REQUEST FOR CONFIDENTIALITY

OF MATERIAL SUBMITTED IN RELATION TO CMD 24-H5

In the matter of:

Ontario Power Generation - Request for Authorization to Operate Pickering Nuclear Generating Station Units 5-8 until 2026

This request has been prepared in Canada, in the province of Ontario, in the matter of *Ontario Power Generation - Request for Authorization to Operate Pickering Nuclear Generating Station Units 5-8 until 2026***,** scheduled for consideration in a public hearing, scheduled for June 2024.

I, Riedewaan Bakardien, Senior Vice President of 1675 Montgomery Park Road, Pickering, Ontario L1V 2R5, am an authorized representative of Ontario Power Generation Pickering Nuclear Generating Station (NGS). 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 protection 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**
1.	OPG Letter, J. Franke to R. Richardson, "Pickering NGS - Units 5 To 8: Completion Of Pickering B Probabilistic Safety Assessment Update", December 15, 2022, CD# NK30-CORR-00531-08580, e-Doc 6937341	✓ Entire content ☐ Redacted content as shown: Correspondence letter NK30-CORR-00531-08580 is deemed **confidential** as it contains personal information. Attached report ("Pickering Nuclear Generating Station B Probabilistic Safety Assessment Summary Report", CD# NK30-REP-03611-00021-R002) is deemed **NOT confidential and is also available on opg.com.**

This request is made pursuant to the following paragraph(s) of rule 12 of the CNSC Rules of Procedure:

- Rule 12 (1) (b) the information is confidential information of a financial, commercial, scientific, technical, personal or other nature that is treated consistently as confidential and the person affected has not consented to the disclosure.

Further,

1. The above-noted material should be protected for the following reasons:
   - Rule 12 (1) (b) – The correspondence letter contains personal information (name/phone number) which the person affected has not consented to the disclosure of.
2. I attest that the above-noted material is not available through any public sources.

3. 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.
4. 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).
5. 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.
6. 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:

- OPG Letter, J. Franke to R. Richardson, "Pickering NGS - Units 5 To 8: Completion Of Pickering B Probabilistic Safety Assessment Update", December 15, 2022, CD# NK30-CORR-00531-08580, e-Doc 6937341.
- Attachment to NK30-CORR-00531-08580 and Non-Confidential Summary: OPG Report, "Pickering Nuclear Generating Station B Probabilistic Safety Assessment Summary Report", November 18, 2022, CD# NK30-REP-03611-00021-R002.

Authorized signature:

Riedewaan Bakardien,
Senior Vice President,
Pickering Nuclear Generating Station

2024/03/28
Date

REQUEST FOR CONFIDENTIALITY

OF MATERIAL SUBMITTED IN RELATION TO CMD 24-H5

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- 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 protection the following information, pursuant to rule 12 of the *Canadian Nuclear Safety Commission Rules of Procedure*:

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TABLE 1: MATERIAL TO BE DEEMED CONFIDENTIAL

	Item Name	Portion(s) to be Deemed Confidential
1.	OPG Letter, J. Franke to K. Campbell, "Pickering NGS - Units 5 to 8: Submission of Pickering 'B' Hazard Screening Analysis", January 26, 2022, CD# NK30-CORR-00531-08395, e-Doc 6726780	✓ Entire content ☐ Redacted content as shown

This request is made pursuant to the following paragraph(s) of rule 12 of the CNSC Rules of Procedure:

- Rule 12 (1) (a) the information involves national or nuclear security;
- Rule 12 (1) (b) the information is confidential information of a financial, commercial, scientific, technical, personal or other nature that is treated consistently as confidential and the person affected has not consented to the disclosure; or
- Rule 12 (1) (c) disclosure of the information is likely to endanger the life, liberty or security of a person.

Further,

1. The above-noted material should be protected for the following reasons:
   - The Hazard Screening Analysis Report is considered prescribed information under the Nuclear Safety and Control Act (NSCA), as defined in Section 21 (1) (a) and (c) of the General Nuclear Safety and Control Regulations. This document also contains confidential information as defined by the CNSC Rules of Procedure Rule 12 (1).

a. Rule 12 (1) (a) and (c) – The attached Hazard Screening Analysis Report contains nuclear security information and concerns relating to Pickering NGS, which if disclosed to the public, could endanger the life and security of the plant staff and general public.

b. Rule 12 (1) (b) – The attached Hazard Screening Analysis Report information is considered scientific and technical as it contains design, operation and maintenance information relating to Pickering NGS. Information of this nature is treated consistently as confidential. In addition, the correspondence letter contains personal information (name/phone number) which the person affected has not consented to the disclosure of.

2. I attest that the above-noted material is not available through any public sources.
3. 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.
4. 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).
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- Attachment to NK30-CORR-00531-08395: OPG Report, "Hazard Screening Analysis – Pickering B", December 9, 2021, CD# NK30-REP-03611-00008 R002.
- Non-Confidential Summary: OPG Report, "Pickering Nuclear Generating Station B Probabilistic Safety Assessment Summary Report", November 18, 2022, CD# NK30-REP-03611-00021-R002.

Authorized signature:

2024/03/28
Date

Riedewaan Bakardien,
Senior Vice President,
Pickering Nuclear Generating Station

ONTARIOPOWER GENERATION

Report

OPG Proprietary	
Document Number: **NK30-REP-03611-00021**	Usage Classification: **N/A**
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Title:
Pickering Nuclear Generating Station B Probabilistic Safety Assessment Summary Report

Pickering Nuclear Generating Station B Probabilistic Safety Assessment Summary Report

NK30-REP-03611-00021-R002

2022-11-18

Order Number: N/A
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OPG Proprietary

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Nuclear Project Number	Contract Number	Page
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NK30-REP-03611-00021 R002	ONTARIO POWER GENERATION INC.	

Title:

PICKERING NUCLEAR GENERATING STATION B PROBABILISTIC SAFETY ASSESSMENT SUMMARY REPORT

Project: **PICKERING B**

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Title:

PICKERING NUCLEAR GENERATING STATION B PROBABILISTIC SAFETY ASSESSMENT SUMMARY REPORT

Project: **PICKERING B**

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Ref. Procedure NU-907020-PRO-001

For Release Information, refer to the Document Transmittal Sheet accompanying this document.

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1	2022/11/15	Issued as “Approved for Use”.	U. Shahid	K. Joober [DL]	M. Dahmani
2	2022/11/17	Issued as “Approved for Use”. This revision addresses minor editorial comments.	U. Shahid	K. Joober [DL]	M. Dahmani

[1] Initial Rev. 000 w as issued in February 2013, and the 2017 update Rev. 001 w as issued in January 2018.

TERMS AND ABBREVIATIONS

ABFW	Auxiliary Boiler Feedwater
ACU	Air Conditioning Unit
AIM	Abnormal Incidents Manual
AOO	Anticipated Operational Occurrence
APC	Atmospheric Pressure Change
BDBA	Beyond Design Basis Accident
BDBE	Beyond Design Basis Event
BECS	Boiler Emergency Cooling System
BLEVE	Boiling Liquid Expanding Vapour Explosion
BWR	Boiling Water Reactor
CAFTA	Computer Aided Fault Tree Analysis
CANDU	CANadian Deuterium Uranium
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failures
CDFM	Conservative Deterministic Failure Margin
CEI	Containment Envelope Integrity
CET	Containment Event Tree
CLRP	Conditional Large Release Probability
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CSA	Canadian Standards Association
D_2O	Deuterium Oxide (Heavy Water)
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DSC	Dry Storage Container
ECI	Emergency Coolant Injection
EF	Enhanced Fujita
EFADS	Emergency Filtered Air Discharge System
EME	Emergency Mitigating Equipment
EPRI	Electric Power Research Institute
EPS	Emergency Power System
ERT	Emergency Response Team
ET	Event Tree
EWS	Emergency Water Supply System

TERMS AND ABBREVIATIONS

FADS	Filtered Air Discharge System
FAQ	Frequently Asked Question
FDC	Fuel Damage Category
FHA	Fire Hazard Assessment
FIF	Fire Ignition Frequency
FIS	Fixed Ignition Source
FMLA	Failure Mode and Likelihood Analysis
FSSA	Fire Safe Shutdown Analysis
FT	Fault Tree
FTREX	Fault Tree Reliability Evaluation eXpert
FW	Feedwater
GSS	Guaranteed Shutdown State
HCLPF	High Confidence of Low Probability of Failure
HEP	Human Error Probability
HGL	Hot Gas Layer
HPECI	High Pressure Emergency Coolant Injection
HRA	Human Reliability Analysis
HTS	Heat Transport System
HVAC	Heating, Ventilation and Air Conditioning
HX	Heat Exchanger
IAEA	International Atomic Energy Agency
IE	Initiating Event
IFB	Irradiated Fuel Bay
IST	Industry Standard Toolset
IVR	In-Vessel Retention
LLDS	Low Level Drained State
LOCA	Loss-of-Coolant Accident
LRF	Large Release Frequency
MAAP	Modular Accident Analysis Program
MCA	Multi-Compartment Analysis
MCR	Main Control Room
MSO	Multiple Spurious Operation
NCEI	National Centers for Environmental Information
NGS	Nuclear Generating Station

TERMS AND ABBREVIATIONS

NOAA	US National Oceanic and Atmospheric Administration
NPC	Negative Pressure Containment
NRC	Nuclear Regulatory Commission (U.S.)
NUREG	Nuclear Regulation
NWS	US National Weather Service
OP&P	Operating Policies and Principles
OPEX	Operating Experience
OPG	Ontario Power Generation
OPGSS	Over Poisoned Guaranteed Shutdown State
OSR	Operational Safety Requirements
PAU	Physical Analysis Unit
PBRA	Pickering NGS B Probabilistic Safety Assessment
PBRA-FLOOD	Pickering NGS B Internal Flooding Probabilistic Safety Assessment
PBRA-IFPSA	Pickering NGS B Internal Fires Probabilistic Safety Assessment
PBRA-L1O	Pickering NGS B Level 1 Outage Internal Events Probabilistic Safety Assessment
PBRA-L1P	Pickering NGS B Level 1 At-Power Internal Events Probabilistic Safety Assessment
PBRA-L2P	Pickering NGS B Level 2 At-Power Internal Events Probabilistic Safety Assessment
PBRA-PSA-Based SMA	Pickering NGS B PSA-Based Seismic Margin Assessment
PBRA-WIND	Pickering NGS B High Winds Probabilistic Safety Assessment
PDS	Plant Damage State
PNGS	Pickering Nuclear Generating Station
POS	Plant Operational State
PRV	Pressure Relief Valve
PSA	Probabilistic Safety Assessment
PSF	Performance Shaping Factor
PWR	Pressurized Water Reactor
RAB	Reactor Auxiliary Bay
RBGSS	Rod-Based Guaranteed Shutdown State
RC	Release Category
RCW	Recirculating Cooling Water
RE	Reference Earthquake

TERMS AND ABBREVIATIONS

RLC	Review Level Condition
RLE	Review Level Earthquake
SAMG	Severe Accident Management Guideline
SCD	Severe Core Damage
SCDF	Severe Core Damage Frequency
SCFF	Seismically-Induced Containment Failure Frequency
SDC	Shutdown Cooling
SDS1	Shutdown System 1
SDS2	Shutdown System 2
SDV	Screening Distance Value
SEL	Seismic Equipment List
SMA	Seismic Margin Assessment
SPSA	Seismic Probabilistic Safety Assessment
SRV	Steam Reject Valve
SSC	Systems Structures and Components
SSLB	Secondary Side Line Break
THERP	Technique for Human Error Rate Prediction
UECC	Unit Emergency Control Centre
UFDS	Used Fuel Dry Storage
UPS	Uninterruptable Power Supply
USA	United States of America
VCE	Vapour Cloud Explosion

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EXECUTIVE SUMMARY

The objective of Probabilistic Safety Assessment (PSA) at Ontario Power Generation (OPG) Nuclear is to provide an integrated review of the adequacy of the safety of the current station design and operation for each nuclear power station. The station PSAs are required to comply with the Canadian Nuclear Safety Commission (CNSC) Regulatory Document REGDOC-2.4.2 [R1].

A nuclear PSA identifies the various sequences that lead to radioactive releases, assigns them to different categories of consequences, and calculates their frequencies of occurrence. Additionally, the PSA is used to identify the sources of risk and assess the magnitude of radiological risks to the public from potential accidents due to operation of nuclear reactors while at power as well as during outages. The PSA is a comprehensive model of the plant that incorporates knowledge about plant design, operation, maintenance, testing and response to abnormal events. To the extent possible, the PSA is intended to be a realistic model of the plant.

The Pickering Nuclear Generating Station B (PNGS-B) PSA followed a quality assurance plan consistent with Canadian Standards Association (CSA) standard CSA N286-12 [R2] and CSA N299.1 [R3]. The PNGS-B PSA used computer programs consistent with CSA standard CSA N286.7-16 [R4]. The PNGS-B PSA is also in line with CSA standard N290.17 [R5].

The PSA is prepared following methodologies consistent with best industry practice. The OPG PSA Methodologies have been accepted by the CNSC under compliance with REGDOC-2.4.2 [R1].

The baseline PNGS-B probabilistic safety assessments are documented in several reports:

- A hazard screening assessment identifies the hazards that require assessment in a PSA model.
- The Level-1 and Level-2 internal events At-Power PSA assesses the risk of severe core damage and radioactive releases from internal events occurring while the reactor is at power; i.e., it considers the challenges to reactor core cooling from accident sequences covering Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs) including Severe Accidents while the reactor is at full power.
- The internal events outage PSA assesses the risk of severe core damage from internal events occurring while the reactor is in the Guaranteed Shutdown State (GSS); i.e., it considers the challenges to reactor core cooling from accident sequences during unit outages, including loss of shutdown heat sinks. It also provides an estimate of the risk of large release in GSS.
- The PSA-based Seismic Margin Assessment (SMA) estimates the risk of severe core damage and large release from seismic events occurring while the reactor is at full power and provides an estimate of the containment failure frequency as a result of seismic events.
- The internal fire PSA assesses the risk of severe core damage and large release from internal fires occurring while the reactor is at full power.
- The internal flooding PSA assesses the risk of severe core damage from internal floods occurring while the reactor is at full power, and a bounding estimate of large release because of internal floods.
- The high wind PSA assesses the risk of severe core damage from high winds occurring while the reactor is at full power, and an estimate of large release.
- The non-reactor sources PSA assesses the risk of radioactive releases from sources other than the reactor core.

The completion of the PNGS-B PSA demonstrated that for each hazard OPG's safety goals are met for Severe Core Damage Frequency (SCDF) and Large Release Frequency (LRF).

1. INTRODUCTION

The objective of Probabilistic Safety Assessment (PSA) at Ontario Power Generation (OPG) Nuclear is to provide an integrated review of the adequacy of the safety of the current station design and operation for each nuclear power station. The station PSAs are required to comply with the Canadian Nuclear Safety Commission (CNSC) Regulatory Document REGDOC-2.4.2 [R1].

A nuclear PSA identifies the various sequences that lead to radioactive releases, assigns them to different categories of consequences, and calculates their frequencies of occurrence. Additionally, the PSA is used to identify the sources of risk and assess the magnitude of radiological risks to the public from potential accidents due to operation of nuclear reactors while at power as well as during outages. The PSA is a comprehensive model of the plant that incorporates knowledge about plant design, operation, maintenance, testing and response to abnormal events. To the extent possible, the PSA is intended to be a realistic model of the plant.

The PSA for the identified hazards for Pickering Nuclear Generating Station B (PNGS-B), commonly referred to as PBRA, provides an estimate of the station risk in its current configuration. The PSA reflects the current station design and operation, is consistent with the OPG PSA methodology, and is consistent with best industry practices. The OPG PSA Methodologies have been accepted by the CNSC under REGDOC-2.4.2 [R1]. A separate hazard screening assessment for internal and external events has been completed to confirm that no other identified hazards require detailed assessment in a PSA.

The PNGS-B PSA followed a quality assurance plan consistent with CSA standard CSA N286-12 [R2] and CSA N299.1 [R3]. The PSA used computer programs consistent with CSA standard CSA N286.7-16 [R4]. The PNGS-B PSA is also in line with CSA standard N290.17 [R5].

OPG has safety goals for Severe Core Damage Frequency[2] (SCDF) and Large Release Frequency[3] (LRF), as shown in Table 1. The intent of these goals is to ensure that the radiological risks arising from nuclear accidents associated with the operation of OPG's nuclear power reactors are low in comparison to risks to which the public is normally exposed. The baseline PBRA studies show that the overall risk from the operation of PNGS-B is below the safety goals.

The first PBRA studies for S-294 compliance were completed in 2012 and the previous update was completed in 2017. All PBRA studies are updated in 2022 as part of the regular update cycle under REGDOC-2.4.2 compliance. The updates included:

- Station design, operation and analysis information up to the study freeze date of December 31, 2020;
- Several model and documentation enhancements;
- Event tree and fault tree modelling updates to reflect recent safety analysis, as well as PNGS-B design and operation; and
- The credit of Severe Accident Management Guidelines (SAMG) in the Level 2 PSA including Phase II emergency mitigating equipment (EME).

This report summarizes the probabilistic safety assessments of the PNGS-B and compares the results with OPG's safety goals, as shown in Table 1.

[2] Severe Core Damage is the loss of core structural integrity.
[3] Large Release is a release greater than 1E14 Bq of Cs-137.

1.1 Objectives

The principal objectives of the PNGS-B PSA Studies are:

1. To provide an integrated review of the adequacy of the safety of the current station design and operation;
2. To prepare a risk model in a form that can be used, in conjunction with ancillary application tools, to assist the safety-related decision making process, and
3. To assess risk results and ensure that they are acceptably low.

1.2 Scope

The baseline PNGS-B PSAs are addressed in ten separate assessments – one hazard screening, one non-reactor source and eight PSA models, as follows:

1. A hazard screening assessment for internal and external events, which identifies the hazards that require further analysis in a PSA.
2. A Pickering NGS B Level-1 Internal Events At-Power PSA (PBRA-L1P), which studies the risk of severe core damage from events occurring within the station (e.g., loss of coolant accidents, steam line breaks) while the reactor is at full power; i.e., it considers the challenges to reactor core cooling from accident sequences covering Anticipated Operational Occurrences (AOOs), DBAs, and BDBAs including Severe Accidents.
3. A Pickering NGS B Level-1 Internal Events Outage PSA (PBRA-L1O), which studies the risk of severe core damage from internal events occurring at the station while the reactor is in a GSS. The outage PSA studies severe core damage due to failure to remove decay heat produced during unit outages, including loss of shutdown heat sinks.
4. A Pickering NGS B Level-2 Internal Events At-Power PSA (PBRA-L2P), which studies the frequency and composition of releases to the environment from severe core damage occurring due to events occurring within the station (e.g., loss of coolant accidents, steam line breaks) while the reactor is at full power. This PSA is the extension of the Level-1 PSA (i.e., PBRA-L1P) described in item 2.
5. A Pickering NGS B Level-2 Internal Events Outage PSA is a reduced scope Level 2 outage analysis based on modeling of accident progression and source term estimation and provides an estimation of the large release frequency.
6. A PSA-Based Seismic Margin Assessment (PSA-based SMA[4]), which studies the risk of severe core damage and large release from seismic events (i.e., earthquakes).
7. A Pickering NGS B Internal Fires PSA (PBRA-IFPSA), which studies the risk of severe core damage and large release as a result of internal fire events (e.g., fires caused by failures in station electrical equipment) occurring while the reactor is at full power.
8. An internal flooding PSA (PBRA-FLOOD), which studies the risk of severe core damage and provides a bounding estimate of LRF from floods originating inside the station (i.e., pipe breaks of plant systems) occurring while the reactor is at full power.
9. A high wind PSA (PBRA-WIND), which studies the risk of severe core damage and provides an estimate of LRF from high wind events (e.g., severe thunderstorms, tornadoes) occurring while the reactor is at full power.

[4] PSA-based SMA is also referred to as PBRA-SEISMIC and w ould be used interchangeably in this report.

10. A non-reactor sources PSA which assesses the risk of releases to the environment from non-reactor sources of radioactivity.

The PNGS-B PSAs reports, mentioned in bullet 2 to 10 above, do not cover the following potential sources of risk:

- Hazards from chemical materials used and stored at the plant;
- Other external Initiating Events (IEs) such as external floods, airplane crashes, train derailment, etc.; and
- Other internal IEs such as turbine missiles.

These types of hazards are instead addressed through other screening or deterministic hazard studies, see Section 4. Consistent with industry practice, wilful acts (e.g., sabotage) are not modelled in the OPG PSAs.

The response of all PNGS-B units to various IEs is essentially identical, and it is generally only necessary to model a single unit, with this unit considered representative of all other units. Unit 5 was selected as the reference analysis unit for PNGS-B. Design differences between the four units# 5, 6, 7 and 8 were not incorporated in the reference model, as they are not expected to be significant in terms of risk.

1.3 Organization of Summary Report

In addition to the general information presented in this introductory section, the Summary Report provides the following:

(a) A short description of the PNGS-B station and units (Section 2);

(b) An overview of PSA methods (Section 3);

(c) An overview of the hazard screening method and the internal/external hazard screening assessment (Section 4);

(d) An overview of the methods used for Level 1 PSA (Section 5) and Level 2 PSA (Section 6); and

(e) A discussion of the main results of the PNGS-B PSAs studies (Section 7).

A list of the abbreviations and acronyms used in this summary report are presented at the start of the report.

2. PLANT DESCRIPTION

The following sections provide a short description of the Pickering site and plant.

2.1 Site Arrangement

The PNGS-B comprises four CANadian Deuterium Uranium (CANDU®)[5] pressurized heavy water nuclear reactors, four turbine generators and their associated equipment, services and facilities. The layout of the eight-unit[6] Pickering site (PNGS-A and PNGS-B) is shown in Figure 1. An aerial photograph of the site is shown in Figure 2.

The design net electrical output of each unit is 516 MW(e) at an 85 percent power factor which yields a total station net output of 2064 MW(e). Power is produced at 24 kV and delivered at 230 kV and 60 Hz to the Southern Ontario grid. The station is designed for base-load operation.

Each unit comprises a power source capable of operating independently of the other units with reliance on certain common services. The power generating equipment of each unit is a conventional steam-driven turbine generator. The associated heat source is a heavy water moderated, pressurized heavy water cooled, natural uranium dioxide fuelled, horizontal pressure tube reactor. This type of nuclear steam supply is used in all nuclear power stations built in the province of Ontario.

2.2 Buildings and Structures

The principal structures at the PNGS-B site are as follows:

(a) Four reactor buildings;

(b) A reactor auxiliary bay;

(c) A powerhouse which includes the turbine hall and turbine auxiliary bay running the full length of the station;

(d) A tempering water pumphouse;

(e) A screenhouse;

(f) Six standby generator enclosures;

(g) A pressure relief duct;

(h) A High Pressure Emergency Coolant Injection (HPECI) pumphouse;

(i) A HPECI water storage tank; and

(j) An emergency water/power supply building.

The administration and service buildings, the East Annex, the vacuum building, EME building, the HPECI structures, an Annex building, the Pickering Waste Management Facility, an addition to the Units 1 to 4 service wing, and a heavy water upgrading building serve the entire station.

The containment boundary is formed by the reactor buildings, the pressure relief duct, the vacuum ducts and the vacuum building. Each reactor building is a reinforced concrete structure with cylindrical walls

[5] "CANDU" is a registered trade-mark of Atomic Energy of Canada Limited.
[6] Currently Units 1 and 4 at PNGS-A are operating and Units 2 and 3 are in safe storage. Both Units 2 and 3 have been de-fueled and the D_2O in both the moderator and the Heat Transport System (HTS) drained completely.

and an elliptical dome. The vacuum building is also a reinforced-concrete structure with a cylindrical wall and a flat roof. A tank in the top of the vacuum building contains water for the dousing system. A reinforced concrete ring around the vacuum building, outside the perimeter wall near the base, provides additional pressure retaining capability. The pressure relief duct, also a reinforced concrete structure, is rectangular in section and is linked to the vacuum building by steel vacuum ducts 1.8 m in diameter.

Unit Emergency Control Centres (UECCs), one for each unit, are located under the pressure relief duct.

The Reactor Auxiliary Bay (RAB) runs the full length of the station and is a conventional four-story steel frame building fitted around the northern halves of the four reactor buildings. In addition to the Main Control Room (MCR) and Irradiated Fuel Bay (IFB), the RAB houses some reactor auxiliary systems.

The service wing extension located at the eastern end of the PNGS-A station, i.e., in the center of the eight units, provides additional space for waste management, laboratories, stores, locker and change facilities, maintenance shops, fuelling machine dismantling facilities and offices.

2.3 Reactor

The reactor consists of an array of tubes in a cylindrical, heavy water filled structure, referred to as the calandria assembly. Inside the calandria are the fuel channel assemblies, which contain the fuel, as well as reactivity monitoring control units. The whole assembly is enclosed in the calandria vault, a concrete vault filled with light water.

The ends of the calandria assembly are called the end shields and are located in openings in the calandria vault wall. The end shields form part of the vault enclosure. The end shields, in conjunction with the shield plugs in the fuel channels, provide sufficient shielding against radiation from the reactor and its fuel, to permit personnel access to the fuelling machine areas when the reactor is shutdown. An arrangement of embedded pipes carries light water to provide cooling for the concrete of the vault. A typical PNGS-B reactor assembly is illustrated in Figure 3.

2.4 Heat Transport System

The Heat Transport System (HTS) consists of two identical loops, one for the north half of the reactor and one for the south half. Each loop consists of fuel channels filled with natural uranium fuel bundles surrounded by pressurized heavy water, boilers, circulation pumps and associated piping and valves. The coolant in the fuel channels removes the heat generated by the fuel. During normal operation the heat from the fuel is generated via the nuclear fission, following shutdown heat is generated from the fuel via fission product decay. The circulating coolant then transports this heat to the boilers. This is the primary heat sink for the reactor, thus the system is often referred to as the primary heat transport system.

