63	15.2.4.5	<p>There are two BL-AOO events in this group:</p> <ul style="list-style-type: none">• FW pump trip - one pump (FWPT)• Inadvertent opening of one TBV	<p>RPV inventory decrease as a result of RPV level control malfunction, as an example, should be analyzed as an AOO to demonstrate effectiveness of DL-2 safety functions for inventory control, power control and fuel cooling.</p>	5.4
64	5.4.8, Table 15.6-8	<p>Very -Small LOCA 3.4E-03/year</p> <p>Each of the two suction lines also contain a vacuum breaker hole at the top of the pipe loop immediately inside of the two nozzles to prevent inadvertent draining. These two suction lines are connected externally to the reactor through the associated RIVs to the CUW. The two suction lines internal to the reactor allow the CUW to remove water from the RPV bottom head area.</p>	<p>Small breaks are considered only as a DBA, without evidence in the PSAR that frequency of small instrument line breaks or valve leakages with substantial uncertainty over the predicted event frequency is below 1E-2/yr as per Event Identification and Classification requirements in REGDOC-2.4.1 which categorizes very small breaks as AOOs in Appendix A for illustration purposes and 2.5.2.</p>	5.4

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66	15.5.4.6.1	Scram initiated from high containment pressure	<p>For DL-3 reactor shutdown function, the first scram signal is credited for DBAs. REGDOC-2.5.2 Clause 8.4.1 requires, for <u>each</u> credited means of shutdown, a direct trip parameter to initiate reactor shutdown to meet the respective derived acceptance criteria. Where a direct trip parameter does not exist for a given credited shutdown means, there shall be two diverse trip parameters specified for that means. REGDOC-2.4.1 provides additional guidance for direct and indirect trips for inherently safe designs such as BWRX-300.</p> <p>BWRX-300 has only one DL3 shutdown means and is also equipped with a DL4a shutdown means (ARI + Control rods run-in), which is not credited in the CN-DBA analyses in PSAR. For all DBAs, recommend including sensitivity analyses to demonstrate that if the first (DL-3) signal for scram is not credited for a DBA, event consequences crediting DL4a shutdown means continue to meet the acceptance criteria for DBAs. Alternatively provide justification, such as using DL4a, demonstrating how the intent of reactor scram requirements of REGDOC-2.5.2 and REGDOC-2.4.1 is met for BWRX-300 for all DBAs.</p> <p>This comment regarding trip parameter requirements is a general comment that is applicable to all DBA events that require reactor scram.</p> <p>Demonstration of robust trip (shutdown) coverage is especially important for the positive pressure feedback events which can lead to a simultaneous increase in reactor power and less effective fuel cooling.</p>	5.2
68	15.2.4.6.2	FW conservatively assumed to trip at time zero and coasts down with a 3 second time constant	The actual rundown timing is unavailable to demonstrate that the assumption is conservative.	5.8
70	15.3.1, Tables 15.3-1 and 15.3-2	<p>Deterministic Safety Analysis Acceptance Criteria</p> <p>AOO:</p> <p>...</p> <p>Tcenter < Tmelt</p> <p>cladding strain acceptance criteria</p> <p>...</p> <p>DBA:</p> <p><20 mSv</p> <p>...</p>	The derived acceptance criteria in Tables 15.3-1 and 15.3-2 do not fully meet REGDOC-2.4.1 and Canadian regulatory practices and equally importantly OPG's economic interests for AOOs. For AOOs, which are fairly frequent events such as Turbine Trip, the AOO analyses need to demonstrate that DL-2 functions alone, without any DL-3 or DL4 actions, are effective such that there is no damage to fuel, structures, systems and components. The fuel should remain fit-for-use and ASME Service Level B be met. For AOOs, appropriate DAC are avoidance of boiling transition for fuel, and Service Level B for overpressure protection. As an example, fuel center melting temperature acceptance criterion used for AOOs is not appropriate due to potential for significant fuel clad deformation, fuel matrix changes and potential failure due to various fuel failure and damage mechanisms. For DBA's, appropriate Derived Acceptance Criteria examples include no fuel failure (with exceptions allowed), no fuel melting, and no large structural damage to the core and no prompt criticality. Document the rationale in compliance with REGDOC-2.4.1 requirements and guidance (Clauses 4.3.2, 4.3.4 and Appendix B) when setting DAC for AOOs and DBAs.	5.7
71	Table 15.3-1	"the containment remains within its design limit values."	Indicate whether temperature is a local or through wall average.	6.2.1