The heat transport system interfaces with a number of systems: the shutdown cooling system, which removes decay heat when the reactor is shutdown; the feed and bleed system, which provides pressure and inventory control for the coolant; the D_2O recovery system, which recovers lost heavy water (D_2O) from leaks; and the Emergency Coolant Injection System (ECIS), which adds light water after the occurrence of a loss of coolant accident beyond the capacity of the D_2O recovery system.

2.5 Moderator System

During normal plant operation the moderator system is used to slow the neutrons produced by the reactor to maintain a critical fission reaction. Heat is generated in the moderator by the neutrons as they slowdown, and energy is transferred to the moderator from the calandria tubes, shell, tube sheets and, reactivity mechanisms. During normal operation a small fraction of the heat produced by the fuel is transferred to the moderator. The moderator system includes heat exchangers to remove this heat. After an accident, the moderator can be used as an additional heat sink to remove decay heat from the reactor. This additional heat sink is an important, unique feature of the CANDU reactor design.

2.6 Steam and Feedwater System

As described above, the main role of the primary heat transport system is to transport the heat generated in the fuel channels to the boilers. The role of the boilers, then, is to transfer this heat and boil the light water on the secondary side of the boilers. The steam generated in the boilers is then used to drive the turbine generators to convert the thermal energy to electrical power. During this process, the boiling water condenses. The condensate is returned to the feedwater system and eventually returned to the boilers to continue the process.

2.7 Boiler Emergency Cooling System

The Boiler Emergency Cooling System (BECS) is designed to provide a short term, high pressure supply of cooling water to the boilers in the event of a total loss of feedwater. This system is designed to be used until an alternative heat sink can be placed in service.

2.8 Steam Relief System

The steam relief system protects the boilers from overpressure. The system is also used for rapid cooling of the primary heat transport system when needed.

2.9 Shutdown Cooling System

The Shutdown Cooling (SDC) system provides an alternative method to remove decay heat from the primary heat transport coolant when the reactor is shutdown. The system consists of a set of pumps and Heat Exchangers (HXs) that are normally isolated from the primary heat transport circuit but can be connected when needed. The SDC system has a much smaller capacity to remove heat than the main boilers, as the reactor produces significantly less heat in the shutdown state. The SDC system is the preferred heat sink when the unit is in the GSS.

2.10 Reactor Regulating System

The reactor regulating system is designed to control the power of the reactor during normal operation. The reactor regulating system uses several control mechanisms including the liquid zone control, and the insertion of neutron absorbing rods, to regulate reactor power.

2.11 Powerhouse Emergency Venting System

The Powerhouse Emergency Venting System (PEVS) is used to mitigate harsh environments caused by high temperature or high humidity in the powerhouse, which contains the turbines and other equipment, due to steam line breaks.

2.12 Special Safety Systems

Four special safety systems are incorporated into the plant design to limit radioactive releases to the public following any abnormal event:

(a) Shutdown System No. 1 (SDS1);

(b) Shutdown System No. 2 (SDS2);

(c) Emergency Coolant Injection (ECI) System; and

(d) Negative Pressure Containment (NPC) System.

2.12.1 Shutdown Systems

The reactor is equipped with two separate and isolated shutdown systems. SDS1 is a rod based system that drops neutron absorbing rods into the reactor core. SDS2 is a liquid injection system that adds a neutron absorbing fluid into the moderator. The two shutdown systems are part of the four special safety systems.

2.12.2 Emergency Coolant Injection System

The ECIS automatically provides make-up cooling water to the heat transport system following a postulated Loss-Of-Coolant Accident (LOCA). The ECIS does not operate during normal plant operation but is in poised standby mode. The PNGS-B ECI system includes an initial high pressure injection from the HPECI system which is shared with PNGS-A and a low pressure recovery injection which is common to paired units (5/6 or 7/8) in PNGS-B. This system is one of the four special safety systems.

2.12.3 Negative Pressure Containment System

The NPC system provides a physical barrier designed to limit the release of radioactivity to the environment which might result from a process or system failure. The containment system is a reinforced concrete envelope around the nuclear components of the reactor cooling system, with provisions for controlling and maintaining a negative pressure within the envelope before and after accidents.

The NPC system includes a number of sub-systems required for providing normal and post-accident functions such as reactor building cooling, pressure suppression, control of hydrogen, and air discharge filtration. This system is one of the four special safety systems.

2.12.4 Support Systems

Support systems are considered in the probabilistic safety assessments as they provide common services to the systems described above. Failure of the support systems can result in failure of the mitigating systems credited to remove heat after an IE. The following systems are modelled as support systems in the PSA.

2.12.4.1 Electrical Power Systems

(a) Normal Power Supply

The electrical systems at Pickering B are organized into four classes: Class IV power is the main site electrical power supplied from a combination of the provincial electrical grid and the station generating unit transformers; Class III power is the back-up supply to Class IV and includes three standby generators for each paired unit (5/6 or 7/8); Class II (AC power) is primarily used to supply control and monitoring systems; Class I (DC power) is primarily used to supply motive power to electrical breakers. Class II and Class I both have battery backup supplies.

(b) Emergency Power Supply

The Emergency Power Supply (EPS) is a system qualified to withstand seismic events and is completely independent from the station normal Class IV and Class III power sources. The purpose of the EPS is to provide power supply to essential station safety functions (reactor shutdown, removal of decay heat, monitoring of post-accident events) in the event of a total loss of normal station power supplies.

2.12.4.2 Service Water Systems

The service water systems provide cooling water for various loads. The service water systems for PNGS-B consist of:

(a) Service Water System:

The service water system provides cooling water from Lake Ontario for various loads. Service water is drawn from Lake Ontario through an open canal bounded by two rock filled groynes extending into the lake. The water is drawn from the canal to an open forebay, then through a common screen house into an enclosed concrete duct or intake channel. The service water system is divided into two sub-systems referred to as low and high pressure service water. The high pressure service water system draws its water from the low pressure service water pumps discharge and provides a pressure boost via a second set of pumps to deliver service water at higher elevations in the plant. Service water is used once and returned to the lake.

(b) Recirculated Cooling Water System:

The Recirculated Cooling Water (RCW) system provides clean, demineralized cooling water to equipment that might become contaminated or plugged if supplied by lake water via the service water system. The RCW system recirculates water via a set of pumps and cools the water via a set of heat exchangers. The normal service water system is used on the secondary side of the RCW heat exchangers for cooling purposes.

(c) Emergency Water System:

The Emergency Water System (EWS) is a redundant water supply, designed to provide cooling water in the event that other sources of water fail. The emergency water system has a separate screen house and pump house to obtain water from the common Pickering forebay.

2.12.4.3 Instrument Air Systems

The instrument air supply is a support system providing compressed air. This compressed air is used for various plant activities including operating valves, starting motors, and inflating airlock seals.

2.12.4.4 Cooling and Ventilation Systems

The cooling and ventilation systems provide heating and cooling to the station buildings. The cooling and ventilation systems support equipment operation in various locations such as Class I and II electrical room, reactor building moderator room, EPS electrical room, and the standby generator rooms.

2.12.4.5 Emergency Mitigating Equipment

As a result of Fukushima, OPG has implemented Emergency Mitigating Equipment (EME) for Pickering B NGS. The EME was designed to cope with a total loss of heat sink caused by a complete loss of all AC power.

EME response is provided in two phases:

- The PNGS 'B' Phase 1 EME includes portable equipment (pumps and generators) that can be deployed in an event to restore power to critical loads and provide emergency water make-up to critical demands including the Boilers, HTS, Moderator, Shield Tanks, and IFB.

- Phase 2 EME is directed at longer term actions and deals with successful prevention of severe accident conditions by making increased use of new/temporary water and power supplies to continue recovery of affected units to a stable state.

Phase 1 EME consists of:

(a) Four portable 15 kW diesel generators available to provide emergency power to PNGS NGS ‘B’. The generators are normally located in the EME storage facility at the east end of the Pickering site. There are also four Uninterruptable 120 V Power Supplies (UPS) that are stored in UECC and used to provide power to instrumentation until the power is restored by the diesel generators.

(b) Six portable diesel pumps to provide emergency water make-up. The HL260M pump is deployed to the west side of the PNGS NGS ‘B’ Screenhouse to provide primary make-up to the HTS, Boilers, HPECI storage tank, and vacuum pump seals, and to provide contingency make-up to the moderator if required. The four HL5M pumps are deployed to each unit Reactor Auxiliary Bay (RAB) to provide primary make-up to the moderator, and to the shield tank (if required), and to provide contingency make-up to the boilers and HTS if required. The HL160M pump is deployed to the east side of the PNGS NGS ‘B’ Screenhouse to provide primary make-up to the IFB, and to provide contingency make-up to the HTS and moderator if required.

Phase 2 EME consists of:

(a) Two 1.14 MW generators that are used along with portable switchgear to repower the Emergency Power Supply (EPS) System. EPS can then be used to supply power to:

   (1) EPS DC power supply.

   (2) UECC heating, lighting and instrumentation.

   (3) EWS main pump, recovery pump, strainers and travelling screens.

   (4) Boiler room and FM Vault ACU fans.

   (5) FADS.

   (6) ECI Recovery pumps and sump pumps.

2.13 Two-Group Separation

The PNGS-B design uses group separation to minimize the possible consequences of events that could cause widespread damage, and to provide defence in depth. Each group contains equipment to shut down the reactor, remove decay heat, and monitor the reactor status. The Group 1 and Group 2 systems are physically separated.

The following systems are Group 1:

- SDS1: Shutdown System No. 1
- ECI: High Pressure Emergency Coolant Injection
- SDC: Shutdown Cooling
- FW: Main Boiler Feedwater
- ABFW: Auxiliary Boiler Feedwater
- Class IV, III, II, I Electrical Power Distribution
- Instrument air (normal distribution)

- Service Water

The Group 1 control functions are performed from the MCR.

The following systems are Group 2:

- SDS2: Shutdown System No. 2
- NPC: Negative Pressure Containment
- EPS: Emergency Power System
- EWS: Emergency Water Supply System
- BECS: Boiler Emergency Cooling System
- SRVs: Steam Reject Valves
- ECI Recovery: Emergency Coolant Injection Recovery System
- EFADS: Emergency Filtered Air Discharge System

The Group 2 systems are seismically qualified to withstand a Design Basis Earthquake (DBE). The Group 2 control functions are performed from UECC.

3. OVERVIEW OF PSA METHODS

PSA is based on the idea that the product of the frequency of occurrence of an event and the consequence of the event represents a useful and meaningful quantity. This product is defined to be the risk from the event and is expressed in units of consequence per unit of time. For example, consider a residential sump pump that fails on average once every four years. If the consequence of the pump failing is $1000 in property damage, then the average risk from failure of the pump is $250 per year.

Risk provides a means of quantifying the degree of safety inherent in a potentially hazardous activity as well as a common basis for comparing the relative safety of dissimilar types of activities and industrial processes. One of the principles of the PSA process is that the larger the numerical value of risk for a particular event or combination of events, the more important the event is to safety. Thus, measures to reduce calculated risk improve the level of safety. PSA represents the process by which risk is quantified, leading to the identification of the dominant contributors to risk. If necessary, the dominant contributors can be used to create strategies to reduce risk and improve safety.

For a Nuclear Generating Station (NGS), the events studied are those leading to damage to fuel both in the core and out of core or releases of radioisotopes into the environment. Consistent with the requirements of the CNSC REGDOC-2.4.2 [R1] standard, OPG has completed hazard screening, Level 1 and Level 2 PSA to assess the risk from PNGS-B:

- A hazard screening assessment was performed to confirm which hazards can be screened out from probabilistic safety assessment, and identify which hazards need to be assessed by a PSA. This includes non-reactor sources as well;
- Level 1 of the PSA assesses the frequency of varying degrees of fuel failures, which lead to release of radioactivity into containment; and
- Level 2 of the PSA assesses the frequency and magnitude of the release of this radioactivity from containment to the outside environment.

OPG's safety goals in Table 1 for PSA correspond to the Level 1 and Level 2 PSA results.

Level 1 PSAs have been prepared for full reactor power operation for the following types of IEs based on the hazard screening results:

- Internal IEs (e.g., steam line break, loss of coolant accidents);
- Seismic events;
- Internal fire (fires initiated by in plant sources, e.g., electrical equipment);
- Internal flooding (floods originating from water sources internal to the plant); and
- High winds (including both straight line winds and tornadoes).

An assessment of risk while a single unit is in GSS was prepared for internal initiating events. Outage PSAs have not been prepared for seismic events, high winds, fire, and internal flooding for the reasons described below:

- An outage seismic PSA was not performed as the risk from a seismic event while a unit is shutdown is acceptably low or is bounded by the seismic risk for an at-power unit. The key factors supporting this assertion are that:
    1. A seismic event and failure to remain shutdown is not a significant contributor to risk.

2. Given the above, seismic risk is dominated by sequences involving the failure of all heat sinks. A seismic event will have a similar effect on the heat sinks for shutdown and high power units: the in-service heat sinks and the Group 1 emergency heat sinks are expected to fail, but operation of the Group 2 emergency heat sinks will be largely unaffected by a single unit outage. The SCDF over the range of events with a 1E-04 occ./year return frequency remained essentially unchanged at 1.3E-07 /year due to crediting of non-seismically qualified Structures, Systems or Components (SSCs) for accident mitigation under low-intensity earthquakes.
3. Initial reactor power is at least two orders of magnitude less for a shutdown unit than for an at-power unit. Therefore, the fuel temperature will be lower, accident progression will be slower, and the amount of energy deposited into containment will be lower. This reduces the potential for consequential challenge to containment integrity from a severe accident in a single shutdown unit.
4. The inventory of radioactive material available for release to the environment is less for a shutdown unit due to the decay of short-lived isotopes.
5. The operation of key containment systems is largely unaffected if a single unit is shutdown.

On average, a unit is shutdown for a planned outage for approximately 10% of the operating cycle. Therefore, the exposure to low frequency events such as seismic events is much lower for a shutdown unit than for an at-power unit. Risk management programs are adequate to control the risk from a seismic event while a unit is shutdown. Thus, the risk is smaller for the unit in outage.

- An internal fire outage PSA was not performed as the risk from an internal fire while a unit is shutdown is either acceptably low or is bounded by the risk from internal fires for an at-power unit. The key factors supporting this assertion are that:

1. An internal fire and failure to remain shutdown is not a significant contributor to risk.
2. Given the above, the fire risk is dominated by sequences involving the failure of all heat sinks. The low SCDF for an at-power unit can be attributed to a combination of:
   - Low Initiating Event frequency;
   - Reliable fire detection and suppression systems; and
   - Physical separation between Group 1 and Group 2 systems.

   The SCDF for the 2022 Pickering B Internal Fire At-Power PSA is 7.75E-07 /year.
3. Initial reactor power is at least two orders of magnitude less for a shutdown unit than for an at-power unit. Therefore, the fuel temperature will be lower, accident progression will be slower, and the amount of energy deposited into containment will be lower. This reduces the potential for consequential challenge to containment integrity from a severe accident in a single shutdown unit.
4. The inventory of radioactive material available for release to the environment is less for a shutdown unit due to the decay of short-lived isotopes. The operation of key containment systems is largely unaffected if a single unit is shutdown.

On average, a unit is shutdown for a planned outage for approximately 10% of the operating cycle. Therefore, the exposure to low frequency events such as fires is much lower for a shutdown unit than for an at-power unit. Risk management programs are adequate to control the risk from a fire while a unit is shutdown.

- An outage internal flood PSA was not performed as the risk from an internal flood while a unit is shutdown is either acceptably low or is bounded by the flood risk for an at-power unit. The key factors supporting this assertion are that:
  1. An internal flood and failure to remain shutdown is not a significant contributor to risk.
  2. Given the above, the risk from internal floods is dominated by sequences involving the failure of all heat sinks. The SCDF due to internal floods for an at-power unit is very low (2.0E-07 /year). The low SCDF for an at-power unit can be attributed to a combination of:
     - Low Initiating Event frequency; and
     - Physical separation between Group 1 and Group 2 systems.
  3. Initial reactor power is at least two orders of magnitude less for a shutdown unit than for an at-power unit. Therefore, fuel temperatures will be lower, accident progression will be slower, and the amount of energy deposited into containment will be lower. This reduces the potential for consequential challenge to containment integrity from a severe accident in a single shutdown unit.
  4. The inventory of radioactive material available for release to the environment is less for a shutdown unit due to the decay of short-lived isotopes. The operation of key containment systems is largely unaffected if a single unit is shutdown.

  On average, a unit is shutdown for a planned outage for approximately 10% of the operating cycle. Therefore, the exposure to low frequency events such as floods is much lower for a shutdown unit than for an at-power unit. Risk management programs are adequate to control the risk from a flood while a unit is shutdown.

- An outage high wind PSA was not performed as the risk from a high wind while a unit is shutdown is low and it is bounded by the risk from high winds while a unit is at high power. The key factors for this assertion are that:
  1. A high wind event and failure to remain shutdown is not a significant contributor to risk.
  2. Given the above, the risk from high winds is dominated by sequences involving the failure of all heat sinks. Results from the Level 1 High Wind PSA for high power units indicate that risk is dominated by straight line winds. Straight line winds are conservatively assumed to be perfectly correlated, i.e., they affect all four units simultaneously. Therefore, a high wind will have a similar effect upon the in-service heat sink and the emergency heat sinks of both shutdown and at-power units. The at-power SCDF over the range of events with a 1E-04 occ./year return frequency, is 9.9E-06 /year.
  3. Containment integrity may be challenged by:
     - The energy released from the reactors during a severe accident;
     - Wind induced failures, including failure from missile strikes; and
     - Random containment failures either prior to the severe accident or during the post accident mission.

     The above challenges to containment integrity are either unaffected if a single unit is shutdown or bounded by the challenges from the three high power units.
  4. The inventory of radioactive material available for release to the environment is less for a shutdown unit due to decay of short-lived isotopes.

On average, a unit is shutdown for a planned outage for approximately 10% of the operating cycle. Therefore, the exposure to low frequency events such as high winds is much lower for a shutdown unit than for an at-power unit. Risk management programs are adequate to control the risk from a high winds event while a unit is shutdown.

The full scope Level 2 PSA has been prepared for at-power internal events. Reduced scope at-power Level 2 assessments have been prepared for seismic events, outage internal events, internal fires, internal flooding and high winds as follows:

- The Level 2 assessment for seismic events considers the likelihood of consequential failure of containment due to an earthquake, and then provides a bounding assessment of LRF due to seismic failure modes of containment following severe core damage caused by a seismic event. It is conservatively considered that the LRF estimate is equal to the SCDF estimate.
- The Level 2 assessment of outage internal events reviews the potential for unique containment challenges or bypass pathways in the outage state caused by severe core damage from an internal IE occurring while the reactor is in the GSS and provides an estimated LRF.
- Level 2 assessment for fire events is based on a bounding estimate of LRF.
- Level 2 assessment for internal flooding is based on a bounding estimate of LRF.
- Level 2 assessment for high winds is based on a bounding estimate of LRF.

Additionally, bounding assessments for non-reactor sources (IFB and used fuel dry storage) were performed.

In the following sections, the methods used for hazard screening, Level 1 and Level 2 PSAs and non-reactor sources PSA are described.

4. HAZARD SCREENING METHODS

A hazard is an event or natural phenomenon that has the potential to pose some risk to the facility. Hazards can be divided into two groups: external and internal. External hazards include events such as flooding and fires external to the plant, tornadoes, earthquakes, and aircraft crashes. Internal hazards include events such as equipment failures, operator induced events, flooding and fires internal to the plant. The purpose of hazard screening analysis is to determine which hazards can be screened out from probabilistic safety assessment, and identify which hazards need to be assessed by a PSA. Both reactor sources and non-reactor source were considered.

The Pickering B PSA addresses two non-reactor sources:

1. fuel in wet storage in the Irradiated Fuel Bays (IFBs); and
2. fuel in dry storage in the Used Fuel Dry Storage (UFDS) facility.

These sources were assessed because, following an initiating event, they are the only non-reactor sources that have the potential to release more Cs-137 than the threshold for a large release (1E14 Bq).

4.1 External Hazard Screening for Reactor Sources

External hazards are defined as hazards that are initiated outside the OPG exclusion zone or are hazards that are outside the plant's direct control. These hazards could be in the form of natural hazards (ice-storms, flood, etc.) or man-made hazards (chlorine leak from a rail-car derailment, aircraft crash, etc.).

4.1.1 Overview of External Hazards Screening Method

The external hazards screening method involves three main steps:

1. Identify all the external hazards applicable to the site.
2. Determine consequences of hazards and accident scenarios. Screen-out events qualitatively, based on the consequence of events.
3. Determine likelihood of event occurring. Screen-out events quantitatively, based on the likelihood of event occurring, or on the likelihood of the event leading to severe accident (e.g., SCDF or CCDP).

The hazard screening flow diagram of steps is shown in Figure 4. A generic list of the hazards is developed based on a literature review and is reviewed and rationalized by a group of risk assessment experts to come up with a refined master list. Once the hazards are identified, the screening process begins with qualitative assessment of hazards impact and consequences of events, followed by quantitative assessments.

The qualitative screening steps QL1 to QL7 discussed below are the criteria for qualitative screening:

[QL1] The first qualitative criterion is if the event is of equal or lesser damage potential than similar events for which the plant has been designed.

After the hazards are identified and determined their impact could be beyond the design basis of the plant, the scenarios need to be defined for each hazard, and it needs to be determined how far from the station they take place and how they can potentially impact the plant's operation.

[QL2] For each scenario, it has to be determined if there are other bounding events. If the hazard imposes lower risk (frequency and consequence) than another hazard, it can be screened out.

[QL3] Once the hazard distance is determined, it can be assessed whether it can be screened based on the distance from the plant.

For screening purposes, a Screening Distance Value (SDV) is defined by the International Atomic Energy Association (IAEA), which is the distance from a facility beyond which, potential sources of a particular type of external event can be ignored. The SDV is different for different hazards. Generally, the safe distance is a distance beyond which a hazard source is too weak to impact nuclear safety.

[QL4] If the event is included in the definition of another event or bounded by other event, it can be screened out from any further assessment.

[QL5] Events that progress slowly and it can be demonstrated that there is sufficient time to eliminate the source of the threat or provide an adequate response, can be screened out.

[QL6] If the event does not cause an IE (or the need to shutdown) and does not result in loss of a safety system, it can be screened out.

[QL7] If the event does not result in actuation of a front-line system (i.e., a system that directly performs accident mitigating functions), then it is not necessary to evaluate the consequences of the hazard, and it can be screened out.

The first step is to identify the applicable hazards. The identification process includes an initial screening to remove initiators that clearly are not credible events for sites in question and that do not require development of methodology as a result.

The applicable hazards are then screened first by qualitative criteria, and if the hazard cannot be screened qualitatively, it is screened by the quantitative frequency-based criteria as shown in Table 2. Hazards that cannot be screened by either the qualitative or quantitative criteria require additional analysis in a separate assessment.

4.1.2 Human-Induced External Hazards

All human-induced (man-made) external hazards identified for PNGS-B were reviewed and examined against the methodology described in Section 4.1.1. All human-induced external hazards were screened out, and do not require a PSA. A list of the human-induced hazards assessed is presented in Table 3.

4.1.3 Natural External Hazards

A Review Level Condition (RLC) needs to be defined for each natural hazard during the screening assessment and is used to assess the impact on the nuclear safety. The RLCs are normally defined for a beyond design basis event, as the natural hazards within the design basis should not have any significant impact on the plant's operation and safety. The concept of RLC implies a particular level of hazard which challenges the Systems, Structures and Components (SSCs) on the site. Selection of RLC is based on:

- Canadian and International regulations and standards; and
- Information on credible hazards at the plant site;
- Or alternatively, the RLC can be established for the corresponding screening frequency.

PSA screening analysis for natural external hazards was conducted in accordance with the methodology described in Section 4.1.1. A set of RLCs were defined and used in the screening analysis. Among the twenty-six natural external hazards, all were screened out, except for seismic events, extreme low and high temperatures, high wind events (including hurricanes and tornadoes), ice storms, and lake animals. A list of the natural external hazards considered is presented in Table 4. Seismic and high winds (including straight-line wind and tornadoes) PSA assessments were performed; see details in Section 5.5

and Section 5.6 respectively. In addition, the effects of freezing rain, ice storms, geomagnetic storms, blockage of intake tunnel (e.g., by fish, algae), and the effects of extreme low / high temperatures were modelled in the internal events PSA.

4.1.4 Combined External Hazards

Combinations of external hazards may have a significant impact on diverse safety systems at the same time. Therefore, evaluation of the combination of events is an essential part of the external hazards screening for PSA to ensure the consequences of combinations are not disproportionate. Combined external hazards include combinations of man-made hazards with natural hazards, human-induced hazards with other human-induced hazards, as well as combinations of natural hazards. In particular, some combinations of natural hazards can be correlated (e.g., high winds and flooding can both occur in summer storms) and could potentially produce the most severe impacts challenging the safe operations of the nuclear plants. Review of the international practices shows that combinations of external hazards are considered only if the hazards are correlated and dependent. Independent combinations of beyond design basis hazards usually have an extremely low likelihood of occurrence. The objective of the assessment is to ensure the combinations would not have significant impacts on diverse safety systems at the same time, and do not impose disproportional risks to the station's safe operation. Several hundred combinations of external hazards were assessed. The combined hazard assessment did not identify any hazard combination that requires additional PSA assessments.

4.2 External Hazards Screening for Non-Reactor Sources – IFB

For hazards that have not been screened out, hazard screening analysis is performed taking into account the frequency of the hazard, the magnitude of the hazard and the effect of the hazard upon the IFB. The frequency and magnitude of external hazards such as seismic events, high winds and external floods prepared as part of the reactor PSAs may be used for IFB PSAs.

4.2.1 Human-Induced External Hazards

The methodology used for screening the human-induced external hazards for IFB is the same as described in Section 4.1.1. A list of the human-induced external hazards assessed is presented in Table 5.

The hazards not screened out in Table 5 were addressed in the non-reactor source PSA (see Section 6.13).

4.2.2 Natural External Hazards

The natural external hazards were assessed through a hazard screening assessment to determine if they are applicable to the IFB at the PNGS-B.

Similar to Section 4.1.3, the RLCs defined for the reactor units are considered applicable to the IFB because the reactor units and IFB are at the same site. The list of natural external hazards can be found in Table 6.

The hazards that were not screened out in Table 6 were addressed in the non-reactor source PSA (see Section 6.13).

4.2.3 Combined External Hazards

Specific combinations of external hazards are not explicitly reviewed. Instead, it is judged that the effect of any combination of hazards (correlated, consequential, and coincidental) would be bounded by the IFB Loss of Heat Sink scenario.

4.3 External Hazards Screening for Non-Reactor Sources – Used Fuel Dry Storage (UFDS)

Once the fuel has resided in the irradiated fuel bays for a minimum of ten years, the residual decay heat is sufficiently low to allow this fuel to be moved to dry storage. The hazards are postulated during the following three stages of the Used Fuel Dry Storage (UFDS):

- On-site transfer operations;
- Operations inside the DSC processing building; and
- Dry Fuel Storage (long-term storage).

In order to release Cs-137, which is the radionuclide of concern for the LRF in a PSA, the fuel would need to be melted. The fuel in the DSCs no longer generates enough heat to require active cooling. The hazard screening for the UFDS therefore makes use of this condition, i.e., if the hazard cannot raise the temperature of the dry fuel, then the hazard can be screened out.

4.3.1 Human-Induced External Hazards

Table 7 has been developed to align the listing of the human induced external hazards for the UFDS with those for the reactor in Section 4.1.2. All human-induced external hazards with a potential to impact UFDS are screened out, and do not require a PSA.

4.3.2 Natural External Hazards

Table 8 lists the natural external hazards and provides their screening analysis based on the approach adopted for the analysis of the natural hazards in Section 4.1.3. All natural external hazards are screened out, and do not require a PSA.

4.3.3 Combined External Hazards

Given that individual external hazards do not involve the high temperatures required for a large release of Cs-137 from the UFDS, the combinations of external hazards do not need to be assessed for the UFDS.

4.4 Internal Hazards Screening for Reactor Sources

4.4.1 Overview of Internal Hazards Screening Method

The development of internal hazards screening methodology involved five main steps:

1. Carry out a hazard identification study;
2. Define appropriate hazard screening parameters;
3. Identify hazard screening criteria;
4. Develop a screening calculation methodology, where necessary, to take account of:

- Frequencies of events, SCDF, LRF, CLRP or CCDP associated with a hazard.
- Consequences of events associated with a hazard, including assessment of impact versus distance and/or discrete damage states.

5. Produce the overall results of hazard screening methodology.

The screening flow diagram of steps is the same as for the external events as shown in Figure 4. A preliminary list of the hazards is developed based on a literature review, as well as a site walk down to review vulnerable areas within the powerhouse to identify any additional hazards. As many internal hazards have already been assessed in detail by the different PNGS-B PSA studies, the hazard screening only considered internal hazards not already assessed in PBRA.

For each of the hazards identified, one or more parameters are selected that define the internal hazard and/or its potential impact, and for which discrete and quantifiable criteria can be developed. The qualitative criteria are the same as those for the external events as described in Section 4.1.1. If all qualitative criteria have been examined and the hazard has not been screened out by the seven deterministic criteria, the quantitative screening is required. The five quantitative screening criteria are presented in Table 2.

4.4.2 Internal Hazards Screening Results

The internal hazards identification included mechanical, chemical, electrical hazards, etc., initiated from the inside of the plant; an updated operating experience (OPEX) review was also conducted. The internal hazards identified are listed below:

- Mechanical missile impact;
- Explosions within the generating station main buildings;
- Release of oxidizing, toxic, radioactive or corrosive gases and liquids from on-site storage;
- Release of stored energy;
- Dropped or impacting loads;
- Transportation impact (e.g., vehicles, movement of toxic on-site goods);
- Electromagnetic interference; and
- Static electricity.

The above internal hazards were assessed and all of them were screened out, some based on the consequences (qualitatively), and some based on their low probability of occurrence (quantitatively). Internal hazards for which a PSA already exists (e.g., internal fires, internal floods) were not considered. As a result of the screening assessment, no new internal hazard was identified to be included in the PNGS-B PSA.

4.5 Internal Hazards Screening for Non-Reactor Sources - IFB

Screening assessments for the hazards for the IFB is based on the following considerations and insights:

- Loss of IFB heat sink; and
- Loss of IFB water inventory.

The above hazard conditions can adversely impact the ability to prevent IFB fuel to uncover, as follows:

- Reduced IFB water level can result in increased radiation field in and near the IFB with the potential to inhibit corrective operator field actions such as equipment repair and IFB inventory make-up;
- Boiling of IFB water can result in harsh environment with the potential to cause IFB equipment failure and inhibit corrective operator filed actions; and
- IFB inventory leakage events (e.g., pipe break) can cause IFB equipment failure and inhibit corrective operator filed actions.

The internal hazards that were not screened out were assessed further as part of the non-reactor PSA (see Section 6.13). A list of screening of the internal hazards for IFB is presented in Table 9.

4.6 Internal Hazards Screening for Non-Reactor Sources - UFDS

The hazard screening assessment for internal hazard is similar to the assessment conducted for UFDS for external hazards (see Section 4.3).

All internal hazards with the potential to impact UFDS have been screened out based on the criterion stated in Section 4.3. As such these hazards do not require a PSA for the UFDS. A list of screening of the internal hazards for UFDS is presented in Table 10.

5. LEVEL 1 PSA METHODS

The goal of a Level 1 PSA is to identify occurrences at the plant that can cause a transient that would challenge fuel cooling, identify what systems can be credited to mitigate the event, assess what the impact of the transient may be on the mitigating systems, and to determine and quantify the degree of fuel damage that would occur if the mitigating systems fail.

Typically, the first PSA study for a station will be a Level 1 At-Power internal events PSA. Much of the effort of this study is in constructing models of what mitigating systems can be credited for a given transient, and how the mitigating systems can fail. In PSAs for other types of IEs, e.g., internal fire, internal flood, seismic events, and high winds, much of the effort is associated with determining the impact these events have on the mitigating systems. The descriptions of the methodology for the various Level 1 studies in the following subsections reflect different requirements for the different studies.

The Level 1 At-Power PSA model is used to aid in the development and quantification of the internal events outage, seismic, internal fire, internal flooding, and high wind PSAs.

5.1 Level 1 At-Power Internal Events

The At-Power Internal Events PSA for PNGS-B has been developed following the methodology for preparation of a Level-1 Internal Events At-Power OPG PSA guide.

The major activities of a Level 1 Internal Events PSA are listed below:

(a) Identification of IEs based on a review of station-specific operating experience, and knowledge gained from previous probabilistic safety assessment studies. The identification of IEs is discussed in Section 5.1.1.

(b) Development of a scheme to group sequences into a manageable number of consequence categories based on degree of fuel damage, as discussed in Section 5.1.2.

(c) Development of Event Trees (ETs) needed to establish what consequences can occur following a particular IE, given success or failure of the systems credited with mitigating the IE. Development of the PBRA event trees is discussed in Section 5.1.3.

(d) Development of system level Fault Trees (FTs) needed to quantify the probability of failure of the mitigating systems credited in the event trees (including support systems that interface with mitigating systems). The development of the FTs is discussed in Section 5.1.4.

(e) Development of a component reliability database with, to the extent possible, information specific to PNGS-B. The reliability database is needed to support the FT analysis mentioned above. The sources of the data in the component reliability database are also discussed in Section 5.1.4.

(f) Assessment of the effect of human error on accident progression and system performance using Human Reliability Analysis (HRA). The potential for human errors must be incorporated along with hardware failures in the system level FTs and ETs, and the Human Error Probabilities (HEP) systematically estimated and assigned. Human errors are referred to as “human interactions” in PBRA. The HRA is discussed in Section 5.1.5.

(g) Integration of ETs with the system FTs, and risk quantification. This step combines the accident sequences developed in the ETs with the system logic contained in the system FTs to produce integrated FTs representing each of the fuel damage categories. The integration process is described in Section 5.1.6.

Although the above listed tasks are carried out in the indicated order, the process is iterative in nature and entails re-assessing the results of a previous task based on insights gained from a subsequent one.

The major activities of the Level-1 At-Power methodology are summarized in the subsections below.

5.1.1 Initiating Events Identification and Quantification

An IE is a disturbance at the plant that challenges reactor operation or fuel integrity either by itself or in conjunction with other failures. Identifying and quantifying the IEs is the first step in the Level 1 PSA process.

In PBRA-L1P, consistent with the above definition, the IEs under consideration are primarily those plant failures that could lead directly, or in combination with other failures, to damage to fuel in the reactor. The list of PBRA IEs includes events leading to a hostile environment in the powerhouse, i.e., steam line breaks and feedwater line breaks.

The objective of the IE selection task is to obtain as complete a coverage as possible of credible IEs. To create the IE list, past OPG PSAs were reviewed, as were the plant operating experience and station condition records, and other published PSAs. In addition, insights gained from the FT modelling, discussed in Section 5.1.4, identified other IEs.

The complete list of IEs considered in PBRA-L1P is provided in Table 11.

The IEs are quantified primarily using Bayes' Theorem. In a Bayesian approach, the prior distribution is calculated based on the operating experience and assumptions regarding the behaviour of the events at the various plants, while the posterior distribution is calculated based on the prior distribution and the station-specific operating experience. This technique allows general experience and knowledge about a given event to be combined with actual operating experience gained with the station under study. It is especially useful for quantifying the frequency of events unlikely to be experienced within the lifetime of a single station. This is the standard industry method.

5.1.2 Fuel Damage Categorization Scheme

Each accident sequence, consisting of an IE and failures of mitigating systems, may potentially result in a different end state at the plant. The plant end states will vary in terms of the severity and timing of fuel damage. Fuel damage categorisation is carried out to simplify the subsequent evaluation of consequence and risk. Each Fuel Damage Category (FDC) represents a collection of event sequences judged to result in a similar degree of potential fuel damage. The FDCs are used as end-states in the Level 1 event trees, discussed in Section 5.1.3. In addition, groupings of the fuel damage categories are used to transition from the Level 1 PSA to the Level 2 PSA, discussed in Section 6.1.

The range of events or event sequences covered by the FDCs is defined by the scope of the PBRA. From the ET analysis, described in Section 5.1.3, general types of accident sequences can be identified. They are presented below, in general order of decreasing severity of fuel damage:

(a) Severe Core Damage:

- Sequences with the potential for loss of core structural integrity.

(b) Severe Fuel Damage:

- Loss of fuel cooling requiring the moderator as a heat sink; and
- Prolonged loss of heat sink.

(c) Limited Fuel Damage:

- Inadequate cooling to fuel in one or more core passes following a large LOCA with successful ECIS initiation; and

- Sequences leading to fuel damage in one channel with and without an accompanying automatic containment isolation (button-up).

(d) Negligible Fuel Damage:

- Loss of HTS integrity followed by successful ECIS initiation with no significant fuel damage.

The lower consequence threshold for significance is deemed to be the occurrence of a loss of HTS integrity resulting in ECIS initiation. Although fuel damage is not likely, the event is considered to have the potential for significant economic consequence. At the other extreme are the events that have the potential for severe consequences involving the loss of core structural integrity. Within this general framework, the objective has been to generate a categorisation scheme with as few categories as possible, while continuing to distinguish between the most important event characteristics that affect consequence.

The FDCs used in PBRA are presented in Table 12. These FDCs are also used to calculate the frequency of severe core damage, for comparison to the relevant OPG safety goal. Severe core damage is defined to be the sum of the FDC1 and FDC2 frequencies.

5.1.3 Event Tree Analysis

The potential for accidental release of fission products contained in nuclear fuel constitutes the main risk from a nuclear power plant. In the Level 1 analysis, ETs are used to systematically review the possible ways that radioisotopes can be released from the fuel and to distinguish between varying levels of fuel damage and isotope release resulting from different accidents.

Since a nuclear plant is a complex system, the search for accident sequences must be conducted in a systematic and structured manner. This analysis requires both a thorough understanding of the plant design, operation, maintenance and testing, and the ability to translate that understanding into a model of the plant that captures the logic of the sequences leading to fuel damage.

These sequences are constructed using inductive logic. The graphical representation of this inductive logic is called an ET. The start of this inductive method is the IE, usually a plant malfunction. Following the identification of the IEs, the next step is to consider what systems are required to mitigate the event and show how the accident could progress if failures of the mitigating systems were also to occur, until a previously defined end state is reached.

Event tree analysis requires the following to be predefined:

(a) A list of IEs to be considered;

(b) Definition of sequence end states; and

(c) Definition of mitigating systems and corresponding ET branch point labels.

A generic event tree for a large LOCA at a CANDU plant is presented in Figure 5 as an example. A LOCA is typically a pipe break in the heat transport system. Following a large LOCA, three systems are postulated to mitigate releases of radioisotopes: the shutdown systems, ECI and the heat sink function of the moderator system. The potential plant state must be assessed if one or more of these systems fail. These three systems form the branch points in the event tree. The event tree is read from the left, starting at the IE “IE-LOCA”. The first systems credited with preventing fuel damage are the shutdown systems. Failure of both SDS1 and SDS2 is represented by the event tree branch point “SD”. The shutdown systems, SDS1 and SDS2 are fast acting, diverse and independent systems. The convention used to interpret an event tree is that success of the system is the top path and failure is the lower. If the shutdown systems fail, rapid loss of core structural integrity is expected. This sequence is assigned to the FDC1 end state. If reactor shutdown is successful, the decay heat from the fuel must still be removed to prevent fuel damage. Two systems are credited for this function: automatic ECI injection and the moderator as a heat sink. If ECI fails, represented by the event tree branch point "ECI", then the moderator is credited to prevent severe core damage. However, if the moderator system fails, a slow loss of structural integrity is expected. This end state is FDC2, one of the fuel damage categories included in the definition of severe core damage. If the moderator system is successful, the less severe FDC3 category is assigned.

If both shutdown and ECI are successful, the end state FDC9 is reached. This category represents no significant fuel damage, and no releases to the public, but has significant economic consequences.

Once the Level 1 event trees have been created, the mitigating systems that have been identified in the event tree analysis require FT modelling to calculate the probability of failure of the mitigating function. FT analysis is described in the next section.

5.1.4 Fault Tree Analysis

A FT is a logic diagram that models the possible causes of a particular fault, usually a system failure, and is used to calculate the probability that the fault occurs. In PBRA, FTs are used to quantify the probability of the failure of the mitigating systems that appear in the event trees discussed in Section 5.1.3, and for their support systems. Table 13 lists the systems modelled by FTs in the PBRA-L1P study. Figure 6 depicts the relationship between the event trees and FTs. System FT analysis is used to calculate the probability of an event tree branch point given a specific set of events that fail the system.

Each FT is a logic diagram developed for a failure mode of interest and is based on the understanding of system design and operation. At the top of the diagram the event itself is noted and termed the "top event". The process of FT analysis is a deductive, systematic way of failure analysis whereby an undesired state of a system is specified (i.e., top event), and the system is analyzed in context of its environment and operation to find all credible ways in which the undesired state can occur. Thus, through this process, the contributors to the top event are identified.

The "CAFTA" software code is used for developing and quantifying the FTs [R7].

For example, consider SDS1. For this system, the failure mode of interest might be "fails to shutdown the reactor when required". Figure 7 shows a partially completed FT with this event at the top. Starting from this top event, the FT analyst poses the question "How can this event occur?". The answers to this question become the inputs to this top event. For example, Figure 7 shows that Shutdown System 1 can fail if the rods fail, the shutoff logic fails, or if a combination of shutoff logic and rod failures occur. For each of these contributors, the process of examining how they can occur is repeated, until no further insights can be obtained about the behaviour of the system. Typically, the FT is developed either to predefined system boundaries, or to the individual system components.

In constructing a FT model, a number of design and operational features are assessed.

(a) System capability: For example, how many rods need to operate to shutdown the reactor?

(b) Fault detection: For example, if a component has failed, when and how can its failure be detected?

(c) Common Cause Failures (CCF): For example, if a component failed due to any number of causes, application of CCF will force the analyst to ask the question, would any other similar component fail for the same reason?

(d) Failure criteria: For example, what fundamental failure modes lead to failure of the Shutdown System 1?

(e) Fault tolerance: For example, if the electrical systems have failed what is the impact on the shutoff rods?

The basis for system capability and the failure criteria is based on analysis from a variety of sources, including the safety analysis contained in the PNGS-B Safety Report, Operational Safety Requirements (OSR), Abnormal Incidents Manuals (AIMs), and assessments and regulatory submissions.

In principle, the FT analysis technique is straightforward. An undesired event is postulated and then, deductively, its contributors are identified. However, this process requires a detailed understanding of the system design and function, and how it behaves under fault conditions.

Once the FT is constructed, it is linked with the system reliability database, a database containing the information to calculate the probability of each event in the FT. In PBRA, failure rate, test and maintenance data are assigned to the FT primary events from a central type code table that is linked to the system reliability database. This type code table defines failure rates for the various components at the PNGS-B. The use of the CAFTA compatible reliability database and a central type code table ensures that the same type of component is assigned the same failure rate for the same failure mode in all system FTs.

The nuclear industry has adopted a Bayesian approach for obtaining component failure rates. The Bayesian approach is based on the use of both the generic “prior knowledge” and the plant-specific data in deriving the failure rates. Three industry sources, U.S. Nuclear Regulatory Commission (NRC) [R6], T-book [R8], and Westinghouse Savannah River Company [R9], are used for obtaining generic data. PBRA plant-specific data is used for the Bayesian update.

The reliability database also contains information on human errors modelled in the FT and event trees. The analysis of human errors and their quantification are discussed in the next section.

5.1.5 Human Reliability Analysis

Human errors can affect the performance of systems, and in some cases be significant contributors to risk. Thus, HRA is an important part of PBRA. The potential for human errors must be incorporated along with hardware failures in the system level FTs, and human error probabilities systematically identified and assigned.

The overall objective is to include all human interactions that can potentially lead to a significant increase in the probability of component or system failure and that are not already reflected in the plant failure rate database.

In principle, every piece of equipment or system in the plant is susceptible to failure because of human error; however, human errors that contribute directly to the failure of individual components are included in the equipment reliability database and need not be identified explicitly in FTs (e.g., miscalibration of transmitters). The human errors of interest to the FT analyst arise under five sets of circumstances:

(a) where an otherwise operable component, subsystem or system can be disabled (i.e., prevented from performing its design function) prior to an IE;

(b) where an equipment failure occurs but the operator does not respond to the failure prior to an IE;

(c) where an operator action or a closely related series of actions can cause more than one piece of equipment in parallel or redundant pathways to fail or become disabled simultaneously prior to an IE;

(d) where the operator can fail to make the appropriate response to return the plant to a stable state following an IE (by not taking any action at all or by taking the required action but in an inappropriate way); and

(e) where the operator can plausibly interfere with correct responses by inhibiting or activating a system.

A human interaction in a FT identifies an opportunity for a human to make an error. Only those opportunities that arise in carrying out established plant operating practice are included; specifically, errors that can be made in carrying out maintenance, testing, normal plant control, and post-IE control activities. In most cases, these errors would be made while carrying out formal procedures. Random, spurious, wilful, or vengeful actions are not included.

In order to systematically quantify the human interactions in the PBRA, OPG uses a human interaction taxonomy. This taxonomy classifies the human interactions in PBRA-L1P into three parts: Part 1 contains the simple interactions that, by definition, occur prior to an IE; Part 2 contains complex human interactions that occur prior to IEs; and Part 3 contains the complex interactions that occur after an IE.

Simple human interactions have the following characteristics:

(a) They are based on written or learned procedures (as opposed to cognitive or creative tasks).

(b) They involve directly manipulated components (e.g., a valve handwheel or a handswitch) or directly viewed main control room display devices.

(c) They occur prior to an IE.

The task of assigning preliminary (screening) HEPs for the simple human interactions is made easier and faster using a simple method requiring only selection of an unmodified basic HEP and predefined modifying factors. This method quantifies the human interaction based on the type of task, the location where the task is performed, whether the error can be detected in the main control room, and if any annunciations or inspections can detect the error.

For the complex human interactions that occur prior to IEs, the same process may be followed to obtain a preliminary (screening) quantification. These human interactions are complex because they include system level functions that involve more than just direct physical manipulation of a component, such as the setting of computer control program parameters or modes.

Post-IE complex human interactions usually occur during abnormal conditions and are, therefore, more difficult to identify, analyze, and quantify. Additionally, interactions involved in handling unit upsets are also unlike other interactions as they may take place in dynamic and uncertain situations. Such actions depend upon the cognitive functions of diagnosis and decision-making. These actions are knowledge-based; they are based on fundamental principles of process and safety system operation and on understanding of the interactions amongst these systems.

For the post-IE complex human interactions, the preliminary (screening) human error probabilities are assigned based on three criteria: whether the task is straightforward, of average complexity, or very complex; the time available; and the quality of indication available in the main control room to indicate that action is required.

Human interactions that are identified as risk significant can be further refined by the HRA Specialist on a case-by-case basis using a methodology such as Technique for Human Error Rate Prediction (THERP) [R10].

5.1.6 Fault Tree Integration and Evaluation

The FT and associated failure rate data contain the information necessary to calculate the top event probability and identify the dominant contributors to failure for the individual system. Integration is the process of merging the system FTs with the event trees to create logic for the fuel damage (i.e., Level 1) and release categories (i.e., Level 2). The end goal of the integration step is to develop a model that can be used to calculate the frequency of occurrence for each of the end states, i.e., the fuel damage categories and release categories. Combining this information in one model allows dependencies between systems to be identified and quantified correctly.

The information required to quantify the fuel damage categories is stored in the FTs and event trees. In order to combine the two, the ET logic is converted into FT logic with a top event for each FDC. These FTs are referred to as the high level logic. The events in the high level logic are the IEs and the branch points from the ETs. The high level logic is then integrated with the mitigating system FTs; the top events in the mitigating system FTs are inserted where the mitigating system branch point labels exist in the high level logic model. Finally, the support systems are added to the integrated high level logic FT. Figure 8 illustrates this process.

The CAFTA software [R7] is used to evaluate the FT models and FTREX program is used as the solution engine to quantify the results [R11]. CAFTA was developed by the Electric Power Research Institute (EPRI).

The solution of the integrated FT for each FDC is expressed as a listing of the combination of an IE, equipment failures, and human errors that leads to the occurrence of the integrated FT top event, with each combination containing the minimum number of failures that must occur to cause the top event. Such failure combinations are called minimal cutsets.

The solution of the FT calculated using CAFTA is truncated. That is to say, contributors below a certain frequency are not included in the solution. Truncation is necessary because of computational limits. The truncation limit selected should be low enough that all significant contributors are captured. The Level 1 At-Power PSA Guide recommends that the solution of the integrated FT for each FDC be truncated at either 4 orders of magnitude below the most likely minimal cutset in that FDC or at 1E-12 occ/year, whichever is the highest. For FDC2, the top cutset frequency is in the 6.9E-08 occ/year range, and a truncation of 6.9E-12 occ/year is used.

Following the development of the baseline PSA results, an additional understanding of the station risk is obtained by supplementing the baseline solution with the following:

- Accident sequence quantification to provide sequence by sequence cutset ranking;
- Importance analysis to identify systems and components that are important to the FDC results;
- Parametric uncertainty analysis to determine the mean values and lower and upper limits of the two-sided 90% confidence interval for the frequency of each FDC and SCD; and
- Sensitivity analysis used to evaluate the impact on the results of a number of assumptions made in the ET analysis and FT analysis, as well as assumptions impacting the quantification of IEs, undeveloped events, and human error events.

Recall from Section 3 that risk has two components: the frequency of occurrence and the consequences. Section 5.1.1 to Section 5.1.6 described the methods used to quantify the frequency of occurrence of the fuel damage categories. The Level 1 analysis is used as an input to the Level 2 analysis described in Section 6. The remaining subsections in Section 5 describe the differences in methodology for Level 1 assessment for the outage state, internal fire, internal flood, seismic and high wind IEs.

5.2 Outage Internal Events

The Level 1 At-Power PBRA considers internal events occurring at 100% full power operation. However, the PNGS-B has periods of planned outage to perform routine maintenance and testing that cannot be done during full power operation. Typically, a unit has a planned outage for less than 10% of the operating cycle. The reactor power continues to decrease exponentially after reactor shutdown.

The 2022 PBRA-L1O has been developed as a bounding assessment of the 2017 PBRA-L1O and follows the methodology for preparation of a Level-1 Outage PSA.

The Outage PSA uses many of the same techniques as used in the At-Power PSA. The PSA process for outage uses IEs, event tree analysis and FT analysis, much like the At-Power PSA. However, different IEs can occur in the outage state, and the event tree and FT must reflect the plant configurations during the outage (e.g., HTS pressurized or depressurized). The plant configurations modelled as part of the outage PSA are typically described as Plant Operational States (POS).

Determining the possible plant configurations is a major part of the outage PSA and is described in the next section.

5.2.1 Plant Operational State Analysis

The purpose of POS analysis is to define the various outage plant scenarios and group them into fewer, representative and bounding states for which the plant status, configurations and system failure criteria are considered sufficiently stable. POS analysis is unique to Outage PSA. During unit outage, plant system configurations and parameters are dynamic, changing with respect to time. The dynamic nature of outage, specifically system configurations, process parameters and varying system failure mechanisms, result in an excessively large number of unique plant scenarios to be analyzed. In the definition of the POSs, only normally planned plant configurations are considered.