72	Table 15.3-2	Containment atmosphere remains sufficiently mixed such that deflagration or detonation thresholds are not exceeded.	Table 15.3-2 should provide quantitative criteria with clear thresholds so that compliance with the acceptance criteria can be quantitatively demonstrated in the safety analysis.	6.2.1

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74	3.16.9, 4.3.1.1, 15.5.5.1	<p>Although the neutron kinetics are programmed to model prompt critical transients, the methods used to generate neutron kinetic data, such as neutron speed, have not been qualified for prompt critical events.</p> <p>The fuel reactivity acceptance criteria are established in GESTAR (Reference 4.4-8) and each of the following fuel parameters must be negative throughout the life of the core:...Net prompt reactivity feedback originating from prompt heating of the moderator and fuel for a super prompt critical reactivity insertion accident (e.g., control rod drop accident)</p> <p>Control Rod Drop Accident – Practically Eliminated</p>	<p>Avoidance of prompt criticality for AOOs and DBAs, such as Turbine Trip, Load Rejection, or Spurious Isolation of RPV, is required but is not included in the list of acceptance criteria for DBAs in the PSAR (Table 15.3-2) and is not demonstrated in Chapter 15 analyses. Instead, it was briefly mentioned in Section 4.3.1.1 as a design basis and may have been addressed in an external document that was not reviewed.</p> <p>Prompt criticality is not allowed per REGDOC-2.5.2 (Rev 2, draft): "Prompt criticality is avoided in any postulated accident unless it is demonstrated (e.g., experimentally, operating experience) that the resulting energy deposition does not result in damage to fuel or the reactor coolant boundary."</p>	5.7
76	15.4.1	<p>Automatic, reliable actuation of the control rods with either stored energy or motors to shut down the reactor and maintain it in a guaranteed shutdown state via latching mechanisms</p>	<p>Use of the word "guaranteed" here may be understood to be within the context for compliance with the requirements in REGDOC-2.5.2, specifically Clause 7.11 Guaranteed Shutdown State. It is not clear in the PSAR how and if BWRX-300 complies with the requirements of Clause 7.11, e.g., two independent means of preventing re-criticality when in GSS. Although obvious, consider presenting core defuelling as a second means of ensuring the core remains sub-critical during outages. Include a discussion of the GSS assumptions such as evolution of various xenon, samarium and plutonium concentrations, case when control rods are o/s for maintenance, subcriticality margin and tolerance to errors/failures.</p>	Standalone comment
78	15.4.1, 15.5.5.2.2, 15.5.5.3.2	<p>There are no operator actions credited in responding to the events analyzed in Section 15.5 for the DSA.</p> <p>Operators initiate additional CRDM run-in signals or manually insert CRDM to insert the remaining control rods into the core. If ... unsuccessful, operators inject boron to shut the reactor down. Another option available to the operators is to decrease power by reducing FW flow. ...</p> <p>Operator action to initiate scram is expected...The fuel cladding may experience local failures if initial LHGR and CPR are more severe. ... limited to high-powered fuel rods in a few high-powered bundles near the control rod withdrawn in error. However, the fuel failures are localized, the core remains cooled, and no core damage occurs.</p>	<p>Without credit for operator actions, the reactor may not be actually shut down. Perhaps shutdown is not necessary or desirable for all AOOs, but operator actions are required for some DEC after 72 h. In the case of ICRW, the reactor is left at 112%FP for 72 h with the potential for fuel failures and in the complex sequence load rejection, the reactor is left at about 40%FP for 72 h.</p>	Standalone comment

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79	15.5	Deterministic Safety Analysis	General comment: The level of detail in Section 15.5 is quite different from OPG's Safety Reports. The PSAR provides only a high-level description of the sequence with references to tables and figures and stock wording that repeats in every section. While this is convenient for reducing effort in future revision of the PSAR, it does not provide meaningful analysis of the results. There is no in-depth discussion about the results shown in the Figures. There are no actual results (values) mentioned within the scenario descriptions and only references to acceptance criteria. An in-depth discussion of the results, with details about assumptions, and data directly comparing to acceptance criteria values is the norm for OPG.	5.8
80	15.5	Deterministic Safety Analysis	There is no consideration for pre-equilibrium (initial fuel load) analysis and aging analysis. ICS, PCCS tube aging/fouling should also be considered.	5.8