Pre-Plant Operational States (Pre-POSs) were identified; Pre-POSs were defined as unique outage plant configurations wherein all parameters of interest (system configuration and parameters, e.g., heat transport system pressure, primary heat sink, HTS level) were considered stable for the duration of the state. Pre-POSs are the highest resolution of the outage states. The pre-POSs were grouped into smaller set of POSs. For the PBRA-L1O, eight pre-POSs were identified and have been grouped into five representative POSs. The five POSs were used in other aspects of the Outage PSA, including accident sequence analysis using event trees. Table 14 provides a summary of the final POSs used in the PBRA-L1O model. The parameters used to define the POSs are listed in the leftmost column.

5.2.2 Initiating Event Identification and Quantification

The development of a Level-1 Outage PSA requires the identification, grouping and quantification of a set of outage IEs that could occur during the identified outage POSs. An outage IE is defined as a malfunction that can, either independently or in conjunction with other plant conditions or configurations, lead to fuel damage when the unit is in the guaranteed shutdown state.

The process described below is used to identify, group and quantify outage state IEs:

- The outage IE identification process uses a number of different steps and different sources of information, so that the basis for the Outage PSA is as comprehensive as possible.
- The identified IEs are grouped on the basis of similar mitigation requirements, in order to simplify the accident sequence analysis.
- The frequency of occurrence of each IE (or IE group) is estimated, so that the overall risk of core damage can be calculated.

Table 15 presents the list of outage IEs for the PBRA-L1O, and which POS each IE can occur in. Some IEs can occur only in specific plant configurations. For example, ice-plugs are used during some maintenance activities on the HTS but can only be used while the HTS is depressurized (i.e., POSC and POSD).

5.2.3 Outage Event Tree Analysis and Fuel Damage Category Analysis

The event tree process for the internal outage events trees is similar to that used for the At-Power event trees described in Section 5.1.3.

The overall process followed to develop the ETs for PBRA-L1O is as follows:

1. For each unique IE/POS combination, identify the mitigating systems credited for the IE based on a review of the accident analysis and plant operating procedures.
2. Determine the end states of interest in the ET analysis. For the PBRA-L1O, the outage fuel damage categories are shown in Table 16.
3. Develop the accident sequence logic depending on the success and failure of the mitigating functions credited for the IE.
4. Add the branch point label for each mitigating system failure as the logic is being developed on the failure branch of the ET.
5. Assign a FDC to each ET sequence end state.

5.2.4 Outage System Fault Tree Analysis

The FT analysis process for the internal outage PSA is the same as for the At-Power PSA. However, the FT models are significantly different to reflect the outage configurations of the system.

The system FT models are specific to the outage PSA. Each FT includes a brief overview of the system analyzed, top event definitions, assumptions, failure criteria, FT diagram, data table, results expressed as minimal cutsets, system failure probability and importance indices. Table 13 lists the systems modelled by FTs in PBRA-L1O.

5.2.5 Reliability Data Analysis

The objective of reliability data analysis is to derive the reliability data assigned to the primary events modelled in the PBRA-L1O system FTs. Primary events include basic events (e.g., component hardware failures), conditioning events (i.e., events used to specify a condition or restriction that applies to the FT logic), developed events (i.e., specific fault events related to external interfaces which are typically developed in separate FT models), and undeveloped events (i.e., specific fault events not amenable to further development and so quantified using specialized methods).

Like in the At-Power PSA, a Bayesian approach is used for obtaining component failure rates. Conditioning events, developed events, and undeveloped events, for which component failure rates are not applicable, are also quantified using one of the following methods:

- Operational events are quantified from observation of operating experience; or
- Analytical events have a probability of occurrence that is determined from the results of analytical models outside of the FT, engineering judgement, or both.

5.2.6 Human Reliability Analysis

The possibility of component or system failure due to human error is recognized by the inclusion of human interactions in the FTs and ETs. The scope of the HRA includes inadvertent errors by plant operators or maintainers that may contribute to the failure of systems or components but excludes consideration of arbitrary or wilful actions. Ultimately, the human error probabilities are combined with equipment failures in the system FT to provide the overall probability of the top event. In the ETs, the human error probabilities are addressed along with system and/or equipment failures to provide accident sequence frequencies.

While the methodology for quantifying human interactions in the Outage PSA is generally the same as in the At-Power model (see Section 5.1.5), the effort required to identify, quantify and model human interactions in Outage PSA is not trivial. The human interactions during outage states require the consideration of the many testing and maintenance activities, procedures, and manual initiation of certain mitigating systems. The HRA specialist considers the outage POSs and system configurations to better understand required operator actions, recall actions, and possible testing and maintenance activities during a given POS.

5.2.7 Model Integration, Quantification, and Additional Analysis

Once the ETs and FTs are developed, they are linked to determine the frequencies with which various fuel damage consequence categories can occur. Categories, here, are groupings of sequences with similar consequences. As the linked models can be of large size, computer aided methods are used to carry out the computations. The results are expressed in terms of the expected number of occurrences of the consequence category per unit time (i.e., frequency). Only those failure combinations that have frequencies greater than a certain cut-off value are listed. The frequency of the consequence category is obtained by summing the frequency of each sequence belonging to that category.

For outage severe core damage consequence categories (e.g., FDC2-SD), the magnitude of the associated consequence was assessed. The risk estimate is obtained by summing the frequencies of all FDC2-SD sequences. These are used in absolute terms to assess the overall safety design adequacy, and in relative terms to identify the dominant risk contributors. The acceptability of the PNGS-B risk estimates is judged based on comparison with the risk-based safety goals.

5.2.8 Level 1 Outage Internal Events PSA Bounding Assessment

The 2022 PBRA-L1O update is a bounding assessment, undertaken in accordance with the principle in REGDOC-2.4.2 that the level of detail in a PSA should be consistent with the level of risk. The OPG Outage PSA Guide forms the general basis for conducting the outage PSA update. Given the relatively low risk from outage units compared to other contributors to station risk – this bounding assessment meets a graded approach for 2022 PBRA-L1O.

The overall objective of 2022 PBRA-L1O bounding assessment is to provide severe core damage frequency (SCDF) estimates for outage internal events, in a manner consistent with the applicable OPG outage PSA Guide, reflecting the current Pickering design and operation to the extent practical for a reduced scope bounding assessment.

5.3 At-Power Internal Fire

The OPG Internal Fire PSA Guide has been developed based on the United States Nuclear Regulatory Commission (U.S. NRC) Fire PSA methodology, NUREG/CR-6850 [R14], its supplement [R12], and EPRI guidance and U.S. NRC endorsed responses to Frequently Asked Questions (FAQs). The major activities of the fire PSA methodology and its application in the development of the Pickering NGS B Internal Fire Probabilistic Safety Assessment (PBRA-IFPSA) are summarized in the subsections below.

An internal fire PSA is built from the internal events PSA. The scope of the PBRA-IFPSA model is limited to internal fires occurring while the unit is at power with the potential to cause severe core damage. Internal fires considered are those resulting from ignition events within fixed equipment (e.g., electrical panels, pumps, etc.) as well as transient ignition events resulting from human activities in the plant (e.g., combustible material storage, hot work, etc.). The purpose of a fire PSA is to establish whether the design and operation of the plant poses an acceptable level of risk to the public and to identify the major sources of risk due to internal fires.

The PBRA-IFPSA model considers sequences that result in severe core damage. Severe core damage states include the FDC1 and FDC2 sequences. However, severe core damage in PNGS-B is dominated by the FDC2. In the PNGS-B Fire PSA FDC2 is chosen as the representative Level 1 PSA risk measure for most fire-induced events given:

- The low frequency of FDC1 sequences in the internal events model (see Section 7.1);
- The fail-safe design of the two shutdown systems (SDS1 and SDS2); and
- The physical separation of SDS1 and SDS2.

Exceptions to this guidance are noted if the Multiple Spurious Operations (MSO) assessment shows that for some fire scenarios failure to shutdown cannot be precluded for which FDC1 can be selected.

The Fire PSA Guide prescribes a phased evaluation of internal fire risks. In each phase, appropriate technical bases and methods are applied; the difference is in the degree to which simplifying assumptions are made as the significant contributors to risk are addressed.

As the fire PSA is developed based on the internal events PSA, the major tasks in the fire PSA are associated with identifying possible fire scenarios, the zones the fires can impact, affected equipment and cables, and selection of representative internal events sequences and quantifying the consequences of the fire scenarios.

The OPG Fire PSA methodology is broken down into 18 tasks:

Task 1 – Plant Boundary Definition and Partitioning

Task 2 – Fire PSA Component Selection

Task 3 – Fire PSA Cable Selection

Task 4 – Qualitative Screening

Task 5 – Fire-Induced Risk Model

Task 6 – Fire Ignition Frequencies

Task 7 – Quantitative Screening

Task 8 – Scoping Fire Modeling

Task 9 – Detailed Circuit Failure Analysis

Task 10 – Circuit Failure Mode Likelihood Analysis

Task 11 – Detailed Fire Modeling

Task 12 – Post-Fire Human Reliability Analysis

Task 13 – Seismic-Fire Interactions Assessment (outside the scope of the PNGS-B Fire PSA, addressed through alternate methodology)

Task 14 – Fire PSA Level 1 Quantification

Task 15 – Uncertainty and Sensitivity Analysis

Task 16 – Fire PSA Documentation

Task 17 – Fire PSA Level 2 Quantification

Task 18 – Alternate Unit Assessment

The integration of these tasks is shown in Figure 9. The methods described in the OPG Internal Fire PSA Guide are iterative. Several of the tasks listed above involve calculation of SCDF due to fires in various plant locations. With each subsequent calculation, the methods used to assess the risk for the various scenarios are refined. This iterative approach is used to identify high risk areas and to focus the detailed fire analysis on these areas. A summary of the methodology used for PBRA-IFPSA is provided in the following sections.

5.3.1 Plant Boundary Definition and Partitioning (Task 1)

This first task in the fire PSA involves the division of the plant into discrete areas called Physical Analysis Units (PAUs). This requires defining the global boundary analysis to ensure that those plant areas where a postulated fire could impact the PSA are included in the analysis. Once the global analysis boundary is defined, the buildings that are within the boundary are examined for potential sub-division into PAUs. The PAUs used in the PBRA-IFPSA assessment are based on those identified in the PNGS-B Fire Protection Program documented in the Fire Hazard Assessment (FHA) and Fire Safe Shutdown Analysis (FSSA). This approach allows the fire PSA to rely on the existing programmatic controls and design requirements for maintaining the integrity of the associated compartment boundaries.

5.3.2 Fire PSA Component (Task 2) and Cable Selection (Task 3)

The development of a fire PSA requires identifying components and their associated cables' locations necessary for safe shutdown and long-term decay heat removal following a fire. A fire can affect the equipment / cables credited for safe shutdown by either being in the same area as the credited equipment or by being in the same area as the cables related to the credited equipment.

The purpose of these tasks is to identify the equipment / associated cables to be explicitly credited in the fire PSA, determine where in the plant, and in which PAU they are located.

The selection of PSA-credited equipment / cables required for safe shutdown following a fire is based on the systems credited in the PNGS-B FSSA.

5.3.3 Qualitative Screening (Task 4)

This task involves the identification of fire analysis compartments that can be shown to have little or no risk significance without quantitative analysis. The PAUs can be screened out if they do not contain PSA-credited components or cables, and cannot propagate fires into PAUs containing such components and cables, and if they cannot lead to a plant trip due to either plant procedures, an automatic trip signal, or OP&P requirements.

5.3.4 Fire-Induced Risk Model (Task 5)

This task involves the development of a logic model that reflects plant response following a fire. This requires modification and / or manipulation of the PBRA-L1P model to produce a fire-induced risk model, including fire-induced impact on operators' response following a fire, as discussed in Task 12. That model is used to calculate Conditional Core Damage Probabilities (CCDPs) for postulated fires (e.g., scenarios from Tasks 7, 8 and 14).

5.3.5 Fire Ignition Frequencies (Task 6)

To calculate the risk due to an internal fire, Fire Ignition Frequencies (FIFs) for each PAU identified in Task 1 should be assessed.

Unit 5 is used as the reference unit for PNGS-B with consideration of applicable shared systems and areas that could impact Unit 5 operation. The calculation of FIFs for Unit 5 required calculation of FIFs for all of the PAUs that are within analysis boundary. This was accomplished, as per guidance provided in NUREG/CR 6850 [R14] and Supplement 1 [R12], by:

1. Conducting Fixed Ignition Sources (FISs) walkdowns of Unit 5 PAUs and common PAUs;
2. Assuming that Unit 5 is spatially representative of the other three operating units, replicating the Unit 5 FISs walkdown data for PAUs in Units 6, 7 and 8 where applicable;
3. Transient FIF development based on engineering judgment from site personnel who are familiar with the daily activities of the plant.

The generic frequencies in NUREG-2169 [R13] were updated to include review and consideration of the Operating Experience (OPEX) for PNGS until December 31st, 2020. The fixed ignition sources fire frequency, the transient ignition sources fire frequency and the total fire ignition frequency were calculated for each PAU identified in Task 1.

5.3.6 Quantitative Screening (Task 7)

The development of a fire PSA allows for a quantitative screening of PAUs based on their contribution to fire risk. This task considers the cumulative risk associated with the screened PAUs (i.e., the ones not retained for detailed analysis) to ensure that a true estimate of fire risk profile (as opposed to vulnerability) is obtained. With the information from the fire risk model and FIFs (described in Sections 5.3.4 and 5.3.5), the contribution to severe core damage for each PAU can be calculated. Based on the severe core damage contribution of each PAU, the areas of the plant are further screened, using quantitative screening criteria.

Areas of the plant that are screened out of the analysis during this step are still retained and included in the final fire-induced risk estimates (e.g., SCDF and LRF). They are excluded from further refinement of fire scenarios in risk-significant areas.

5.3.7 Scoping Fire Modeling (Task 8)

This task is intended to provide a conservative and simplified means to develop an initial refinement to the bounding treatment in Task 7. It involves the use of generic fire models for various fire ignition sources such that simple rules can be used to define and screen fire ignition sources (and therefore fire scenarios) in an unscreened PAU. The generic fire models can be also used to develop simplified treatments for specific fire ignition sources and their impact on nearby targets (cables) and thereby eliminate the need for numerous explicit detailed fire modeling analysis. The information from these models was combined with walkdown information using raceway identifiers to characterize the extent of fire impact to plant systems.

This task has two main objectives:

- To screen out those FISs that do not pose a threat to the targets within a specific fire compartment; and
- To assign severity factors to unscreened FISs.

To meet these objectives, Task 8 developed fire scenarios for the unscreened PAUs from Task 7 quantitative screening and assigned CCDP cases to each scenario.

5.3.8 Detailed Circuit Failure (Task 9) and Failure Mode Likelihood Analysis (Task 10)

The development of a full-scope fire PSA requires detailed circuit failure analysis and circuit Failure Mode and Likelihood Analysis (FMLA). The purpose of these analysis is to identify additional components and cables to include in the scope of this analysis. Tasks 9 and 10 only need be applied to cables that were not previously analyzed in the FSSA. Since no components and cables were added, these analyses were not required and therefore have not been performed.

5.3.9 Detailed Fire Modeling (Task 11)

Detailed fire modeling can be used to perform fire ignition source (scenario) specific fire modeling to address risk-significant scenarios in cases where the Task 8 results in Section 5.3.7 are producing overly conservative treatments. Detailed fire modeling is performed only in those instances where such analyses produce substantially improved results as compared to those obtained from Task 8. This task was not performed for individual risk-significant scenarios, but was performed for:

- MCR Abandonment scenarios; and
- Hot Gas Layer (HGL) and Multi-Compartment scenarios.

The abandonment times for operators in the PNGS-B MCR envelope were assessed for electronic equipment fires and ordinary combustible fires within the MCR envelope.

The purpose of multi-compartment analysis is to calculate the probability of compartment interaction caused by a HGL due to smoke/heat propagation. The calculation involves multiplying the probability of a HGL in the PAU (i.e., the probability that the fire creates a hot smoke layer) by the PAU barrier failure probability (i.e., failure of fire doors, dampers and penetrations).

5.3.10 Post-Fire Human Reliability Analysis (Task 12)

A review of PBRA-L1P was performed to identify the post-initiator operator actions modeled as human failure events along with their associated HEP. Pre-initiator operator actions and operator actions associated with non-fire induced events were excluded from consideration.

For each fire-related BE that represents a post-initiator operator action modeled as human failure, HEP multipliers were developed for fire PSA adjustments. Multipliers have been developed taking into consideration the key performance shaping factors of the actions that can be impacted by the fire event. These include:

- Whether the action is performed inside the control room or in the field;
- The time available to perform the action;
- Whether the fire makes the location of the action inaccessible; and;
- The impact of the fire on the indicators and controls necessary to diagnose and execute the action.

Based on the factors above, all the post-initiator operator actions from the PBRA-L1P were reviewed and their HEP values were adjusted by multiplying factor of 1 to 30. No additional post-initiator operator actions were credited for potential post-fire shutdown actions that were not already modeled in the PBRA-L1P model. All the adjusted post-fire HEP values were applied in the final fire-induced risk quantification.

5.3.11 Seismic-Fire Interactions (Task 13)

The seismic and fire interactions are treated in a separate analysis and not part of the fire PSA.

5.3.12 Fire Level 1 PSA Quantification (Task 14)

The development of a fire PSA requires the integration of the fire risk model with the damage consequences calculated for each scenario. The fire risk quantification is typically an iterative process. As various analysis refinement strategies are developed, they are incorporated into the fire risk model.

The scope of work for fire quantification involves the use of the fire PSA model, described in Section 5.3.4, to perform quantifications for the purposes of obtaining SCDF estimates for each of the fire scenarios.

The scoping fire modeling (Section 5.3.7) provided a conservative and simplified means to develop an initial refinement to the bounding treatment in the quantitative screening (Section 5.3.6). The scope of work for this task involves the use of the fire PSA model with the adjusted post-fire HEPs (Section 5.3.10) and the performance of quantifications of a new set of CCDP cases refined specifically for this task for purposes of obtaining SCDF estimates. In the quantitative screening (Section 5.3.6), the SCDF estimates were done at the PAU level. In the final quantification, information gathered during walkdowns conducted for scoping modelling and additional analysis of other PNGS-B design inputs (e.g., equipment and cable tray layout drawings) was used to refine treatment of PAUs that had high estimated SCDFs in initial bounding assessment (Section 5.3.6). This refinement typically divided risk significant PAUs into multiple fire IEs (scenarios) to represent individual fire ignition sources. In some cases, multiple fire ignition sources in a PAU were grouped and treated as a single fire IE so long as such grouping did not result in overly conservative risk estimates.

The SCDF contribution from the PAUs that were screened out as part of quantitative screening analysis was included in the final fire-induced SCDF estimate.

5.3.13 Uncertainty and Sensitivity Analysis (Task 15)

The development of a risk assessment inherently results in the introduction of uncertainty in the analysis results. In general, the sources of uncertainty for each of the fire PSA development tasks are discussed in the industry reference document [R14]. The treatment of uncertainty and sensitivity is primarily limited to those fire scenarios where the refinements described in Tasks 8 through 12 were applied, because these scenarios would have been significant risk contributors. Other fire scenarios that were subjected to less refinements are expected to maintain a degree of conservatism so that their treatment would more closely resemble that of an 'upper bound' analysis.

Parametric uncertainty analysis, sensitivity analysis, importance analysis and a cliff-edge effect analysis were performed. The parametric uncertainty analysis and importance analysis for human actions, components, and systems are consistent with that performed for the PBRA-L1P.

5.3.14 Fire PSA Documentation (Task 16)

A fire PSA requires proper documentation to allow review of the fire PSA development and its results to provide a basis for any future uses of the fire PSA which is a standard practice.

5.3.15 Level 2 Analysis (Task 17)

This task is built on the results of the Level 1 quantification to consider the Level 2 impacts of fire scenarios in terms of LRF. If scenarios were identified in the Level 1 that would affect multiple units such as fires impacting the MCR or fire impacting common systems and/or common cable trays, then the multi-unit impact on Level 2 functions was quantified. More details on this task can be found in Section 6.9.

5.3.16 Alternate Unit Analysis (Task 18)

The scope of work resulted in specific numerical results for the Unit 5 PAUs and other site PAUs that are common to all four units. Quantification of separate SCDFs and release frequencies for Units 6, 7, and 8 are not specifically included. Because fire risk characterization is needed for the entire plant site, the anticipated symmetry / consistency in the design and construction of the entire four-unit site is being relied upon to support a qualitative approach to the Alternate Unit Assessment task.

A side-by-side comparison of the Unit 6, 7 and 8 PAUs to the analyzed Unit 5 PAUs was created using fire zone information from the FSSA, the FHA, and Tasks 1 and 6. Equipment layout drawings and general arrangement drawings were also consulted. A walkdown was performed with the initial release of the fire PSA in 2012 to assess the differences between the units. A confirmatory walkdown was performed in 2022 for the three top PAUs that were identified as dominant risk contributors to confirm the differences between the units. The walkdowns confirmed the physical differences between the units are relatively minor.

5.4 At-Power Internal Flood

The OPG Internal Flooding PSA Guide describes the methodology used to quantify the risk due to internal flooding. Similar to the fire PSA, the guide prescribes using a two phased approach. If the results of the first phase are satisfactory, then only the first phase is implemented. For PNGS-B, a Phase 2 Flood PSA was not required.

Like the fire PSA described in Section 5.3, the impacts of internal flooding events are related to the physical location of equipment in the plant. The station must be divided into areas, and the potential initiators in each area assessed, and the impacts of the initiators determined.

The PBRA-FLOOD analysis is focused primarily on the pinch point areas (areas with credited Group 1 and Group 2 equipment), as these areas represent the highest potential for risk significant internal floods. Bounding assessments on the areas of the plant where flooding may fail the whole of Group 1 or Group 2 are also included within the analysis. Areas of the plant where portions of credited Group 1 or Group 2 equipment may be exposed to flooding events are also reviewed. The impact of flooding events on Phase 1 EME deployment is also considered. PBRA-L1P is used to determine which components need to be evaluated for flooding impacts and is also used as the basis for the quantification of the internal flooding SCDF.

The construction of the Internal Flood PSA requires the following tasks, which are also shown in Figure 10:

Task 1 - Identification of Flood Areas and Systems Structures and Components (SSCs)

Task 2 - Identification of Flood Sources

Task 3 - Plant Walkdowns

Task 4 - Internal Flood Qualitative Screening

Task 5 - Potential Flood Scenario Characterization

Task 6 - Internal Flooding Initiating Event Frequency Estimation

Task 7 - Flood Consequence Analysis

Task 8 - Evaluate Flood Mitigation Strategies

Task 9 - PSA Modelling of Flood Scenarios

Task 10 - Level 1 Internal Flood PSA Quantification

The flooding PSA focuses on sequences that lead to severe core damage (FDC1 and FDC2) caused by an internal flood. Failure to shutdown sequences (FDC1) are not quantified as the frequency of FDC1 is several orders of magnitude lower than FDC2 in the PBRA-L1P model (see Section 7.1) and the potential for flooding events to adversely affect the shutdown systems, which fail safe on loss of power or loss of actuation inputs, is minimal.

5.4.1 Identification of Flood Areas, and SSCs (Task 1)

Like the fire PSA, the first step of the flooding PSA is to partition the plant into the flood areas that will form the basis of the analysis. As part of this task the flood areas were defined based on physical barriers, mitigation features, and propagation pathways. The flood areas were defined based on the partitions in the FSSA.

Once the flood areas were defined, the SSCs in each flood area modelled by the internal event PSA were identified.

For the PBRA-FLOOD model, once the flood areas were identified, they were screened using qualitative screening criteria as described in the following section. After the initial screening, those unscreened areas were reviewed for the impact on equipment credited in the PSA, and the possible flood sources in the area.

5.4.2 Identification of Flood Sources (Task 2)

This task identifies the potential flood sources in the plant and includes the following sub-tasks:

- Identify or confirm flood sources in each flood area (flood source is one basis for flood area screening);
- Determine or confirm flooding mechanisms associated with each flood source;
- Determine or confirm the characteristics of each flooding mechanism (flood characteristics are another basis for flood area screening);
- Identify or confirm drains and sumps of each flood area and determine capacity of the mitigation features (mitigation features are on basis for flood area screening); and
- Identify flood propagation paths.

5.4.3 Plant Walkdowns (Task 3)

This task supports the other flooding PSA tasks by identifying or confirming plant data by observing it at the plant during walkdowns.

5.4.4 Internal Flood Qualitative Screening (Task 4)

This step performs a qualitative screening considering the sources of flooding, the flood propagation pathways and the consequences of the flood. The objective is to qualitatively screen out low risk internal flood scenarios.

To identify the flood areas that can be screened out, the following items are considered:

- Screening criteria for flood areas:

  - The area contains flooding mitigation systems capable of preventing unacceptable flood levels, and the nature of the flood does not cause equipment failure;
  - A technical basis is provided to justify that the mitigation systems credited for screening out flood areas have sufficient capability;
  - The nature of flood does not cause the failure of the mitigating systems or any other equipment required to prevent core damage for flood initiated sequences; and
  - There are no propagation pathways to other flood areas.
- Screening criteria for flood sources:
  - The flood source is insufficient to cause failure of the equipment; or
  - The area flooding mitigation systems are capable of preventing unacceptable flood levels and the nature of the flood does not cause equipment failure through other failure mechanisms; or
  - The flood only affects the system that is the flood source and the Internal Events PSA already addresses this type of failure; or
  - Mitigating human actions are shown to be effective.