81	15.5	Deterministic Safety Analysis	General comment: Although there is mention of analyses run at BOC, MOC, and EOR, only one analysis for the limiting cycle exposure is shown in the PSAR. There is no requirement to provide extensive analyses for all states, but evidence needs to be provided that the analyzed cases are indeed limiting cases for each DBA/AOO scenario for an event category REGDOC-2.4.1 provides guidance for grouping of events, specifically Clause 4.2.2.6: "In the safety analysis of AOOs and DBAs for Level 3 DiD, bounding events should be identified for each applicable acceptance criterion within each category of events." Similarly, REGDOC-2.5.2 Clause 7.3.2 provides guidance: "The rationale for the choice of these selected bounding events should be provided."	5.8
82	15.5	Deterministic Safety Analysis	All the analyses results presented in Chapter 15.5, have provided some high level qualitative explanations but lack detailed explanation of various changes in core flow, reactivity, core average void fraction, feedwater and steam line flows, ICS flows, fuel and clad temperature over time. The documentation for DSA in the PSAR is not comprehensive and sufficiently detailed. REGDOC-2.4.1 Clause 4.5 requires safety analysis documentation to be comprehensive and sufficiently detailed to allow for a conclusive review. Additional guidance for expected level of details can be found in Clauses 4.4.2.9, 4.4.5.2 and 4.5.	5.8
83	15.5		It is not clear how the (core) average void fraction is defined (e.g., static, neutron flux-square-weighted) and computed, and its significance to void reactivity changes (reactor physics) and core flow rate (thermalhydraulics) is not discussed. As an example, core average fraction decreases steadily after 0.5 s (Figure 15.5-26) reaching about half of its initial value at 3 s. The void reactivity is not monotonous and is slightly negative at 3 s. This void fraction-void reactivity behaviour in the first 3 s is unexpected, and lack of one-to-one correspondence is not explained. This is possibly due to average core void fraction, while it is not clear how it is defined or calculated, not being a good measure for void reactivity.	5.3

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84	15.2.4.6, 15.5.9.2.5	<p>The leakage detection system detects breaks in large pipes connected to the RPV but may not be capable of detecting breaks that are smaller than the area of a circle with a 19 mm diameter.</p> <p>A circumferential rupture of an instrument line connected to the primary coolant system is postulated to occur outside primary containment in the RB...Primary coolant flows at the maximum rate for a typical instrument line that has a ¼” flow restricting orifice.</p>	<p>Provide analysis for a break the size of a 19 mm diameter circle that is not in a line with an orifice. (The analyzed small LOCA credits the ¼” orifice. That means the RPV isolation has to detect and isolate all break discharges larger than this analyzed flow rather than 19 mm in diameter. The minimum detectable break size must be smaller than the ¼” orifice for the existing scope of analysis.)</p>	A.2.3
91	15.5.1.2.1	TRACG Code Applicability and Accuracy; Other Codes	<p>The discussion in the PSAR about code qualification is not focused on BWRX-300 design, which is different than most other BWRs due to natural circulation driven flows, and ICS condensate return line to the chimney region impeding natural circulation driving forces. Code applicability and accuracy for each AOO and DBA IE needs to be assessed and documented, with particular attention to the code accuracy being based on validation exercises from applicable separate effect and integral effect tests or OPEX from operating stations. This comment is also applicable for analysis of Normal Operation, which is largely lacking in the PSAR – the basis for important code predictions for design such as normal operating core flow rates, initial void fraction, decay ratio for stability purposes at various power levels and operating conditions are not provided. Code prediction uncertainties and operational variations need to be accounted in the DSA for DBAs. We also need confidence in code predictions for Normal Operating Conditions for economic reasons to ensure that BWRX-300 will produce the rated output with sufficient safety and operating margins. This comment can be considered largely applicable to other codes used in DSA.</p> <p>For TRACG, PSAR references 15.5-1 and 15.5-3 provide assessments of code qualification and code accuracy for each event category. Similarly for GOTHIC, Reference 15.5-6, another proprietary GE-H report, documents code qualification. Review of these GE-H proprietary reports is not included within the scope of this independent peer review.</p>	5.5
93	15.5.2		<p>Stability analyses for startup are not provided. Also, instability boundaries for core wide and region wide instabilities should be provided on the power flow curve based on frequency domain codes indicating the margins.</p>	5.3

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94	15.5.2	Analysis of Normal Operation including Stability Analysis	<p>Analysis for Normal Operation is generally lacking. BWRX-300 is a unique, new design and particular attention needs to be provided for analysis of normal operation. The analysis should include, at a minimum, justification for overall core flow rate, individual channel flow rate, CPR distribution for each channel, spatial neutron flux and heat deposition distribution, and coolant quality and void fraction distribution. As an example, initial void fraction is important because it determines the amount of maximum positive void reactivity insertion during certain AOOs and DBAs such as Turbine Trip. For stability, the design solely relies on the Decay Ratio acceptance criterion (<0.8). This parameter is found to be quite sensitive to operating parameters such as feedwater subcooling and hence lacks sufficient justification for sole reliance on it for normal operation – need to demonstrate that small uncertainties in the operating parameters or simulation of transients does not lead to instability.</p> <p>The chimney discharge location for those IEs that require ICS operation brings additional concerns regarding stability and might contribute to unexplained and unexpectedly large core flow oscillations before and after reactor shutdown – see core-wide flow reversal in Figure 15.5-82 before 50 sec and after 60 sec, respectively.</p> <p>Information provided regarding Normal Operation analysis in Section 4.4.3.2 is too high level. Information provided for the stability analysis in Section 4.8.6 is also high level and refers to GE-H proprietary reports, e.g., TRACG code qualification and stability analysis methods, that have not been reviewed.</p>	5.3
95	15.5.2	Load following	<p>Load following results in frequent changes to reactor power level and large changes in fuel temperature. Such changes in temperature may adversely affect fuel matrix/cladding integrity. A brief discussion on load following impact on fuel integrity is provided in Section 4.3.4.7 and refers to fuel design limits/methods in Section 4.2. It is not clear if there is any OPEX using GNF2 fuel in frequent load-follow mode. The info presented does not provide assurance that BWRX-300 fuel can withstand frequent load following, including due to potential enhanced fission gas release from the fuel matrix due to frequent temperature changes.</p>	5.3
96	Tables 15.2-2 and 15.6-7	IE: Power/Reactivity Increase Events as an AOO	<p>The fault list is documented in a proprietary report and was not reviewed. Further rationale for selecting the limiting events should be provided. Reactivity and/or regulated power increase events, e.g., as a result of reactor power or reactivity control malfunctions, need to be considered as Initiating Events in the AOO category (and separately in the DBA category) in accordance with the guidance in Appendix A of REGDOC-2.4.1, unless substantial quantitative evidence is shown for their event frequency falling below 1E-2/yr. Analysis of such events is important to confirm various aspects of DL-2 functions; reactivity control, power control, fuel cooling, as well as effectiveness of DL3/DL4a shutdown means for DBAs.</p> <p>AOO analysis needs to demonstrate that Level 2 defenses that limit core power changes are adequate to prevent adverse impacts.</p>	5.4
98	Table 15.5-3 Table 15.5-1	Input Parameters and Initial Conditions and Assumptions Used in Non-LOCA Analyses Key Initial Conservative LOCA Evaluations	<p>Table 15.5-3 is referenced in AOO, DBA, and DEC analysis. However, certain parameters such as initial pressure and temperature that have high sensitivities should be different since DBAs use conservative assumptions that define the Safe Operating Envelope limits while AOO and DEC use realistic assumptions. Different DBA event sequences may have different initial operating values depending on whether a high or low value is more limiting. A table with nominal vs conservative assumptions for each DBA should be included. Similarly, Table 15.5-1 should include nominal values vs. analysis assumptions to confirm that the CN-DBA analysis</p>	6.2.1

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			is performed in an appropriately conservative fashion and can be used to define the Safe Operating Envelope in accordance with CSA N290.15.	