5.4.5 Potential Flood Scenario Characterization (Task 5)

This step identifies and characterizes the potential flood scenarios to be included in the analysis. This task characterizes the consequences for each flood-induced IE by considering the following factors:

- The specific flood area, flood source, flood source failure mode and associated magnitude.
- The type of flood failure mechanism.
- The consequences of the flood.
  - Flood propagation, if any;
  - SSCs damaged by the flood; and
  - The IE for the purpose of formulating event sequences leading to severe core damage. The IE could be the direct consequence of the flood or an immediate plant shutdown which could trigger an adverse event sequence.
- Operator and mitigation system responses to terminate the flood, limit damage to SSCs and to recover the plant from the effects of the flood event;
- The means to be used to define the interface with the Internal Events PSA model for calculating the probability that the flood leads to severe core damage.

5.4.6 Internal Flooding Initiating Event Frequency Estimation (Task 6)

This step identifies flood-induced IEs and estimates their frequency of occurrence. The flooding failure rates are based on generic EPRI data from Reference [R15].

5.4.7 Flood Consequence Analysis (Task 7)

The characterization of the consequences for each flood-induced IE includes consideration of the type of flood sources, flood propagation paths, plant mitigating features, and equipment susceptibility to flood.

5.4.8 Flood Mitigation Strategies (Task 8)

This step is to identify and evaluate the strategies that can be employed by plant operators to mitigate the consequences of the flood. These actions can include terminating the source of the flood by isolating the break, or stopping the pumps that supply the flood source, or opening doors to divert water away from sensitive equipment.

The evaluation of human failure events in the internal flood scenarios differs from the internal events PSA. Specifically, the appropriate scenario-specific impacts on Performance Shaping Factors (PSFs) were considered for both control room and ex-control room actions based on the following items:

- Additional workload and stress (above that for similar sequences not caused by internal floods);
- Availability of indications;
- Effect of flood on mitigation, required response, timing, and recovery activities (e.g., accessibility restrictions, possibility of physical harm); and
- Flooding-specific job aids and training (e.g., procedures, training exercises).

5.4.9 Accident Sequence Modelling (Task 9)

This step includes the finalization of flood scenario development and completing internal flood accident sequence models based on modifying the Internal Events PSA model. The PBRA-FLOOD model is based on small ETs for each flooding scenario. These ETs model the possible mitigating actions described in Section 5.4.8. Based on success or failure of the mitigating actions equipment availability is determined.

5.4.10 Level 1 PSA Quantification (Task 10)

Following the completion of the ET analysis, the next step is to construct an integrated PSA model to evaluate the risk from internal flooding. To quantify the internal at-power flood model, new flooding events are added to the existing integrated loop cut internal events model and this is integrated with the high level logic developed from the flood specific ETs.

Parametric uncertainty, sensitivity and importance measure analyses were included as part of the quantification of the PBRA-FLOOD model. As well, an LRF estimate from the internal flooding events was produced.

5.5 At-Power Seismic

The PBRA-SEISMIC assessment has been developed following the methodology for preparation of a PSA-based Seismic Margin Assessment (SMA) as described in the OPG Seismic PSA Guide. The major activities of the PSA-based SMA methodology and its application in the development of the PBRA-SEISMIC assessment are summarized in the subsections below.

The primary steps in developing the PSA-based SMA are identifying the seismic hazard at the site, constructing an ET and FT model of the plant to represent the credited heat sinks following a seismic event, and creating new equipment failure modes based on the likelihood of equipment failure due to the seismic event. The PSA-based SMA is created based on the internal events At-Power PSA, PBRA-L1P.

The PBRA-SEISMIC model considers sequences that result in severe core damage (FDC1 and FDC2). Like the Flood PSA, FDC1 sequences (failure to shutdown the reactor) are not assessed following a seismic event. Failure to shutdown following a seismic event is highly unlikely as SDS2 is seismically qualified, and selective components of the SDS1 system (mainly the shutoff rods) are seismically qualified. The two shutdown systems are highly reliable, and both have a fail safe design.

Similar to the Fire and Flood studies, the Seismic PSA Guide also outlines a Phased approach with two phases defined:

- **Phase 1 - PSA-based Seismic Margin Assessment** - In Phase 1, a Probabilistic Safety Assessment-based Seismic Margin Assessment (PSA based SMA) is performed based on the methodology described in NUREG/CR-4482 [R16]. This focused approach uses a plant model based on PBRA-L1P with the addition of new seismic failure modes; the new seismic failure events are developed from a seismic margin approach with generic variabilities and the time average seismic risk is calculated in terms of a point estimate of SCDF that does not include a full uncertainty analysis.
- **Phase 2 - Seismic PSA (SPSA)** – In Phase 2, significant conservatisms due to simplifying assumptions in the Phase 1 analysis are reduced, where feasible, in an effort to increase the computed seismic capacity of plant SSCs included in the seismic model; computed fragilities are refined for risk-significant SSCs based on the refined analysis; and the seismic model is extended as needed to provide a reasonably realistic estimate of plant seismic SCDF and allow a quantitative evaluation of significant uncertainties.

For PNGS-B, a Phase 1 PSA-based SMA study was performed, and the results showed that there was no need to transition into Phase 2.

Major elements of the PNGS-B PSA-based SMA consist of the following tasks:

Task 1 - Seismic Hazard Characterization

Task 2 - Plant Logic Model Development

Task 3 - Seismic Response Characterization

Task 4 - Plant Walkdown and Screening Reviews

Task 5 - Seismic Fragility Development

Task 6 - Seismic Risk Quantification

The integration of these tasks is shown in Figure 11.

In addition to the above tasks, the impact of seismically-induced internal fires and seismically-induced internal floods on seismic risk at PNGS-B has been evaluated qualitatively, considering potential significant sources at the station.

5.5.1 Seismic Hazard Characterization (Task 1)

The first step in the PSA-based SMA is to model the site-specific seismic hazard. The seismic hazard is a representation of the possible earthquakes and seismic activity that can be experienced at the site. The seismic hazard is a plot of the peak ground acceleration versus the annual frequency that the ground acceleration will be exceeded (typically described as the frequency of exceedance). Figure 12 shows a typical seismic hazard curve. The curve shows that very small ground accelerations are more likely than very large ground accelerations.

The site-specific seismic hazard curve is used to define the earthquake characteristics used in the PSA-based SMA analysis. The earthquake ground motion under analysis is greater than the seismic design of the plant to understand the plant capacity to survive a beyond design basis earthquake. The beyond design basis earthquake under consideration is referred to as the Review Level Earthquake (RLE) or Reference Earthquake (RE).

5.5.2 Plant Logic Model Development (Task 2)

This task involves two related but separate subtasks: development of the high-level plant logic for the risk quantification model, and development of the seismic equipment list (SEL), which lists the SSCs credited in the PSA-based SMA. This task relies upon the Internal Events PSA and other safe shutdown analysis to define the functions and SSCs required to mitigate seismic IE.

In this study, the seismic ET has been developed only for the Level 1 aspect of PSA, whereas the development of the seismic equipment list is applicable to both Level 1 and Level 2 PSA aspects. An ET is not needed for the Level 2 portion of this study as the robustness of containment is assessed using a simplified approach.

5.5.3 Seismic Response Characterization (Task 3)

The next step is to characterize how the site structures respond to a seismic event. The response of the building will not be the same on each elevation. For example, the small earthquakes occasionally experienced in Southern Ontario are typically undetectable to people in the basement or lower floors of buildings, but can be easily detected by people in the higher floors of tall buildings.

The ground oscillation of any seismic event can be described by a combination of ground motion frequencies. This is called the spectrum of the seismic event. Each potential seismic event may have a different spectrum. The different frequencies in an earthquake’s spectrum will be transferred to the site structures in different ways. The response of site buildings determines how the earthquake will affect the credited equipment in the PSA-based SMA and is used to calculate the probability of equipment failure due to a seismic event.

In Phase 1, a generalized scaling approach is used to calculate the response of the site structures. This method is based on the existing design basis earthquake seismic response analysis for the site structures, prepared as part of the Pickering A Seismic Assessment performed between 1996 and 1998, with updates to reflect the shapes of the new seismic hazard curves. In addition to characterizing the overall building response, this task defines the local accelerations for the credited equipment.

5.5.4 Plant Walkdown and Screening Reviews (Task 4)

Plant walkdowns were required to assess the relative vulnerability of equipment to seismic challenges. The walkdowns were performed by fragility experts in order to document the basis for screening equipment in (based on susceptibility) or out (based on ruggedness) of the PSA-based SMA. The plant walkdowns included taking photos and recording observations of the SEL items which were located in accessible areas. The screening level chosen needs to be high enough such that the contribution from screened-out SSCs is not significant to overall seismic risk. In addition, equipment required for crediting EME was also assessed during the walkdown.

5.5.5 Seismic Fragility Development (Task 5)

The likelihood that a given piece of equipment will fail for a given seismic hazard is based on the fragility of the equipment. The fragility of the equipment is a conditional failure probability that the equipment will fail when subjected to a specific acceleration caused by a seismic event. The likelihood the equipment will fail increases as it is subject to greater acceleration. Figure 13 shows an example fragility curve. Figure 13 shows that if the example equipment is subject to an acceleration of 1g, the failure probability is 80%.

The fragility analysis conducted for a PSA-based SMA is limited to that of the Conservative Deterministic Failure Margin (CDFM) whereby the seismic capacity is calculated in terms of a High Confidence of Low Probability of Failure (HCLPF) value using a generic representation of the variability [R17].

5.5.6 Seismic Risk Quantification (Task 6)

The seismic risk is evaluated using a PSA-based SMA model. To build the PSA-based SMA model, the information on the seismic response of the buildings and the seismic fragility of the equipment is used to calculate the probability of equipment failures. The development of the model involves the following key steps:

1. Make high level logic using the seismic ET developed in the task Plant Logic Development.
2. Prepare the FT where the high level FT logic prepared in Step 1 is populated with mitigating and support system FT logic by merging it with the at-power internal events PBRA FT model.
3. Prepare the database, which is comprised of data from the PBRA at-power internal events database and augmented with seismically-induced failure modes.
4. Evaluate and post-process the model.

The model solution generated cutsets. seismic cutsets were evaluated to obtain the plant-level HCLPF. The plant-level HCLPF forms the basis of the fragility curve for the station. The plant-level fragility is convolved with the hazard curve for the station to obtain a mean point estimate of the seismic contribution to SCDF. Non-seismic cutsets, representing random failures of credited system, were also considered in the determination of SCDF following a seismic event, in the same manner as they were for internal events PSAs. A discrete approximation of the hazard curve is made by dividing the hazard curve into nine bins as shown in Table 17. Qualitative uncertainty, sensitivity, importance, and cliff-edge effects analyses were carried out for SCDF estimation.

5.6 High Wind Safety Assessment

The PBRA-WIND assessment has been developed following the methodology for preparation of a high wind PSA as described in the OPG High Wind Hazard PSA Guide. The major activities of the high wind PSA methodology and its application in the development of the PBRA-WIND assessment are summarized in the subsections below.

The primary steps in developing the high wind PSA are identifying the high wind hazard, identifying the high wind targets, developing windborne missile fragilities for the high wind targets, evaluating the pressure fragilities of the high wind targets, developing the high-level plant logic, and quantifying the high wind scenarios. The high wind PSA model is created based on the Internal Events At-Power PSA model, PBRA-L1P.

The methodology for high wind hazard assessment follows six main tasks as listed below:

Task 1 - High Wind Hazard Analysis

Task 2 - Analysis of Windborne Missile Risk

Task 3 - High Wind Fragility and Combined Fragilities

Task 4 - Plant Logic Model Development

Task 5 - Plant Response Model Quantification

Task 6 - Estimation of High Wind Large Release Frequency

In addition, an important support task is the plant walkdown. A walkdown provides information on the special layout of the site, structural vulnerability and to determine the potential for missile generation. Several site walkdowns of the Pickering site were conducted in 2012, 2013, and 2014 as part of the development of the earlier versions of the PNGS A and B high wind PSA studies. A walkdown was completed in October 2021 in support of the 2022 PBRA-WIND update.

Figure 14 shows the relationship between all tasks of the high wind PSA.

The methodology applied in this high wind hazard assessment uses a high-level approach in determining fragilities based on the reference wind speed. The approach used is realistic with conservative assumptions to simplify the analysis where needed.

5.6.1 High Wind Hazard Analysis (Task 1)

Three types of extreme winds have been analyzed for PNGS:

1. Tornadoes
2. Thunderstorm Winds
3. Non-Thunderstorm Winds (Extratropical Storms)

In accordance with the High Wind PSA Guide, the hurricane wind hazard is not modeled because the Pickering site is not subject to hazardous hurricane winds.

The tornado risk analysis methodology uses a statistical approach that considers both broad regions and small areas around the plant. Tornado hazard curves are developed using TORRISK2. TORRISK2 is a specialized tornado simulation version of TORMIS developed for hazard curve production for targets of various sizes, including individual building and plant areas.

The thunderstorm wind speed hazard curves were developed using a stochastic modeling approach where the maximum gust wind speed recorded on each thunder day is used to develop a distribution of thunderstorm wind gusts given the occurrence of a thunder day.

The high wind hazard analysis is used as the basis for defining a set of high wind initiating events for the high wind plant response model. The separate curves for tornadoes and straight winds were discretized into several subintervals of wind speeds for each hazard type. Initiating events were defined for the subintervals of each hazard type and were inserted into the high wind plant response model.

5.6.2 Analysis of Windborne Missile Risk (Task 2)

The purpose of this task is to develop windborne missile fragilities for the plant targets. Windborne missile fragility is defined as the probability of target damage (failure) from windborne missiles for a given value of the reference wind speed used in the hazard analysis.

Windborne missile risk includes:

1. Flying missiles that hit/damage an exterior target.
2. Flying missiles that enter a building and hit an interior target.
3. Flying missiles that originate within a building and hit an interior target.

A list of high wind targets is generated under Task 4: Plant Logic Model Development. The missile risk is derived based on missile sources, plant layout, and plant design information taking into account applicable uncertainties.

Windborne missile risk does not include consideration of failure or collapse of a building onto its own interiors. Windborne missile risk does not include collapse of stacks or other tall structures onto plant SSCs. These failure modes are considered structural interactions and are modelled as part of wind pressure fragility analysis.

Fragility functions are developed for each SSC subject to windborne missile risk. Interior SSCs in highly vulnerable structures may be represented by a single fragility function that does not separately consider missiles, provided the building failure is judged to occur prior to (or simultaneously with) the initiation of significant missile hazard at the site.

The EPRI-developed TORMIS methodology is utilised to estimate the probability of tornado missile impact and damage to nuclear power plant structures and components [R18], [R19].

5.6.3 High Wind Fragility and Combined Fragility Analysis (Task 3)

The purpose of this task is to evaluate the fragility of buildings and/or High Wind Equipment List (HWEL) SSCs due to high wind pressure effects, including the aerodynamic forces produced by the dynamic pressure component of the wind flow and where appropriate, Atmospheric Pressure Change (APC) that may occur from the low central pressure region within tornadoes. This task includes the combination of the various wind failure modes. However, it does not complete the integration of wind and windborne missile fragilities derived in Task 2: Windborne Missile Risk and Task 3: High Wind Fragility Evaluation as these two wind failure modes are combined within the PSA model.

The list of high wind targets is screened based on system dependencies to obtain a sub-list of targets bounding mitigating safety systems fragility. Wind capacity calculations are completed to obtain the median wind capacity and associated uncertainties of these targets based on available design information, National Building codes and walkdown observations. The generated wind capacity and uncertainty values are used to derive the wind fragility curves.

5.6.4 Plant Logic Model Development (Task 4)

This task addresses the identification of high wind targets and development of the high-level plant logic for high wind PSA model. This high-level logic, in turn, forms the basis for the SSCs to be credited for the various high wind scenarios. The high wind plant logic model examines the response of plant SSCs to the defined high wind hazard, and then combines this response with the response of the plant to the resulting IE, given the degraded condition of plant SSCs due to the hazard. The focus of the high wind analysis is estimation of SCDF for a single reference unit, with common unit and adjacent unit impacts on the reference unit considered.

5.6.5 Plant Response Model Quantification (Task 5)

This task is performed to finalize and quantify the high wind scenarios developed by modifying the integrated severe core damage (FDC2) model of PBRA-L1P. This task involves the integration of the high wind hazard and fragility information with the overall plant PSA logic model. This involves linking the fragility information to appropriate sequences and basic events in the plant logic model. The high wind hazard curve used in the high wind hazard characterization is then integrated with the plant logic model containing the fragility information to determine high wind risk in terms of Severe Core Damage. In addition to providing the overall frequencies for each sequence, this quantification identified dominant accident sequences, component failures, and human actions with respect to high wind risk.

The quantification of high wind accident sequence frequencies again requires first quantifying the frequency of occurrence of each initiating event and the logic models developed to represent the failure probabilities of the event tree top events.

The event tree top event failure probability model includes not only the impact of wind speed on plant failure probabilities, but also of random failures unrelated to the wind speed. The high wind initiating event frequencies and event tree top event probabilities were then combined similar to the approaches followed for non-high wind initiating events. By combining the frequencies of high wind sequences over all high wind initiating events, the end state frequencies for high wind risk were determined.

The parametric uncertainty analyses are carried out for the SCDF result. A number of sensitivity analysis cases are performed to examine the risk significance of key assumptions from the high wind plant logic model development and quantification on the FDC2 results.

Sensitivity cases were selected in part based on a review of assumptions with a potential for a large increase in SCDF given a small change to the assumption; that is, with the potential for a cliff-edge effect.

5.6.6 Estimation of High Wind Large Release Frequency (Task 6)

The purpose of this task is to estimate the High Wind LRF based on the results from Level 1 High Wind PSA. In OPG's PSAs for external events, e.g., high winds and seismic events, it has been a common practice to assume that all units at a multiunit station are perfectly correlated, i.e., the accident sequence is exactly the same in all affected units. More details on this task can be found in section 6.12.

6. LEVEL 2 PSA METHODS

Section 5 described the methods used for the Level 1 PSA assessments of PNGS-B. In the Level 1 PSA, the goal is to quantify the frequency of fuel damage. Once the fuel has been damaged, there is the potential for radioactive material to be released from the fuel into containment. The Pickering B NGS design includes a containment system (described in Section 2.12.3) to prevent the release of any radioactive material in the station from being discharged into the environment.

The Level 2 PSA studies the system failures and accident phenomena that might result in a release to the environment, and the timing and magnitude of the release. This information is combined with the PBRA-L1P model to quantify the frequency of possible releases.

The PBRA-L2P model has been developed following the methodology for preparation of a Level-2 PSA. The consequence assessment is performed by simulating the accident sequences using the MAAP5-CANDU version 5.00a code. The major activities of the Level-2 PSA methodology and its application in the development of the PBRA-L2P are summarized in the subsections below.

6.1 Interface with Level 1 PSA

The PBRA-L1P generates results in the form of frequencies of nine FDCs, described in Section 5.1.2, representing a wide range of possible outcomes. The possible outcomes include the most severe involving failure to shutdown (FDC1) to relatively benign where release is limited to the equilibrium fission product inventory of the Heat Transport System (HTS) (FDC9). A subset of the FDCs (1-7), those that involve release of significant quantities of fission products from the core, is used to develop the interface between Level 1 and Level 2, the Plant Damage States (PDSs). The PDSs serve to reduce number of the sequences assessed in the Level 2 analysis to a manageable number while still reflecting the full range of possible accident sequences and their impacts on the plant.

Only two FDCs are used to represent the range of sequences that result in severe core damage, FDC1 for rapid accident progression resulting from failures to shut down the reactor when required, and FDC2 for all other sequences.

FDC1 is conservatively assumed to cause early consequential containment failure and is assigned to a unique PDS, PDS1. FDC2 is not assumed to result in immediate containment failure and is subdivided into three PDSs (2-4) to examine the potential for random and consequential failures of containment systems that could eventually lead to enhanced release to the environment:

- PDS2 represents sequences affecting a single unit with release into containment;
- PDS3 represents sequences affecting more than one unit; and
- PDS4 represents single unit sequences with a release pathway that bypasses containment.

Random containment system failures are associated only with PDS2 and were identified by means of a Bridging ET (Figure 15) that led to the creation of eight subcategories, labelled PDS2A-H.

As described in Section 1, Unit 5 is the reference unit for the PSA Study. In order to develop the logic for PDS3, conservative assumptions were made to partition the FDC2 logic into sequences that impact a single unit, and sequences that could impact more than one unit.

FDCs 3-7 represent the range of accidents that fall under the general heading of “design basis events”. These were allocated to PDS5 and 6 respectively, depending on whether the IE involves containment bypass (PDS6) or not (PDS5).

FDCs 8-9 are excluded from Level 2 analysis on the basis that the radionuclide releases from these in-plant sequences would be negligible.

For Level 2 analysis, the characteristics of each PDS are represented by a single representative accident sequence. By design, the plant damage states group sequences expected to generate similar magnitude and timing of fission product release to containment and containment response. However, the frequency and releases for each sequence will vary to some extent.

The Level 1 PSA is used to identify IEs that are significant contributors to the frequency of the plant damage state. These sequences are then reviewed to select a representative sequence that bounds the consequence. The hybrid approach follows the guidance of the IAEA as this method selects a sequence that "largely bounds" the PDS. The representative sequences chosen for each PDS are summarized in Table 18.

6.2 Containment Event Tree Analysis

In Level 2 PSAs, Containment Event Trees (CETs) are used to delineate the sequence of events and severe accident phenomena after the onset of core damage that challenge successive barriers to radioactive release to the environment. They provide a structured approach for the evaluation of the capability of a plant, specifically its containment boundary, to cope with severe core damage accidents. The entry points into the CETs are the plant damage states that involve severe core damage.

A CET is a logic model that addresses uncertainties in the ability to predict the potential impacts of accident progression and associated physical phenomena on containment response. Figure 16 shows a generic CET. CET top events are not built from system based "success criteria" but from questions that are intended to ascertain the magnitude of phenomenological challenges to the integrity of the containment boundary and the potential for radionuclide release to the environment during the various stages of accident progression (e.g., "Is containment integrity maintained?" or "What is the Extent of Core Concrete Interactions?"). The CET branch points represent major events in accident progression and the potential for fission product release to the environment. The CET also represents the evolution of the progression with time so the same nodal question may appear more than once in the tree as conditions inside containment change. The focus of the CET is to estimate the probabilities of the various ways that containment failure may occur leading to a release of radionuclides to the environment.

Most of the CET branch points represent alternative possible outcomes of a given physical interaction. Depending on the availability of suitable models and data for a given physical interaction or phenomenon, the methods of branch point quantification can vary. The acceptability of these probability estimates is supported via an expert review process.

6.3 Containment Fault Trees

Containment system FTs are required for quantification of the frequencies of the end-states PDS2A – PDS2H in the Level 1/Level 2 PDS2 bridging event tree, which is shown in Figure 15, and includes the following branch headers:

CEI:	Impairment of Containment Integrity Avoided
PRV:	Pressure Relief Valves (PRV) Open to Limit Containment Pressure for LOCA Events
ACU:	Boiler Room and Fueling Machine Vault Air Conditioning Units (ACUs)
IGN:	FMV Ignitors Operate and FMV ACUs Mix Atmosphere

The FT models used in the quantification of the Level 2 PSA are listed in Table 13. FT representations for failure of these containment functions have been developed, reflecting the likelihood that random equipment failure or human error will prevent the operation of the system on demand or during the mission. Containment failures arising as a consequence of severe accident progression are addressed in the CET.

6.4 Release Categorization

The CET analysis generates a multitude of end states associated with each specific severe accident sequence. The CET end states are binned into Release Categories (RCs), to facilitate comparison with safety goals (Table 1) and for use in subsequent applications. The RCs are defined based on two criteria:

- The magnitude of release in Becquerel (Bq) of specific radionuclides considered important to offsite impacts (e.g., isotopes of cesium or iodine); and
- The timing of the release, either early in the accident sequence (where “early” is less than 24 hours) or late (after 24 hours).

Seven RCs cover the full range of possible releases and provide sufficient delineation to evaluate safety goal frequencies. An eighth category is used to represent basemat melt-through, when the core debris is postulated to penetrate the floor of the reactor vault. Table 19 presents the release categories used in the PBRA-L2P analysis. LRF is defined to be the sum of RC1 through RC3.

6.5 MAAP-CANDU Analysis

MAAP-CANDU (Modular Accident Analysis Program – CANDU) is a severe accident simulation code for CANDU nuclear stations [R20]. It is used to calculate the consequences of severe accidents and is designated as a CANDU Owners Group (COG) Industry Standard Toolset (IST) code. MAAP-CANDU originated from MAAP developed for Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) systems by Fauske and Associates.

MAAP-CANDU can simulate the response of a CANDU power plant during severe accident sequences. The code quantitatively simulates the evolution of a severe accident starting from full power conditions given a set of system faults and IEs through events such as primary heat transport system failure, core melt, calandria vessel failure, calandria vault failure, and containment failure.

Severe accident analysis carried out using MAAP-CANDU is the cornerstone of the Level 2 PSA. There are at least five distinct roles for the code, as outlined below:

- To establish the baseline accident progression for each plant damage state and the potential impact of associated physical phenomena on CET top events;
- To determine the sensitivity of phenomena to reasonable variations in key parameter values to support CET branch point quantification;
- To calculate releases to the environment for those sequences for which a non-zero probability of a containment failure mode has been estimated to support categorization of releases;
- To generate results to support systematic sensitivity and uncertainty analysis; and
- To provide information related to plant environmental conditions.

6.6 Severe Accident Management Guidelines

SAMG are entered when plant conditions reach the point where actions being attempted using AIM procedures and/or EME guidelines are no longer effective and severe core damage is considered imminent. The goals of SAMG are to terminate fission product releases from the plant, maintain or return containment to a controlled, stable state, and return the core to a controlled and stable state.