99	Table 15.5-3		Some steady state operation conditions are provided in Table 15.5-3. GE-H proprietary reports documenting additional steady state operation conditions such as various flows, void distribution, temperatures and pressure drops, clad temperature, axial and radial power profile and peaking, power density were not reviewed.	5.8
100	15.5.3.2.3	Loss of Condenser Vacuum (LOCV)	In Figure 15.5-24, the steam flow and feedwater flow are decreasing. However, the FW flow is higher than the steam flow, but downcomer water levels are going down. It is not clear how mass balance is maintained without explaining that scram and TCV closure results in void collapse in the chimney causing a reduction in the indicated level in the downcomer, which causes the RLC to increase the feedwater flow.	A.3.0
101	15.5.3.2.4 LOPP AOO	Results: The ICS continues to limit the pressure increase.	ICS is not modelled in many of the accidents in Section 15.5 yet statements regarding ICS effectiveness are made without explicit reference to a bounding ICS analysis.	A.4.0
102	Figures for 15.5.3 and 15.5.4	AOO and DBA Analyses	<p>Figures need to be updated to include important aspects of analysis. Certain events need to focus on the first few seconds into the transient especially for reactivity and power response, while some other figures need to focus on long term evolution from the perspective of long-term fuel cooling, and stability and reactivity control, and in certain cases both the short-term and long-term plots are required. The x-axes and y-axes scaling of many figures is not very useful to figure out evolution of the event, e.g., whether prompt criticality is a concern, and whether reactor power, pressure and water levels are stabilized. The discussion in the text is also sparse and not convincing. Examples: 1) Figure 15.5-19 shows that the water level is still decreasing. With no coolant make-up capability, it is not clear if long term fuel cooling is assured. 2) Expand the y-axis in Fig 15.5-31 to show reactivity balance details.</p> <p>REGDOC-2.4.1 Clause 4.5 requires safety analysis documentation to be comprehensive and sufficiently detailed to allow for a conclusive review. Additional guidance for expected level of details can be found in Clauses 4.4.2.9, 4.4.5.2 and 4.5.</p>	6.1
103	15.5.4	Analysis of Design Basis Accidents	For many of the events described in this section, the Single Failure assumed is a failure of one ICS train. Given that the ICS is designed to be able to remove sufficient heat with even just one ICS train, there should be consideration of a different SFC in the other DL3 functions such as scram or RPV isolation that could be more limiting. This discussion should be included in the PSAR.	3.5.2

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104	15.5.4		<p>For DL-3 reactor shutdown function, BWRX-300 may potentially be considered as a Reactor with Inherent Safety per Table 3 of REGDOC-2.4.1, but this needs documented justification in Chapter 15. For such reactors, Table 3 requires one or two effective trip parameters, depending on the type of trip parameter (Direct Trip or not). In Chapter 15.5.4 CN-DBA analyses credit the first trip but without providing justification in Chapter 15.</p> <p>In addition, BWRX-300 is equipped with only one DL3 shutdown, while two independent shutdown means are required per REGDOC-2.5.2 and 2.4.1. The IPR team considers that the DL4a shutdown means has the potential to fulfill the requirement for the second shutdown means. It is highly recommended that the CN-DSA for DBAs is also analyzed crediting only the DL4a shutdown means.</p> <p>In addition, if a Direct trip parameter is not available for a DBA, the analyses showing effectiveness of either the DL3 or DL4a shutdown means should only credit the second Trip Parameter for each shutdown means.</p> <p>In addition, a dual trip coverage for both shutdown means may also be needed for those PIEs that result in positive pressure feedback, i.e., those events that result in positive void reactivity feedback due to void collapse and less effective heat removal from the fuel and the vessel.</p>	5.2, 7.1, 7.2, 7.4
105	15.5.4	Crediting ICS operation for reactivity control	ICS operation while at power leads to void collapse in the chimney and a reduction in core flow for some PIEs. This is due to cold condensate water initially in the ICS return lines discharging into chimney and condensing steam in the chimney region which also helps control RPV pressure. This negative impact on pressure by the ICS introduces negative reactivity. Analyses in Section 15.5.4 show that the DL3 shutdown mean has sufficient negative reactivity depth and insertion speed. The reactivity feedback characteristics of ICS operation should not be used to justify ICS discharge location in the chimney as the DL3/DL4a shutdown means are able to shutdown the reactor without any assistance from ECCS/ICS.	5.6
107	Figures 15.5-63, -69		There are no figures for CPR. Also, PCT figures (Figures 15.5-63, 15.5-69) show changes in PCT without explanation.	A.4.2, A.8.1
108	15.5.4.2.4	Spurious Closure of RPV Isolation Valves	<p>For this DBA, the reactor power, inlet core flow rate and average void fraction goes to near zero by 10-30 sec, and core inlet flow rate and reactor power remain near zero during the rest of the transient. Yet, the void fraction increases to a level during normal operation at power, which is not explained in the text. This may be due to absence of the driving force for normal flow path due to discharge location of the ICS in chimney. Confirm whether any fuel in the core is cooled by counter-current flow or parallel channel flow phenomena with no net driving force through the core. TRACG may have been already validated against relevant, ideally full scale, experiments to ensure that flooding limit in one or more fuel channels is not exceeded, or that the code is capable to accurately predict parallel channel flow phenomena in the absence of a net driving force. Counter-current flow/flooding and parallel channels flows in alternating directions are difficult to model accurately and validate.</p> <p>GE-H proprietary code qualification reports have not been reviewed by the IPR team.</p>	5.5

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109	15.5.4.3	Fuel Loading Error (FLE)	No results of analyses have been provided other than public dose. Boiling transition for one or more bundles and clad damage or failure is noted. Analysis needs to be documented in detail for the limiting FLE scenario to demonstrate that the consequences of this DBA are within the Derived Acceptance Criteria for DBA's (Table 15.3-2).	A.12.0
110	15.5.4.3.1, 15.5.4.6.3, 15.5.5.1		Some transients are listed but no transient results were provided (15.5.4.3, 15.5.4.6.3, 15.5.5.1). ICS line breaks are both liquid break and steam break and are different from either steam line break or FW water line break. The break connections to the vessel are also different. ICS is an important system, and this accident should be analyzed.	A.2.0, A.12.0
111	15.5.4.3.1, 15.2.1.2	With no automatic MSIV closure, the activity is transported to an offgas system. The activity release to the environment would occur from the normal offgas release point after holdup in the offgas treatment system. Conservative Deterministic Safety Analysis CN-DSA primary objective is demonstrating the effectiveness of DL3 functions. ... CN-DSA credits only DL3 mitigation functions.	Please clarify the rationale for the SC3 classification for OGS.	6.3.1
112	15.5.4.4.2	Inadvertent Isolation Condenser Initiation - All Trains	The SFC for this event is not stated. The assumed single failure should be clearly stated and justified in the text of the PSAR.	A.7.2
113	15.5.4.4.2	Inadvertent IC Initiation - All Trains	This event credits a scram signal on low RPV pressure. For reactor pressure to decrease, normally operating RPC function(s) must also fail. In general, CN-DBA analysis has to show that DL-3 system can fulfill their safety function without the aid of any DL-1 or DL-2 functions and needs to be performed in such a manner that is conservative. In this case, normally operating RPV pressure control function (e.g., by partially closing the TCV), which is expected continue to operate in any case during the accident, should be credited to function normally such that the RPV pressure is maintained near its normal set-point. In this case low RPV pressure scram signal will not come in. The event consequences are then anticipated to be more severe when the normally operating RPV pressure and level control functions are credited since the ICS will continue to return condensate water to the chimney impeding natural circulation core flow and potentially affecting fuel cooling adversely while reactor is at power. The control logic for the BWRX-300 reactor power/reactivity control (APR and RC&IS) functions for a scenario involving a negative reactivity insertion, such as due to an increase in core void fraction during this event is also expected to operate trying to restore power at the demand set point. The reactor can operate at full power at a new steady state with significantly reduced core flow rate during this IE, which would challenge acceptance criteria for fuel integrity. The analysis for this event needs to be performed in a conservative manner.	A.7.2

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114	15.5.4.5.1, 15.5.5.3.2	<p>The event sequence assumes a spurious CCF causes the loss of all FW flow and ...fails as-is RPC.</p> <p>RPC and RLC are assumed to function normally because this prolongs the event and makes it more severe for cladding temperature effects</p>	<p>The RPV pressure control is also assumed to fail during this DBA. It is not clear if this is a reasonable or conservative assumption since RPC is expected to operate normally. RPC operating normally could prevent MSRV/MSRV initiation and steam flow from the RPV would continue to deplete coolant inventory after reactor shutdown. PSAR needs to document the basis for RPC failing as being conservative, or the analysis should be conducted with RPC to continue to function as designed. The concern relates to TCVs working in such a manner that keeps steam lines open at a controlled pressure. That would reduce the inventory in the RPV continuously if the MS lines are not isolated. The DBA analysis needs to be performed in a fashion that a DL-3 function does not benefit from operation or failure of a DL-1 or DL-2 function. With MS lines open and FW lines isolated, inventory loss will continue, with changes in coolant flow rate through the core which may more adversely affect fuel cooling.</p> <p>In addition, large fluctuations predicted in core flow rate (and even core wide flow reversals), and core void fraction shown in the Figures for this section, even after reactor shutdown, are a concern.</p>	A.10.2