SAMG documentation is treated as guidance, compared to AIM response, which uses procedures. The type of actions included in SAMG range from recovery of systems typical in the prevention of severe core damage (i.e., ECI, moderator cooling) to crediting systems or injections lineups in non-traditional ways that are not typically included in the AIM response.

While Phase 1 EME is used prior to the entry into SAMG as a prevention mechanism, it can also be used within the SAMG framework if not successful in preventing severe core damage.

Credit for SAMG actions has been incorporated into the Level 2 PSA model.

6.7 Integration of the Level 1 and 2 PSA

The purpose of integration is to link the Level 1 ETs with the PDSs via the Level 1/Level 2 bridging event tree and containment FTs and then with the RCs via the CET end-states using the results of the branch point quantification. The product is a complete set of sequences that contribute to each RC, from which the frequency of each RC can be determined.

Importance analysis is performed to identify the dominant contributors to each release category.

Sensitivity and uncertainty analysis are performed on both the frequency quantification and on the MAAP-CANDU consequence assessment.

6.8 Level 2 Outage Assessment

This assessment documents the methods and results of a reduced scope consequence assessment for a limited number of representative sequences occurring during each of Pickering B shutdown POSs. The focus of this reduced scope Level 2 outage analysis is on modeling of accident progression and source term estimation.

The goals of the reduced scope Level 2 outage PSA are as follows:

1. Determine if severe accidents while in a shutdown POS progress more slowly than severe accidents in high power units. If this is the case, then the risk from a multi-unit event occurring while a single unit is operating in a shutdown POS is driven by the transients in the high power units.
2. Determine if severe accidents while in a shutdown POS pose unique challenges to the containment boundary. If no unique challenges are identified, then it is reasonable to assume that the LRF for a shutdown POS will be much lower than the already extremely low shutdown state SCDF.

The consequence assessment is performed by simulating the accident sequences using the MAAP-CANDU 5.00a code.

A LRF bounding estimate for PNGS-B Outage Internal Events PSA has been performed in 2022.

6.9 Level 2 Fire Assessment

This assessment is built on the results of the Level 1 quantification (See Section 5.3.12) to consider the Level 2 impacts of fire scenarios in terms of LRF. If scenarios were identified in the Level 1 that would affect multiple units such as fires impacting the MCR or fire impacting common systems and/or common cable trays, then the multi-unit impact on Level 2 functions was quantified.

The approach for the treatment of Level 2 consisted of three steps. The first two steps involved a screening process. The objective of these screening steps is to identify and exclude those fire IEs that represent a negligible contribution to the overall plant risk. The overall fire PSA development is based on having divided the plant into multiple PAUs. Within each PAU, the fire ignition sources are identified and addressed resulting in a number of individual fire IEs (scenarios).

Those fire scenarios that remained after the screening process were subjected to the third step, an assessment of the impact of the fire scenario on containment and the application of modification factors to generate an estimate of the LRF. Containment failure can be prevented given SCD in a single unit if the accident progression is terminated in the calandria with injection of EME makeup to the calandria; this is referred to as late In-Vessel Retention (IVR). The impact of each of the remaining scenarios on EME is assessed for the potential to support IVR.

For single-unit scenarios assessed to have no impact on containment due to successful IVR, an additional factor representing Conditional Containment Failure Probability is applied to the scenario frequencies to determine their contribution to the LRF. For single-unit scenarios assessed to create a degraded containment condition, or a complete failure of containment, the scenario SCDF is reported without adjustment as the LRF. For scenarios causing the shutdown of two or more units, the SCDF is also reported without adjustment as the LRF irrespective of the impact of the fire itself on the containment systems.

Qualitative assessment on uncertainty, sensitivity, and importance analyses were carried out for the LRF estimation.

6.10 Level 2 Flood Assessment

The LRF is estimated based on a bounding estimate of PBRA-FLOOD using SCDF cutsets manipulations and insights from PBRA-L2P.

To estimate LRF due to internal flooding, the cutsets were classified into one of the following categories:

- Cutsets involving single unit with flooding event inside the Reactor Building (RB);
- Cutsets involving single unit with flooding event outside RB;
- Cutsets involving more than one unit which will be referred to as Multi-Unit.

Cutset manipulations were performed to determine the fraction of each type of sequence that progresses to a large release. The sum of the contribution from each group is then used to estimate LRF caused by internal flooding.

Qualitative assessment on uncertainty, sensitivity, and importance analyses were carried out for the LRF estimation.

6.11 Level 2 Seismic Assessment

The Level 2 seismic assessment is intended to characterize the likelihood of a significant radioactivity release given a seismically-induced severe core damage event. It is conservatively assumed that the seismic event will impact similarly all reactors on the PNGS site. It is known that the Pickering NGS ‘B’ containment is not designed to mitigate the overpressure transient created by such a scenario. Therefore, even a rigorous seismic PSA model would likely show that the LRF is practically equal to the SCDF. Given these insights, the LRF is assumed to be equal to SCDF and no detailed calculation is performed.

In accordance with the OPG Seismic PSA Guide, an alternative analysis of containment robustness in response to seismic event is performed. This is done based on fragilities of the containment SSCs using an additional Level 2 metric - the seismically-induced containment failure frequency (SCFF). Additional walkdowns and fragility calculations, using the same techniques as those described in Section 5.5.5, were used to assess the possible failure of containment due to seismic events.

HCLPF values for containment structures, systems and components were evaluated to determine the limiting HCLPF for containment integrity. A convolution of the plant-level limiting containment fragility with the hazard curve for the station produced the PNGS-B SCFF. The SCFF is estimated to be 4.6E-08 per year over the range of events up to and including those with recurrence frequency of 1E-04 per year. This demonstrates that the containment is structurally robust and can survive earthquakes significantly stronger than the DBE.

6.12 Level 2 High Wind Assessment

A Level 2 High Wind assessment is performed to estimate the LRF based on the results from Level 1 High Wind PSA. In OPG's PSAs for external events, e.g., high winds and seismic events, it has been a common practice to assume that all units at a multi-unit station are perfectly correlated, i.e., the accident sequence is exactly the same in all affected units.

The methodology involves manual investigation of the high wind FDC2 cutsets and use of insights from the full scope PBRA-L2P study. The following are considered:

- The number of units involved in the accident;
- The potential for consequential containment failure after both single and multi-unit accidents, e.g., combustion of hydrogen; and
- The potential for random containment failure, e.g., random containment envelop impairment.

The parametric uncertainty analyses are carried out for the LRF estimate. A number of sensitivity analysis cases are performed to assess the impact of uncertainties in other key assumptions in the PBRA-Wind plant response model development on LRF results.

6.13 Non-Reactor Source PSA

While the hazard screening analysis had screened out all hazards associated with the UFDS facility, selected internal and external natural hazards for the fuel in the IFBs were screened in. Consequences of these hazards are characterized as a loss of IFB heat sink or a loss of IFB water inventory. Bounding simplified quantitative assessments were used for the following hazards:

- Small Aircraft Impact
- Rail Transportation Accident – Cold Toxic Gas Release: Chlorine, Sulphuric Acid and Sulphur Dioxide
- Ship Accident
- Earthquake
- Flooding - Due to Run Off
- Flooding - Due to Combined Events

- Meteorological Extremes - High and Low Temperature, Rainfall, Snow, Freezing Rain, Snowpack
- Meteorological - Hurricanes / Tornadoes, Ice Storms
- Mist
- White Frost
- Frazil Ice
- Geomagnetic Storm
- Bio-fouling
- Combined Hazard Events
- Random IFB Cooling System Failures
- Random IFB Support Systems Failures
- Human Errors
- Internal IFB Fires
- Internal IFB Flooding
- Reactor Hazards That May Impact IFB Cooling System Equipment Operation
- Loss of IFB Water Inventory

An assessment of interactions between accident progressions in reactor units and IFBs was also conducted. Cliff-edge effects were analyzed, and it was concluded that there were no cliff-edge effects requiring further actions to better characterize their effects or to develop mitigating actions. An LRF estimate based on the bounding simplified quantitative assessments for the hazards listed above was performed, and the results are presented in Table 25.

7. RESULTS SUMMARY AND CONCLUSIONS

7.1 Severe Core Damage and Large Release Frequencies

The PBRA study uses two measures to assess the acceptability of risk. These two measures correspond to the OPG risk-based safety goals:

- Frequency of severe core damage (SCDF); and
- Frequency of large release (LRF).

Table 20 compares the results of the Internal Events PSA studies described in Sections 5, and 6, with the OPG safety goals.

OPG has safety goals for the severe core damage and large release frequencies. The safety goal represents the tolerability of risk exposure above which action shall be taken to reduce risk.

The Internal Event PSAs assess the full range of fuel damage and release categories defined in Table 12, Table 16 and Table 19. The frequency of fuel damage for the At-Power Internal Events PSA (PBRA-L1P) is presented in Table 21. The results in Table 21 show that failure to shutdown is a negligible contributor to SCDF.

As described in Section 6.1, the fuel damage categories used as end states in the Level 1 PSA are partitioned into PDSs to use as inputs into the Level 2 PSA. Table 22 presents the frequencies of the PDSs, and Table 23 presents the results of PBRA Level 2 At Power (PBRA-L2P).

The risk results for internal fire, seismic, internal flooding, and high wind events are presented in Table 24. The fire, seismic, flood, and high winds results are all below the OPG safety goals for severe core damage and large release frequencies.

While the LRF due to a seismic event is bounded by the SCDF, the assessment of the containment fragility concluded that containment is robust and can survive earthquakes significantly stronger than the DBE.

7.2 Conclusions

The PNGS-B PSA (PBRA) complies with the CNSC Regulatory Document REGDOC-2.4.2 [R1]. The PSA addresses Level 1 and Level 2 PSA aspects for various internal and external events, both at-power and outage operating conditions, including internal events, internal fire, internal flooding, seismic, high winds, non-reactor sources. In addition, an external and internal hazard screening assessment has been performed.

As described in Section 7.1 the results of all the models prepared to meet the requirements of REGDOC-2.4.2 satisfy the OPG safety goals for severe core damage and large release frequencies, demonstrating that the overall risk to the public is low. OPG continues to meet industry best practices through periodic PSA updates to account for operating experience and changes at the station.

8. REFERENCES

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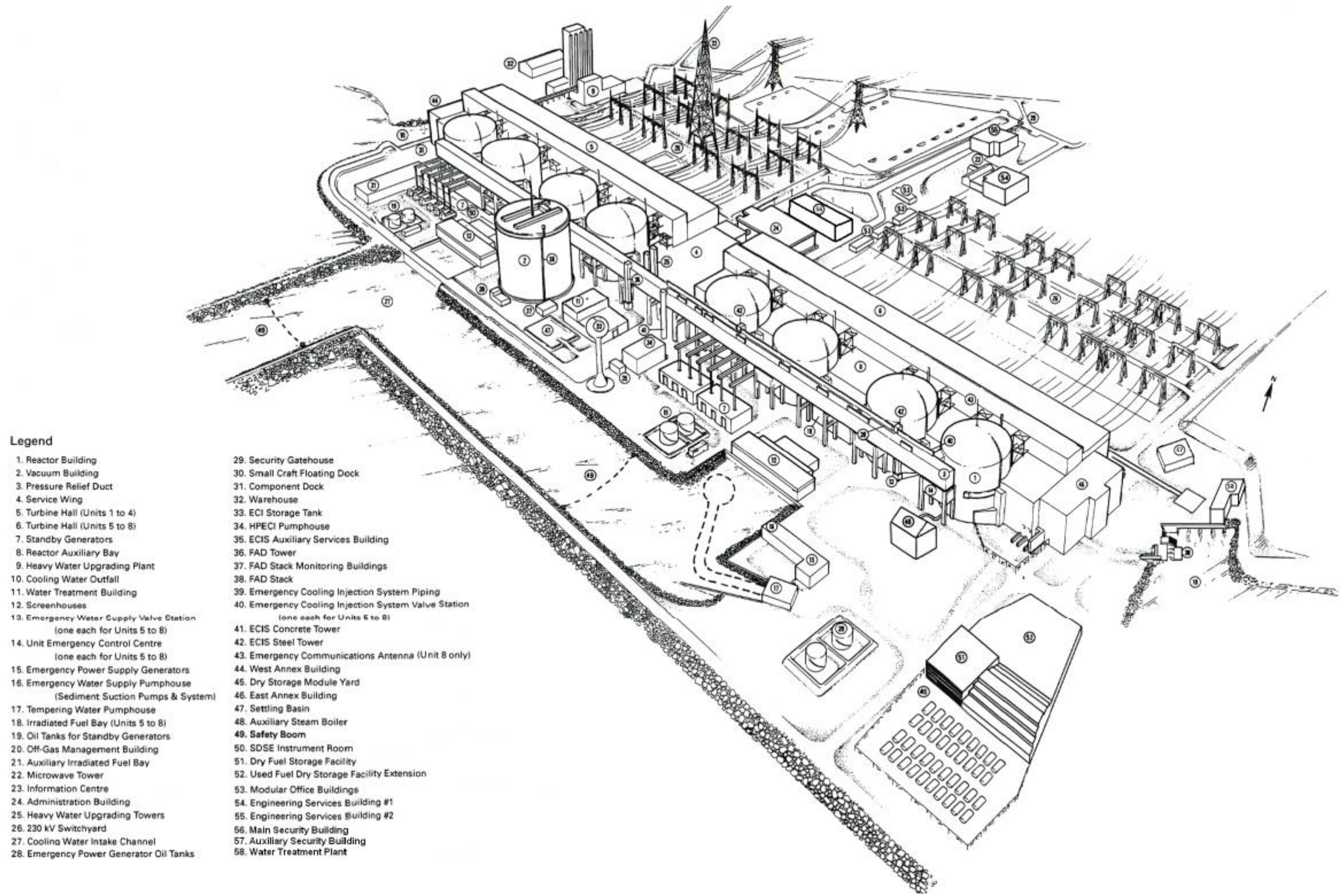

Figure 1: Pickering Site Layout

Figure 2: Aerial Photograph of Pickering Site

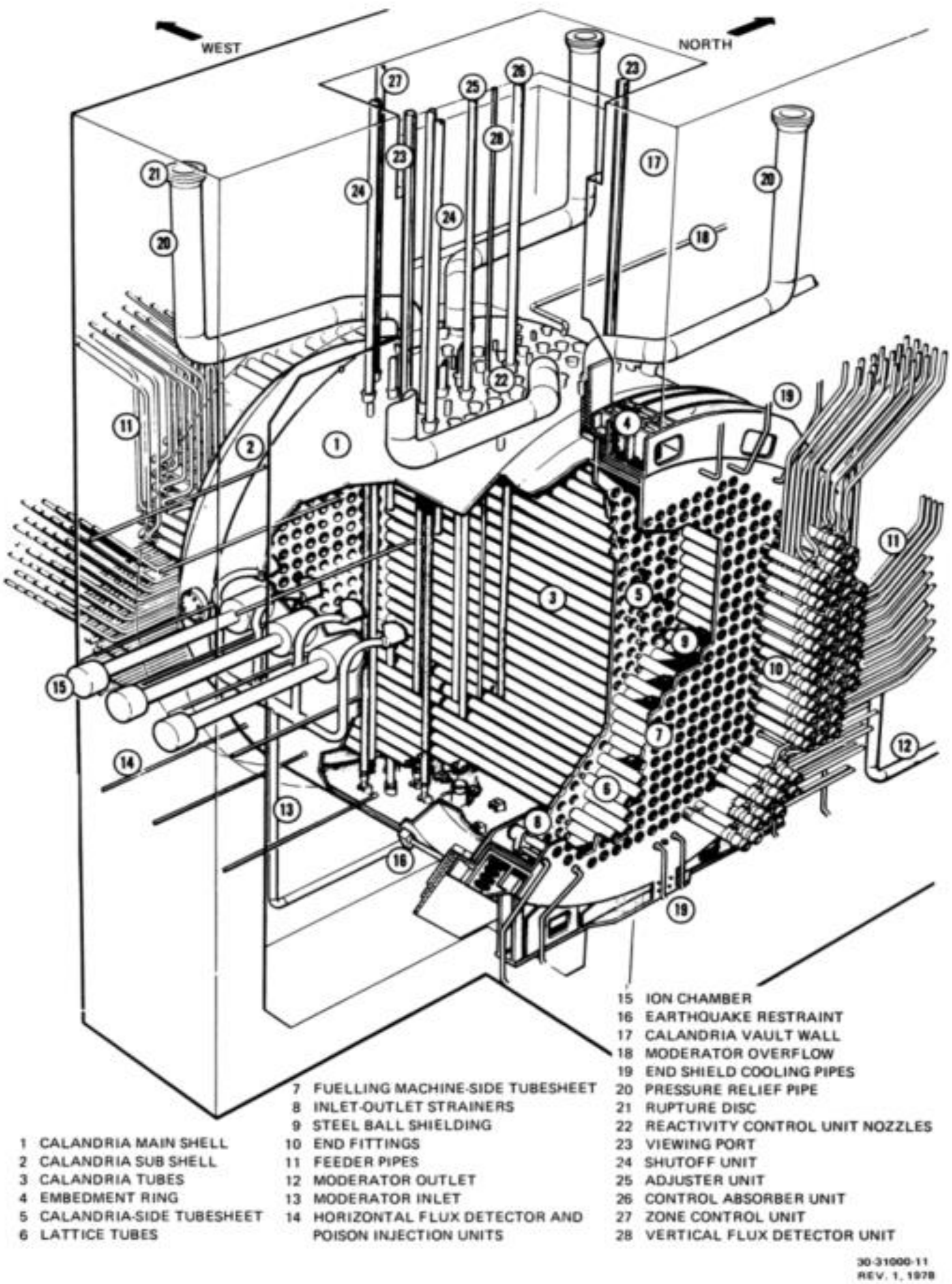

Figure 3: Typical Pickering NGS 'B' Reactor

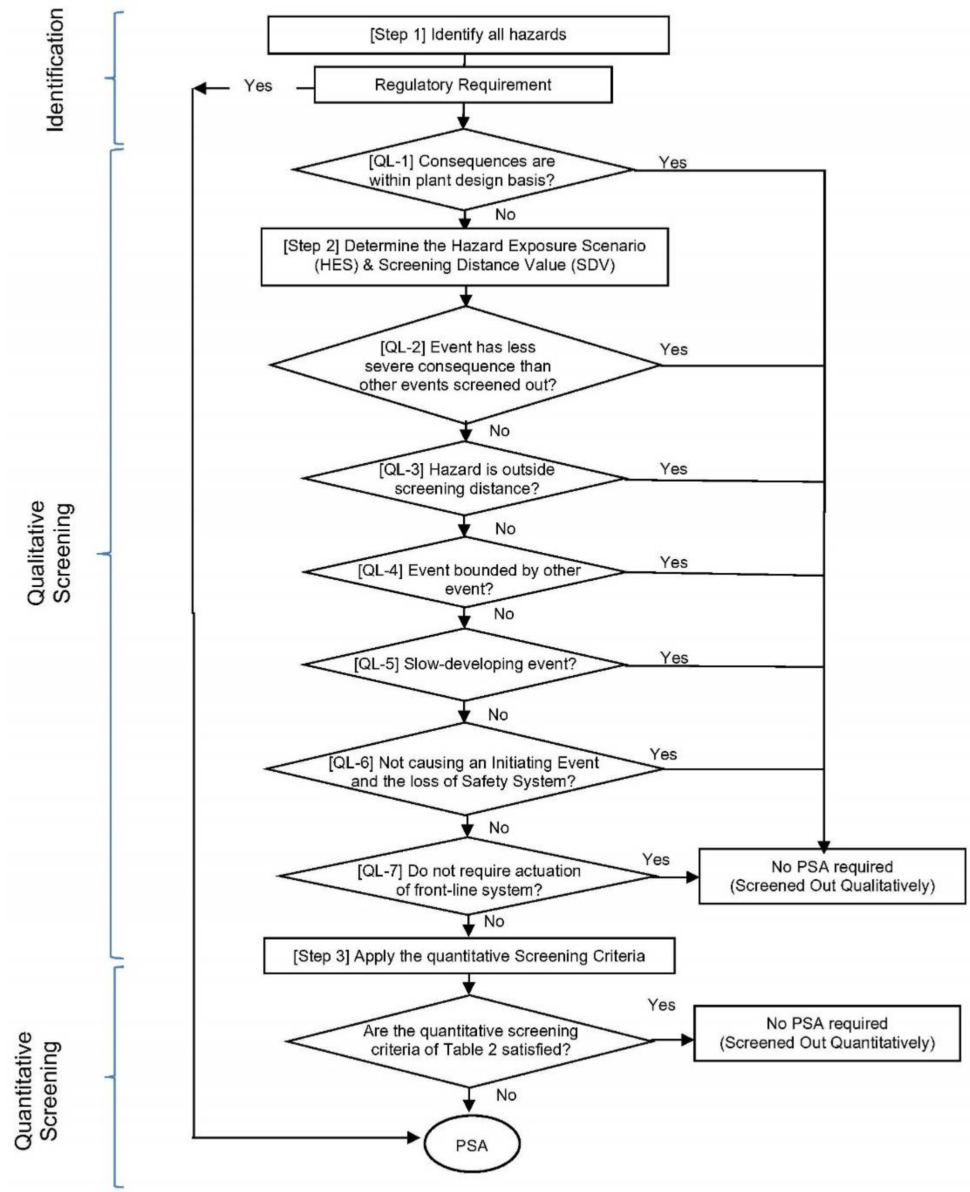

Figure 4: Hazards Analysis Steps

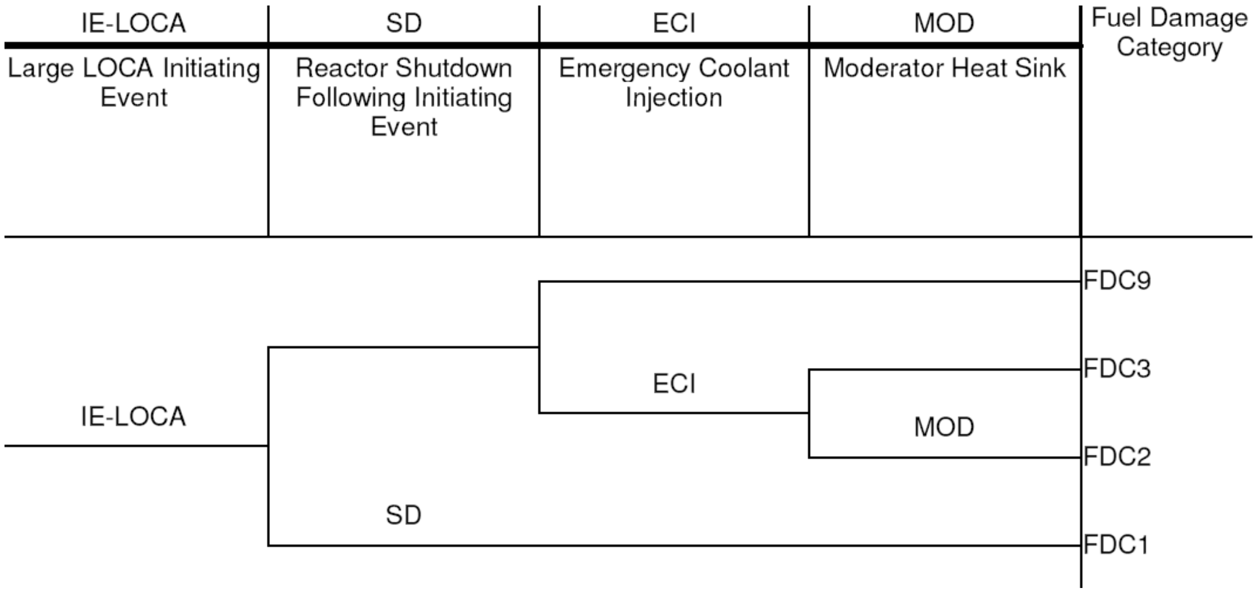

Figure 5: Example LOCA Event Tree

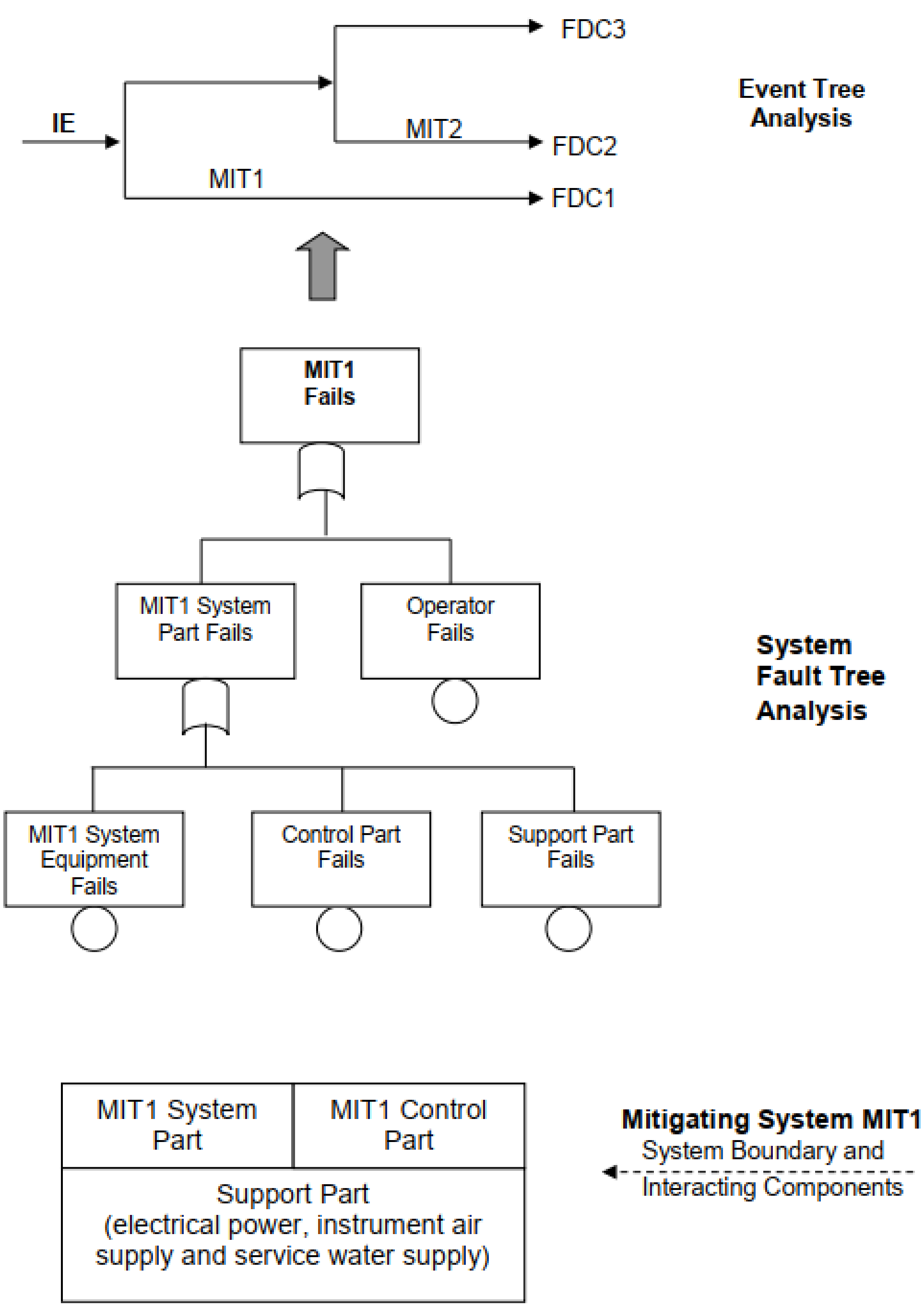

Figure 6: Fault Tree and Event Tree Integration

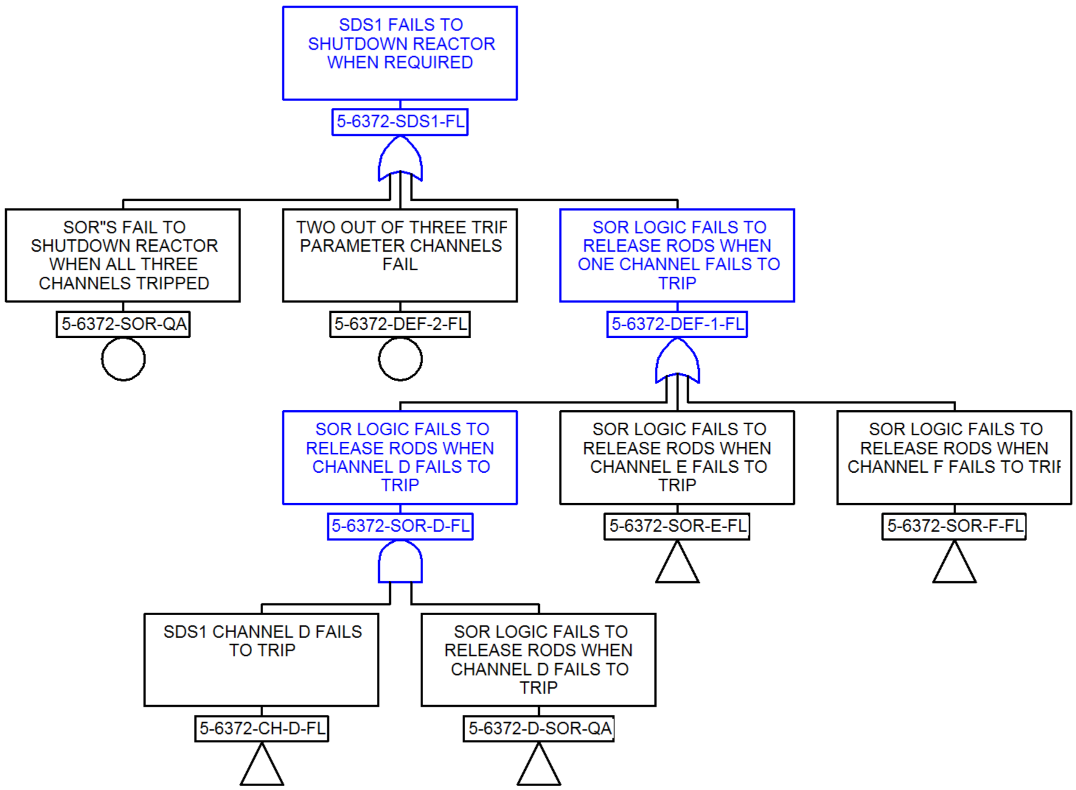

Figure 7: Example Fault Tree

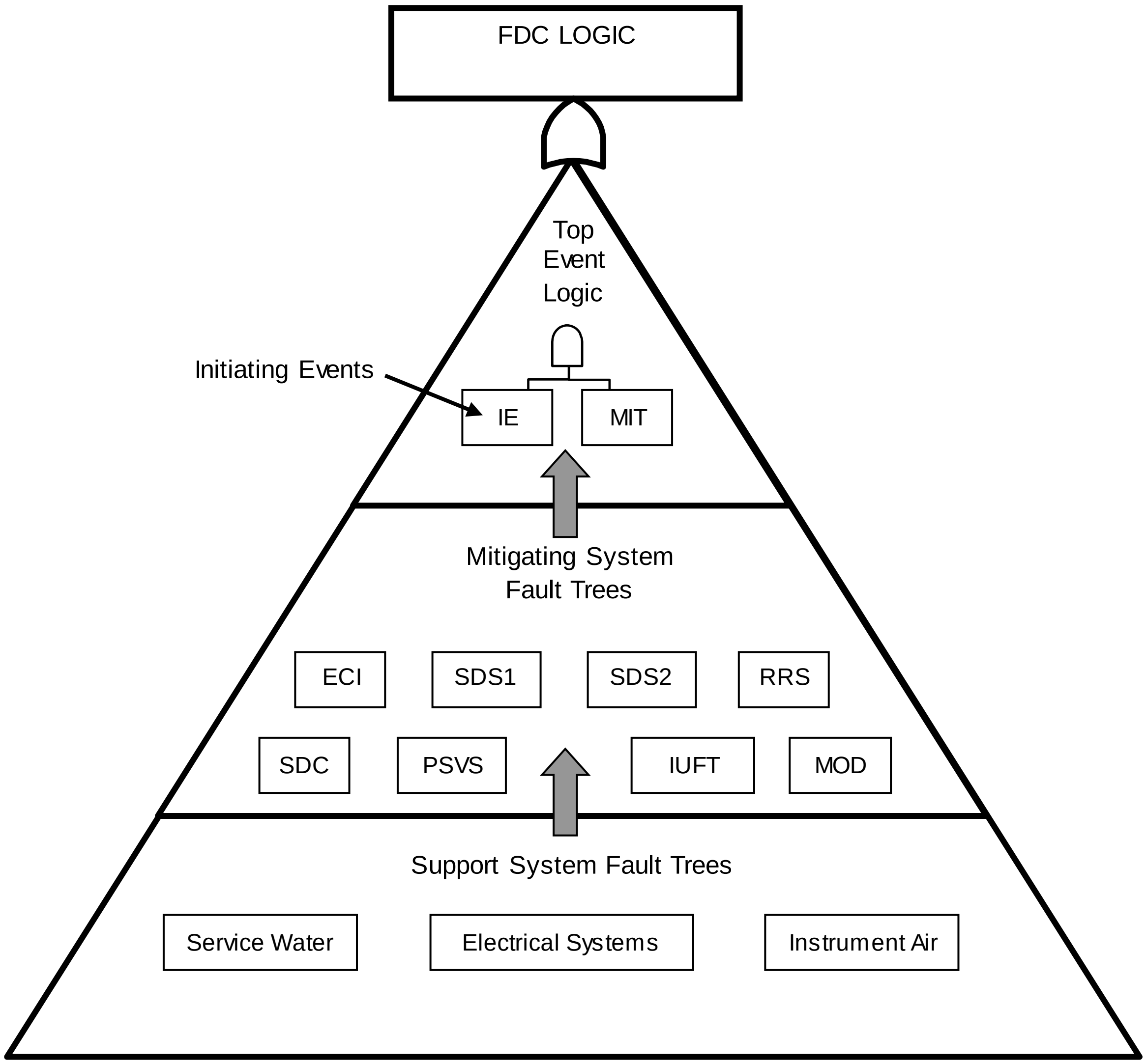

Figure 8: Fault Tree Integration

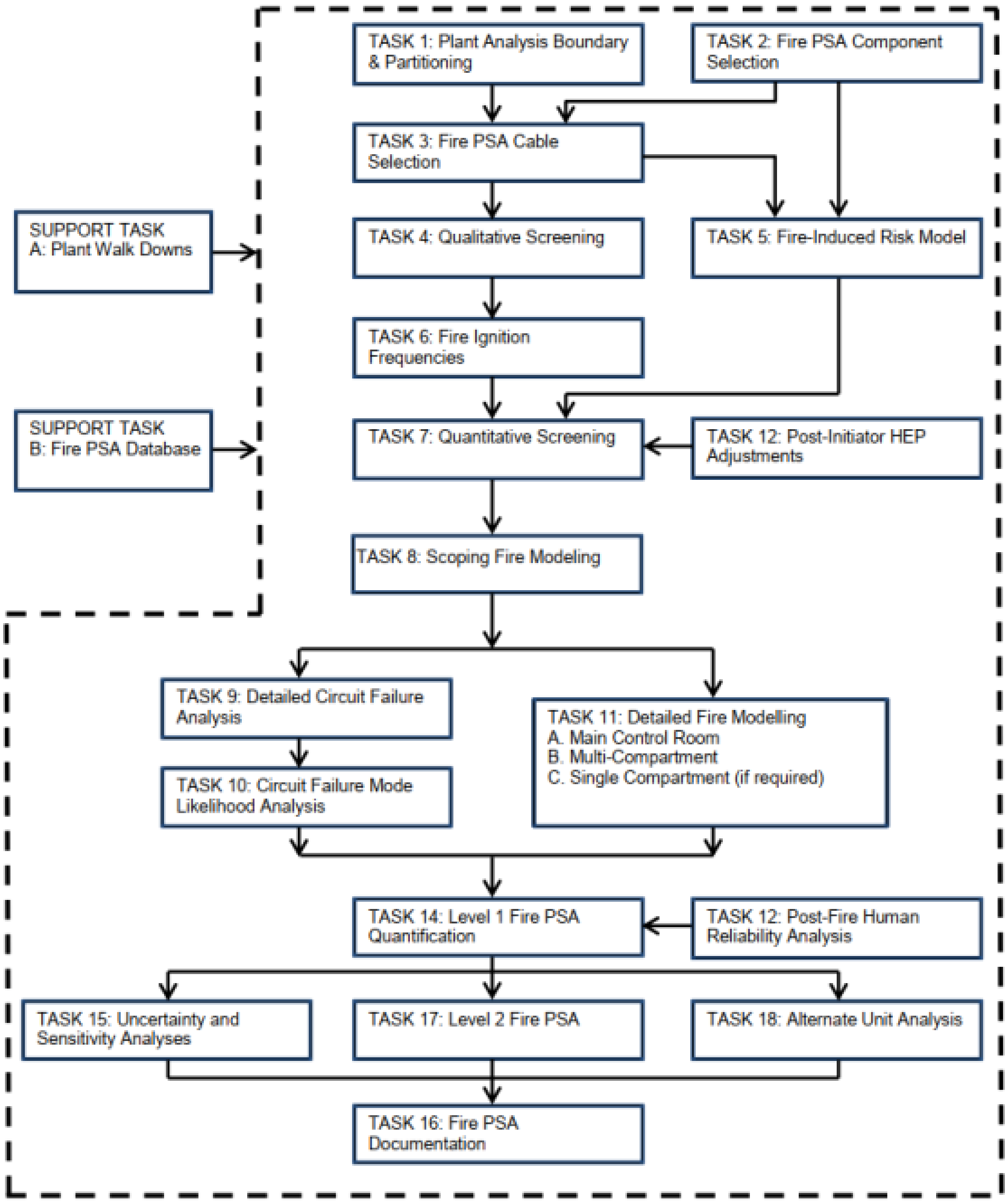

Figure 9: Internal Fire At-Power PSA Tasks

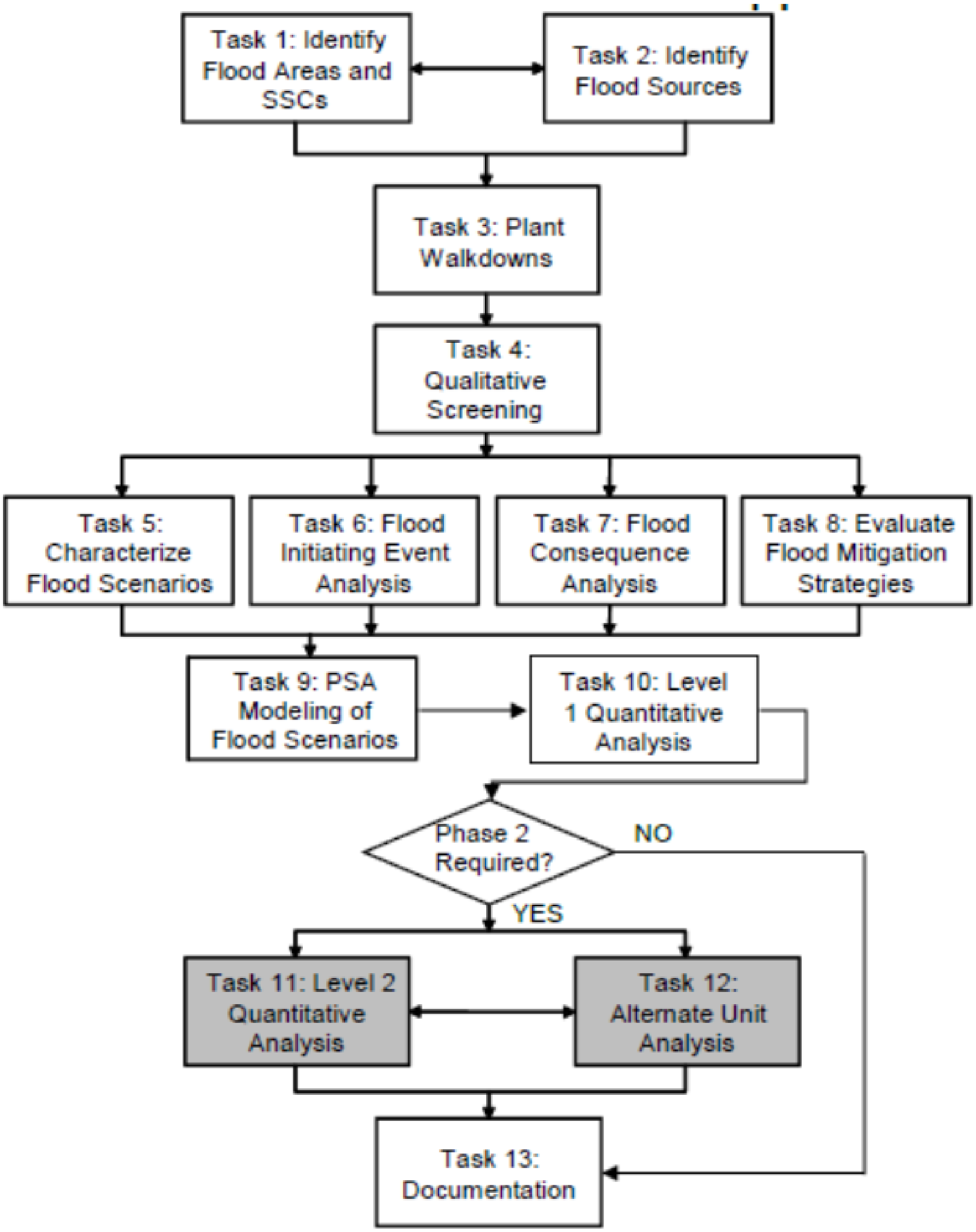

Figure 10: Internal Flood Phase 1 Tasks

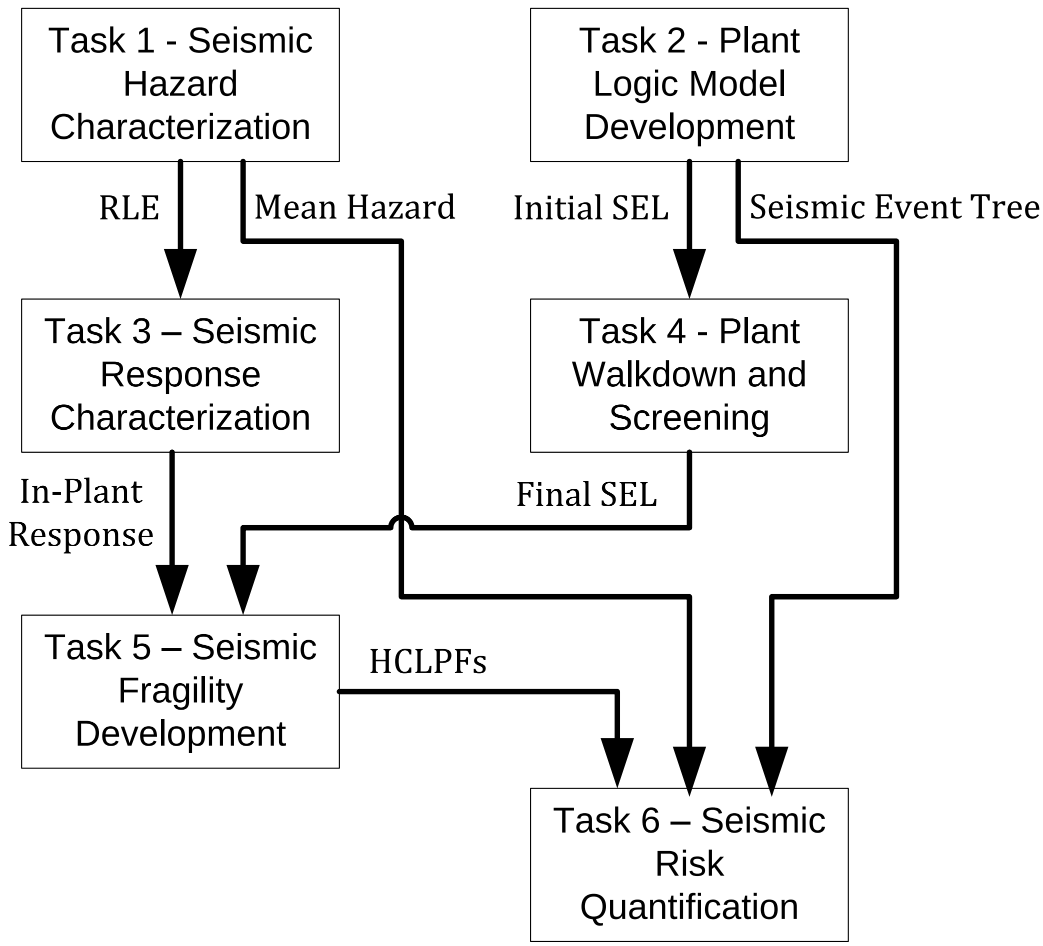

Figure 11: Analysis Tasks for Conducting the PSA-based SMA

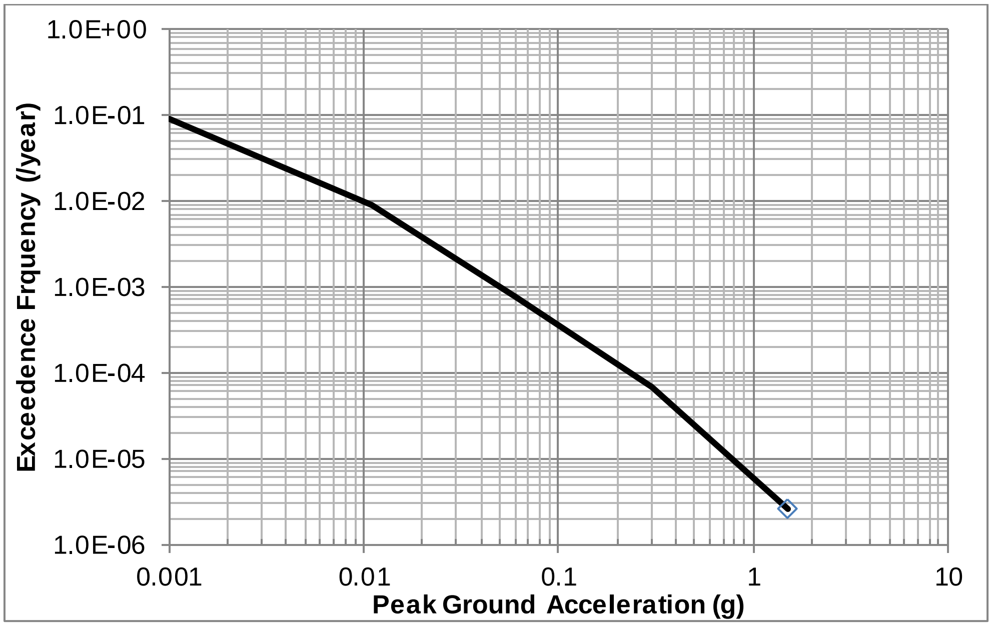

Figure 12: Example Seismic Hazard Curve

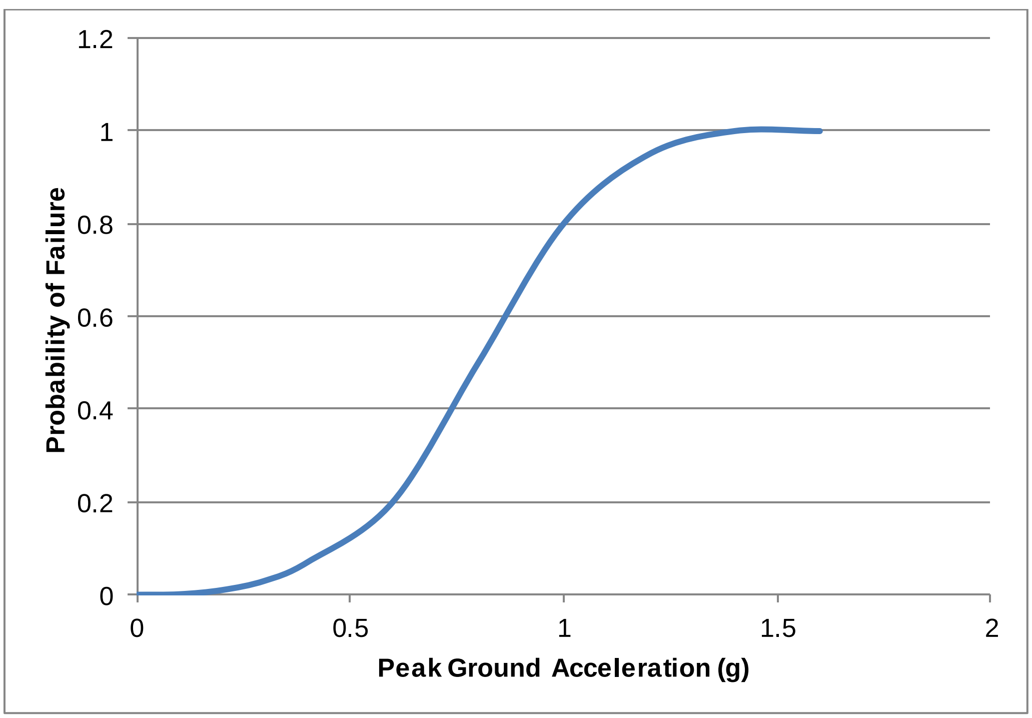

Figure 13: Example Fragility Curve

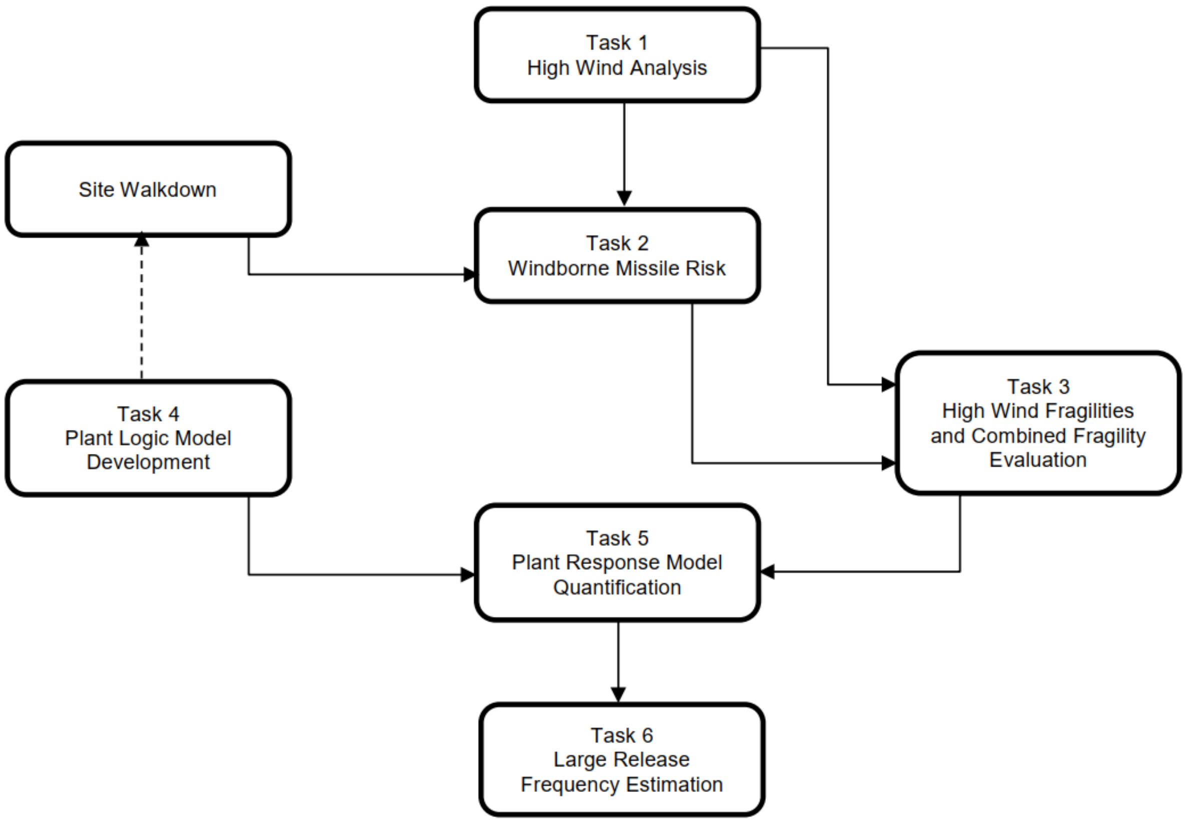

Figure 14: High Wind Hazard Assessment Overview

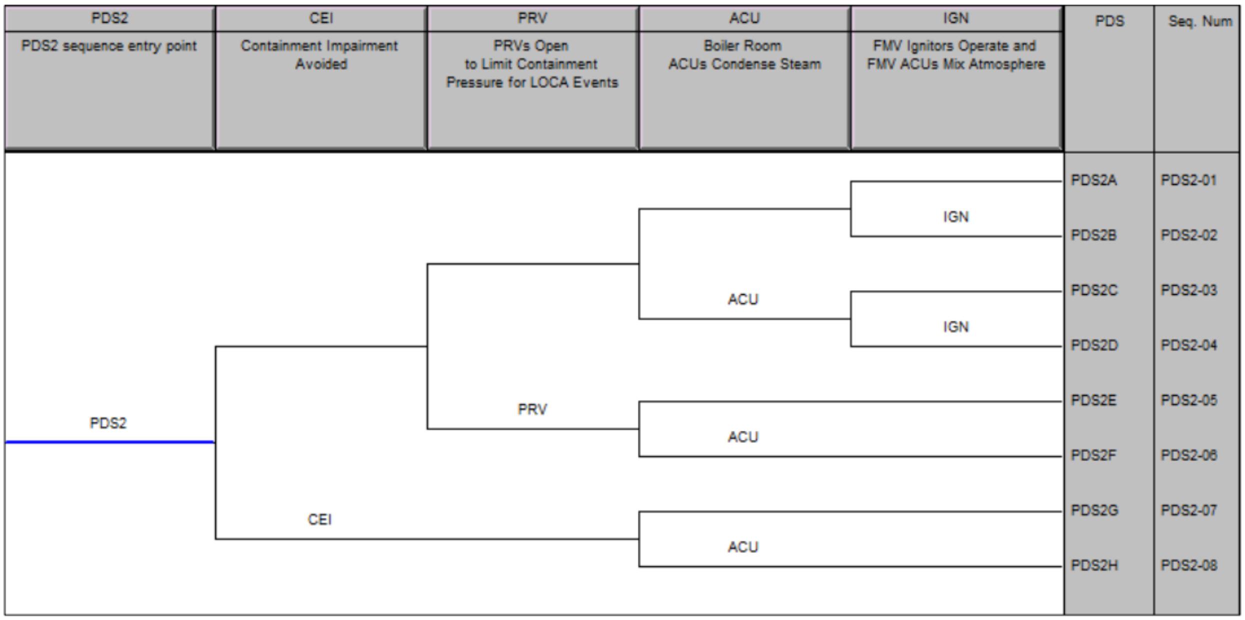

Figure 15: PNGS-B Bridging Event Tree

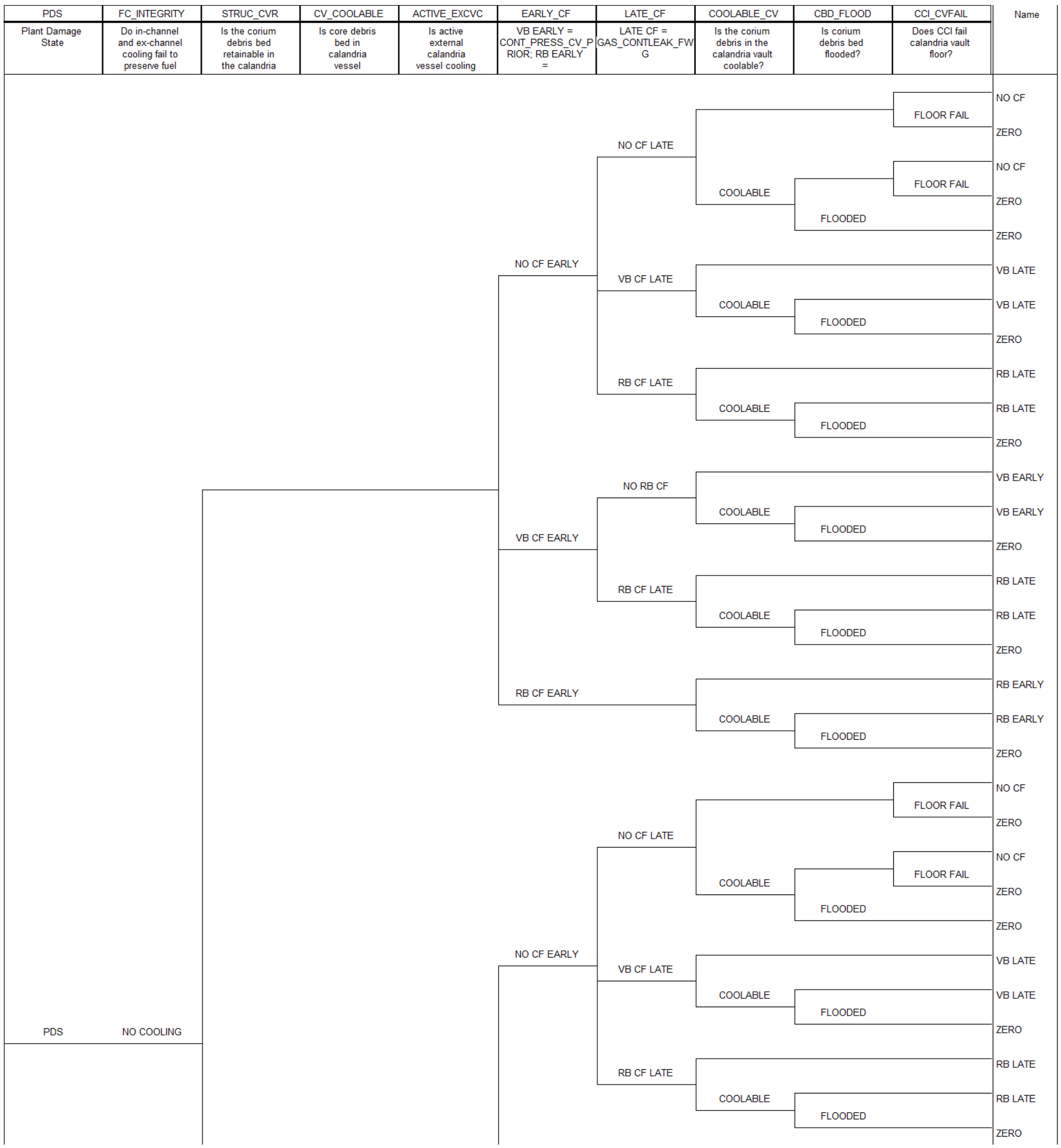

Figure 16: Generic Containment Event Tree

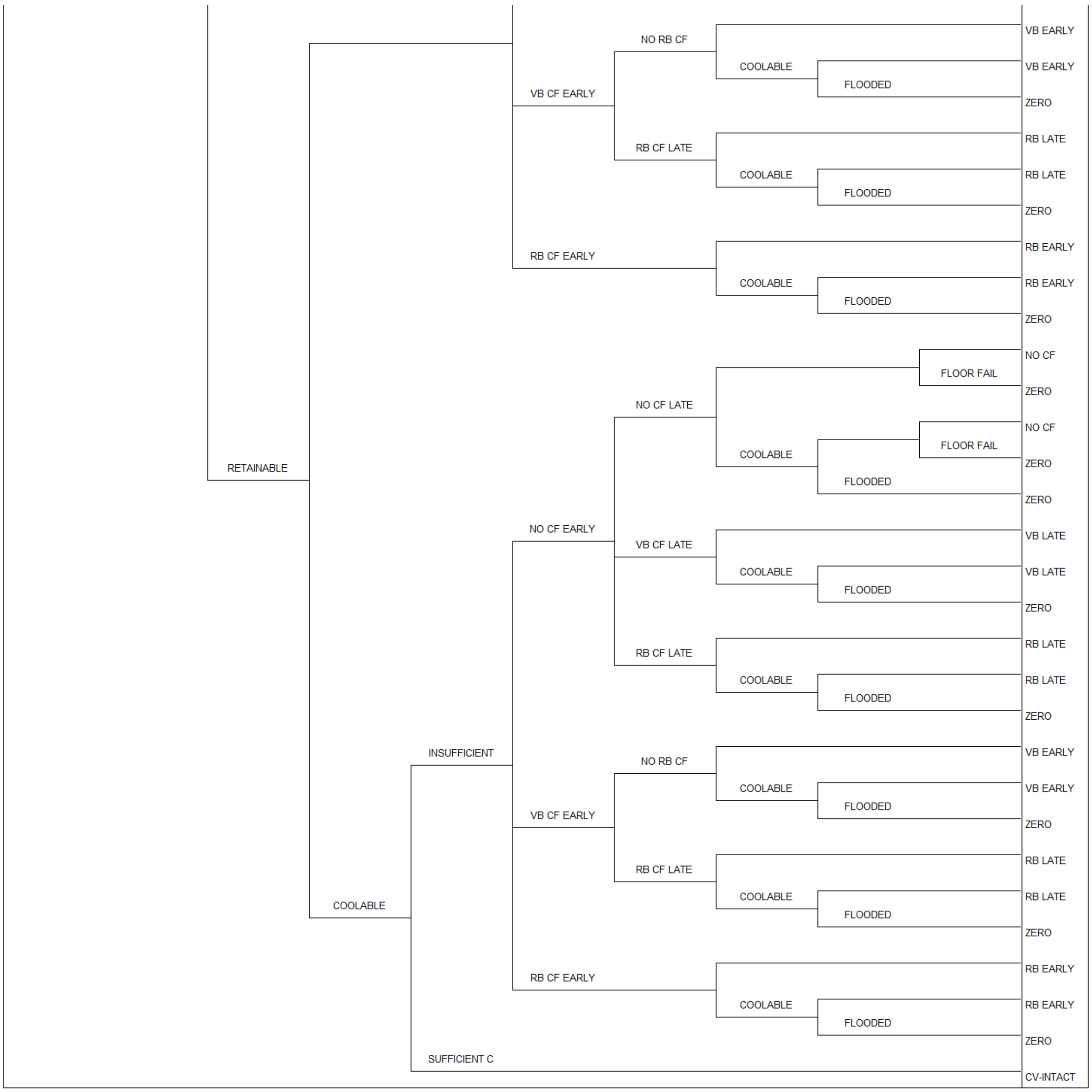

Figure 16: Generic Containment Event Tree (cont'd)

Table 1: OPG Risk Based Safety Goals

Criteria	Average Risk (per year)
Severe Core Damage[1]	10^{-4}
Large Release[2]	10^{-5}

[1] Severe Core Damage is the loss of core structural integrity.

[2] Large Release is a release greater than 1E14 Bq of Cs-137.

Table 2: Quantitative Hazard Screening Criteria

Criterion	Description (Note 1,2,3)	Direct Containment Bypass or Failure (Note 4)	Reference
QN1	SCDF < 10^{-6}/year	No	EPRI 3002005287 [R21]
QN2	Design Basis Hazard Frequency < 10^{-5} /year. and CCDP < 0.1 (Note 5)	No	EPRI 3002005287 [R21]
QN3	SCDF < 10^{-7}/year.	Yes	EPRI 3002005287 [R21]
QN4	Design Basis Hazard Frequency < 10^{-6}/year. and CCDP < 0.1 (Note 5)	Yes	EPRI 3002005287 [R21]
QN5	IE or Hazard Frequency may be screened out if it can be shown that their frequency is < 10^{-7}/year.	Not Applicable	CSA Standard N290.17-17 [R5] and IAEA NS-G-3.1 [R22] (Note 6)

Notes:

1) Similar to the ASME/ANS PRA standard, these criteria are based on a bounding or demonstrably conservative analysis.

2) The criteria in this table are nominally for plants with SCDF from all other hazards totaling ~10^{-5}/year or higher. If the SCDF from all other hazards total much less than 10^{-5}/year, then lower quantitative criteria should be considered.

3) With a cliff edge present, consider reducing the frequency of the screening criteria, such as by a factor of 10 (due to uncertainty in the hazard calculation and the absolute nature of the numeric criteria).

4) “Direct Containment Bypass or Failure” implies that the conditional large release probability (CLRP) is equal to or very close to 1.0, as a result of the hazard’s impact on the plant.

5) These criteria should not be used if potential design vulnerability is identified. The intent of the adjustments for potential design vulnerabilities is to address events whose magnitudes are less than the design basis hazard (i.e., the hazard frequency is greater) and the vulnerability may result in a CCDP that is significant, even though the event magnitude is reduced. If there is an identified design vulnerability, then only the two SCDF criteria i.e., QN1 and QN3 are recommended for quantitative screening of the hazard.

6) IAEA NS-G-3.1 [R22] includes this criteria - In some States, a value for the probability of 10^{-7} per reactor-year is used in the design of new facilities as one acceptable limit on the probability value for interacting events having serious radiological consequences, and this is considered a conservative value of SPL (Screening Probability Level) if applied to all events of the same type (such as all aircraft crashes, all explosions). Some initial events may have very low limits on their acceptable probability and should be considered in isolation.

Table 3: Summary of Criteria Applied for Screening for External Human-Induced Hazards

External Human-Induced Hazard	Screening Criterion
Small Aircraft Impact	QN1
Large Aircraft Impact	QN5
Rail Transportation – Cold Toxic Gas Release: Ammonia, Hydrogen Chloride, and Hydrogen Fluoride	QL-3
Rail Transportation – Cold Toxic Gas Release: Chlorine, Sulphuric Acid, and Sulphur Dioxide	QN1
Rail Transportation – Hot Toxic Gas Release	QL-3
Rail Transportation – BLEVEs	QL-3
Rail Transportation – Vapour Cloud Explosions	QL-3
Rail Transportation – Rail Line Blast	QL-3
Road Transportation – Cold Toxic Gas Release: Ammonia, Hydrogen Chloride, and Hydrogen Fluoride; Hot Toxic Gases, BLEVEs, VCEs, and Explosions	QL-3
Road Transportation – Cold Toxic Gas Release: Chlorine, Sulphuric Acid, and Sulphur Dioxide	QN5
Ship Accidents – Small Vessels	QL-6
Ship Accidents – Large Vessels	QL-3 and QL-4
Nearby Nuclear Event	QL-5
Fixed Sources – Toxic Gas Release: Ajax Water Treatment Plant	QL-3
Fixed Sources – Toxic Gas Release: Duffin's Creek Water Pollution Control Plant	QN1
Fixed Sources – BLEVEs	QL-3
External Fires – Including Forest Fire	QL-3
Thermal Radiation from Fire	QL-3
Orbital Debris	QN3

Table 4: Summary of Criteria Applied for Screening of Natural Hazards

External Natural Hazard	Screening Criterion
Earthquakes	Screened in
Slope Instability	No hazard
Subsidence	No hazard
Soil Frost	No hazard
Flooding Due to Runoff	QN1
Flooding Due to Rivers	QL-6
Flooding Due to Waves	QL-6
Flooding Due to Seiche	No hazard
Flooding Due to Tsunami	No hazard
Flooding Due to Sudden Releases of Water from Natural or Artificial Storage	No hazard
Flooding Due to Ice-Jamming	QL-5
Flooding Due to Other Causes	No hazard
Flooding Due to Combined Events	QN1
Extreme Low Temperature	Screened in
Extreme High Temperature	Screened in
Snowpack	QL-5
Freezing Rain	QL-2
Avalanches	No hazard
Hurricanes/Tornadoes	Screened in
Ice Storms	Screened in
Lightning	QL-6
Meteorites	QN5
Geomagnetic Storms	QL-1
Animals: Lake	Screened in
Animals: Land	QL-3
Animals: Airborne	QL-6

Table 5: Screening for External Human-Induced Hazards for IFB

Human-Induced External Hazard	Screening
Small Aircraft Impact	Screened In
Large Aircraft Impact	Screened Out
Rail Transportation – Cold Toxic Gas Release: Chlorine, Sulphuric Acid and Sulphur Dioxide	Screened In
Rail Transportation – Cold Toxic Gas Release – Ammonia, Hydrogen Chloride and Hydrogen Fluoride, BLEVE Missile, VCE, and Rail Line Blast	Screened Out
Road Transportation – Cold / Hot Toxic Gas Release, BLEVE Missile, VCE, and Explosion	Screened Out
Ship Accident	Screened In
Nearby Nuclear Event	Screened Out
Fixed Sources – Toxic Gas Release	Screened Out
Fixed Sources – BLEVE Missile	Screened Out
External Fires – Including Forest Fire	Screened Out
Thermal Radiation from Fire	Screened Out
Orbital Debris	Screened Out

Table 6: Screening for Natural External Hazards for IFB

External Natural Hazard	Screening Criterion
Earthquakes	Screened In
Flooding – Due to Run Off	Screened In
Flooding – Due to Combined Events	Screened In
Low Lake Levels	Screened Out
Meteorological Extremes – High and Low Temperature, Rainfall, Snow, Freezing Rain, Snowpack	Screened In