116	15.2.4.6, 15.5.4.6.4, Table 15.6-7, Figure 15.5-117	<p>The scenarios for LOCA developed in Section 15.2.4 bound the CN-DBA sequences, demonstrating the fuel and containment integrity acceptance criteria are met for at least 72 hours using only passive heat removal systems.</p> <p>CN-DSA LOCA analyses demonstrate that the reactor level does not decrease below the Top of Active Fuel (TAF) or the fuel cladding does not exceed the normal operating temperatures.</p> <p>Small breaks are analyzed using conservative assumptions demonstrating that fuel and containment integrity are maintained for at least 72 hours using only passive systems after which injection is recovered and the event is terminated. The LOCA acceptance criterion for demonstrating fuel integrity is to show that fuel cladding does not heat-up beyond normal operating temperature. ...</p>	<p>Small (unisolable) liquid line breaks predict a decrease of the collapsed downcomer water levels in the RPV below the TAF at about 57 hours, i.e., before 72 hours. Fuel clad temperature or the two-phase water level in the core is not shown in Figures. The water level in the core is also not shown but is expected to be only slightly higher than that in the downcomer due to expected low void fraction in the core, which is also not shown in the PSAR.</p> <p>Extrapolating the decrease in water level shown in Fig 15.5-117 to seven days (see below), the collapsed water level in the downcomer and core is expected to be well below the TAF. This core uncover is expected to lead fuel heat up beyond the normal operating temperature for fuel cladding.</p> <p>It is not clear how the event will be terminated at 72 hrs. In accordance with BWRX-300 design philosophy, no safety grade water make-up capability can be credited until the DL4b CRD hydraulic system is in service for coolant injection at seven days, provided that it has coolant inventory.</p> <p>The IPR team also considers potential for flooding of containment due to long term coolant addition to the RPV and potential for non-condensable gases in the containment entering into the RPV and reducing ICS effectiveness.</p>	5.2, A.2.3
117	15.5.4.6.1 and 15.5.4.6.3	Main Steam Pipe Breaks Inside the Containment, Conservative Case and Large Isolation Condenser Pipe Breaks Inside the Containment	One ICS train is credited. However, it is unclear which one is credited. Please include details on which train is credited (e.g., last one to initiate, one with the smallest pool volume, etc.) and why it is conservative.	A.2.1, A.2.2, 6.2.1
119	15.5.4.6.4	A small pipe break on the instrument lines may remain unisolated indefinitely. Since the isolation condensers depressurize the RPV, the break flow becomes very small in a few hours. Fuel heat up does not occur even without injection to the RPV.	A liquid line break with an (instrumentation/pressure tap) opening at the lowest elevation in the RPV needs to be analyzed. Figure 15.5-119 suggests that the analysis assumes significant reduction in the coolant discharge flow rate by assuming release of steam instead of liquid when the RPV water level drops to a certain level in about 4000 sec. Clarify that this analysis was performed for the pipe break for the instrument tubing with an opening at the lowest elevation in the RPV downcomer.	A.2.3

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123	15.5.4.2.1	Generator Load Rejection or Turbine Trip	There is an increase in PCT in Fig 15.5-57. How does CPR behave? There is a need to provide expanded plots for first 10 seconds.	A.1.0
124	15.5.5.2.2		Only one transient was analyzed where the control rod worth was 50 % (15.5.5.2.2). It will be informative to do sensitivity analyses on rod worth for some of the other initiating events with relatively high event frequency that have reactivity increase in DEC.	Standalone comment
125	15.5.5.2.2	Complex Sequence of Generator Load Rejection or Turbine Trip	Fig 15.5-135 shows that PCT occurs at 5 seconds. How much is this peak dependent on time step and plot frequency? What is the value of CPR? Fig 15.5-132: Explain that FW flow increases initially as RLC and RPC maintain RPV water level and pressure while TBVs open.	A.1.0
126	15.5.5.2.2	Complex Sequence of Generator Load Rejection or Turbine Trip	Figure 15.5-133 suggests that the total reactivity is zero after about 60 seconds, and Figure 15.5-130 shows reactor power is about 50%FP and increasing until about 160 seconds. It is not clear if and how this event is terminated in a manner that ensures core cooling in the long term.	A.1.0
127	15.5.5.2.3	Loss of Condenser Vacuum	Why does neutron flux oscillate so much during the rod run-in period and go up to 200% (Fig 15.5-136), while core average void fraction is nearly stable, increase in RPV pressure is as expected (Figure 15.5-140) yet total reactivity oscillating around zero? Discuss the phenomena that causes this behaviour. Is this in a region of instability? What happens if the control rod worth is only 50% or if the control rod insertion rate is slower? Explain the peak RPV pressure behaviour (Figure 15.5-137) and PCT (Figure 15.5-141) at around 40 seconds.	A.3.0
129	15.5.5.3.1	All Control Rod Withdrawal at Power (ACRW)	The PCT is shown in Fig 15.5-153 and PCT increases but turns around with scram. What happens if hydraulic scram (DL3) fails and FMCRD (DL4a) drives the rods into the core?	A.12.0
130	15.5.5.3.1, 15.5.5.3.2	Reactivity Withdrawal events	No evidence is provided in Table 15.6-7 or 15.6-8 regarding the frequency of such events to be classified as DEC. These types of abnormal reactor power regulation initiating events should be considered as AOO and DBA to demonstrate that DL2 and DL3 reactivity control/shutdown functions mitigate consequences of loss/malfunction of reactivity control or of reactor power control. See REGDOC-2.4.1 for a list of Initiating Events to be considered.	A.12.0
131	15.5.5.3.2	Inadvertent Single Control Rod Withdrawal at Power (ICRW)	It is important to know how the channels near the removed control rod behave. Was there any oscillation in individual channels? How does the CPR change over time? Fig 15.5-157 shows reactivity. There is not much change but explain whether the control rod part increase when there is no further withdrawal or insertion of control rods is because rod worth increases with lower void reactivity.	A.12.0
132	15.5.5.3.2	Inadvertent Single CR removal	Confirm whether the setpoints for STP (& High Flux) are set so that boiling transition leading to high fuel clad temperatures is precluded, while accounting for code prediction uncertainties and typical BWR unsteady flows. The sensitivity studies in the PSAR for this event indicate fuel failures in the affected bundles. There does not seem to be any automatic DL2/DL3 or DL4a action that can arrest this event from progressing until an operator action can be credited. This comment is in addition to Comment #96 on event classification for this event category.	A.12.0

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133	15.5.5.4	Decrease in Reactor Coolant Inventory - DEC	This loss of flow event predicts an associated power decrease. Clarify that the BWRX-300 automatic reactivity/power control operates too slowly (~0.5% FP/minute) to control reactor power when negative reactivity is inserted due to such an event and include modelling such automatic action in all future DSA.	A.10.3
134	15.5.8	A subset of the more than 700 isotopes from this inventory are used to model DBA dose consequences. The 60 isotopes used for DBAs are the dominant contributors to immersion and inhalation doses from airborne activity released during a DBA. This set of nuclides consist of 54 isotopes identified in WASH-1400 (NUREG-75/014) and 6 isotopes identified in SAND-85-2575 (NUREG/CR-4467). This is the group of isotopes typically used for AST dose evaluations	Confirm that this group of isotopes has been shown to meet the CSA N288.2 requirement in Clause 5.5.1 "The radionuclides considered in the assessment shall contribute more than 95% to the total dose".	6.3
137	15.5.9.2.5	Saturated water flows in the instrument line into containment that flashes to steam, resulting in the maximum iodine release.	This appears to be an error as it is unclear why there is flow into containment when the break is outside of containment.	6.2.1.2
138	15.6.3	However, uncertainty in these analyses will be formally explored in more detail in future PSA work. Based on the PSA work to date, CNSC REGDOC-1.1.2, Section 4.4.5 (Reference 15.6-8) requirements are not met at this time because Operating Manuals, EOPs and SAM program are not defined. The PSA is updated when the design matures and operational documents including Operating Manuals, EOPs, and SAM program are developed. PSA is further updated for the Operating Licence Application as the design is finalized and operational information matures.	The CDF/LRF uncertainties and severe accident consequence DSA are not reported in PSAR Rev 0. The proprietary PSA Summary report was not reviewed.	6.4
139	15.6		Provide a reference covering accident (NRX, Browns Ferry, TMI, Fukushima, etc.) OPEX and how the BWRX-300 design has addressed the lessons learned. It is understood that an OPEX report exists but reference to it could not be found within the PSAR.	7.0
140	15.6.1.10	The CRD system also provides an RPV inventory makeup function.	Small unisolable breaks, with no safety grade make-up capability, e.g., lack of ECCS/ICS with coolant injection, could cause water levels in RPV to go below the Top of Active Fuel in the first 7 days following the event. Clarify and justify whether the CRD inventory make-up function is already credited in CN-DBA and EX-DEC analyses. It is understood that a DL4b RPV coolant injection function using the CRD system is under development that can be credited after 7 days.	5.2