Meteorological – Hurricanes/Tornadoes, Ice Storms	Screened In
Mist	Screened In
White Frost	Screened In
Frazil Ice	Screened In
Geomagnetic Storm	Screened In
Bio-fouling	Screened In
Soil Failures (slope failure, subsidence, soil frost and erosion)	Screened Out
Combined Hazard Events	Screened In

Table 7: Screening of the Human Induced External Hazards for UFDS

Human-Induced External Hazard	Screening
Large Aircraft Impact	Screened out
Small Aircraft Impact	Screened out
Rail Transportation (Cold/Hot Toxic Gas Release, BLEVE Missile, Vapour Cloud Explosion, Rail Line Blast)	Screened out
Road Transportation	Screened out
Ship Accident	Screened out
Stationary Sources of Hazards: • Nearby Nuclear Site Accident, Toxic Gas Release, BLEVE • Thermal Radiation (e.g., from BLEVE hazard with accompanying fireball, jet fire hazard from natural gas pipeline failure, fuel fire following an aircraft crash) • Other Stationary Non-Nuclear Hazards (e.g., Regional Water Treatment Plants)	Screened out
External Fires – including Forest Fire	Screened out
Orbital Debris Crashes	Screened out

Table 8: Screening of the Natural External Hazards for UFDS

External Natural Hazard	Screening Criterion
Earthquake	Screened Out
External Flooding: • Flooding due to Runoff • Flooding due to River • Flooding due to Waves • Flooding due to sudden release of water from natural or artificial storage • Flooding due to ice jamming, lake ice, seiche • Flooding due to underwater landslides • Flooding due to combination of events	Screened Out
Low Lake Levels	Screened Out
Extreme Temperatures	Screened Out
Snow/Snowpack	Screened Out
Freezing Rain	Screened Out
Mist White Frost	Screened Out
Soil Failures: • Slope Instability • Subsidence • Soil Frost • Erosion	Screened Out
Avalanches	Screened Out
Ice Storms	Screened Out
High Winds, Tornadoes, Hurricanes	Screened Out
Lightning	Screened Out
Meteorites	Screened Out
Geomagnetic Storm and Solar Flares	Screened Out
Biofouling	Screened Out
Animals	Screened Out

Table 9: Screening of the Internal Hazards for IFB

Internal Hazard		Screening Criterion
Loss of Heat Sink	Random IFB cooling system failures (e.g., pumps, flow path, valving, control logic, etc.);	Screened In
	Random IFB support systems failures (e.g., power, air, water supply failure);	Screened In
	Human errors (e.g., due to maintenance and testing);	Screened In
	Internal IFB fires;	Screened In
	Internal IFB flooding;	Screened In
	Reactor hazards that may impact IFB cooling system equipment operation (e.g., main stream line breaks, turbine generator fire, etc.)	Screened In
Loss of IFB water Inventory		Screened In
Hydrogen Generation in the IFB Due to Radiolysis		Screened Out
Transfer Mechanism Room Accidents		Screened Out
Conveyor Unloader Accidents		Screened Out
Fuel Module Drop in IFB		Screened Out
DSC Loading Accidents at the IFB		Screened Out

Table 10: Screening of the Internal Hazards for UFDS

External Natural Hazard	Screening Criterion
Turbine Missiles	Screened Out
HT Pump Missiles	Screened Out
Missiles from Valves and Pumps	Screened Out
Explosions in hazardous Materials Storage Building; includes Missiles from Acetylene Explosion	Screened Out
Release of Oxidizing, Toxic, Radioactive or Corrosive Gases and Liquids from On-Site Storage	Screened Out
Release of Stored Energy	Screened Out
Dropped or Impacting Loads, e.g.; Crane Failure, DSC collision during craning (loaded DSC colliding with another DSC, loaded or empty) Transporter collision with a loaded DSC or another transporter) Equipment collision with a loaded DSC during craning due to operator error	Screened Out
Vehicle Impacts – Onsite Vehicle Movement (Outdoor Within Protected Area)	Screened Out
Vehicle Impacts – Within Waste Management Facility	Screened Out
Toxic and/or Dangerous Good – Onsite Vehicle Movements (Cold Toxic Hazards)	Screened Out
Electromagnetic Interference	Screened Out
Static Electricity	Screened Out
Fires	Screened Out
Loss of Support Services to the UFDS (Electrical Power, Control Power, Instrument Air, HVAC, Service Air)	Screened Out
Mishandling of Fuel (e.g., newer than 10 years old fuel transfer to DSC)	Screened Out
Criticality	Screened Out

Table 11: Pickering B At-Power Internal Events PSA Initiating Events

Category	Label IE-30-	Description
Forced Shutdown	FSD	All events resulting in reactor shutdown not included in other IEs
LOCA	LOCA1	A rupture within the capacity of the D_2O feed system (initial discharge rate 1-40 kg/s)
	LOCA2A	Small breaks which require ECIS for refilling and repressurization of the HTS (initial discharge rate 40-100 kg/s)
	LOCA2B	Small breaks which require ECIS for refilling and repressurization of the HTS (initial discharge rate 100-1000 kg/s)
	LOCA3	Large breaks which require high and subsequently low pressure ECI for refilling and do not result in flow stagnation into the core (initial discharge rate >1000 kg/s)
	LOCA4	Large breaks which require high and subsequently low pressure ECI for refilling and lead to flow stagnation into the core (initial discharge rate >1000 kg/s)
	LOCA1-SF	Stagnation feeder break in LOCA1 range
	LOCA2-SF	Stagnation feeder break in LOCA2A range (initial discharge rate 65-165 kg/s)
Pressure Tube Rupture	PTF	Pressure tube failure resulting in an initial discharge rate in excess of 1 kg/s
	PTL	Pressure tube failure resulting in an initial discharge rate of less than 1 kg/s
End-fitting Failure	EFL2	End-fitting break of LOCA2-size outside annulus gas bellows (initial discharge rate up to 1000 kg/s)
Steam Generator Tube Rupture	SGTB1	Boiler tube break within the capacity of the D_2O feed system (initial discharge rate 1-40 kg/s)
	SGTB2	Boiler tube break beyond the capacity of the D_2O feed system (initial discharge rate >40 kg/s)
Loss of HT Pressure/Inventory Control (Low)	LRVO	One or more liquid relief valves open spuriously
	LBVO	A liquid bleed valve opens spuriously
	2LBVO	Both liquid bleed valves open spuriously
	FVFC	Both D_2O feed valves fail closed
	FPFO	Operating D_2O feed pump fails
	XSPR	Bleed condenser spray valve 3332-CV113 opens spuriously

Category	Label IE-30-	Description
	BCRVO	Bleed condenser relief valve fails open
Loss of HT Pressure/Inventory Control (High)	BVFC	Both HT bleed valves fail closed
	FVFO	Any D_2O feed valve fails open
	FP2S	Inadvertent prolonged operation of standby D_2O feed pump when not required
	BCLCVFC	Bleed condenser level control valves fail closed
Loss of HT Inventory Control	D2OFDL	Pipe break in D_2O feed system upstream of check valve 3331-NV1 or -NV2
HT Pump Trip	HTPT	Any HT pump trips
Channel Flow Blockage	LFB	Channel flow reduced by 90 per cent or more
	HTMV	A normally-open HT motorized valve closes spuriously
Moderator Failure	LOMHS	Loss of moderator heat sink
	LOMF	Loss of moderator flow
	LOMI	Loss of moderator inventory
Loss of End Shield Cooling	LOESHS	Loss of end shield heat sink
	LOESF	Loss of end shield flow
	LOESI	Loss of end shield inventory
Steam Line Break	SSLB-IC	Small steam line break inside containment (initial discharge rate 10-100 kg/s)
	SSLB-OC	Small steam line break outside containment (initial discharge rate 10-100 kg/s)
	ISLB-IC	Intermediate steam line break inside containment (initial discharge rate 100-1000 kg/s)
	ISLB-OC	Intermediate steam line break outside containment (initial discharge rate 100-1000 kg/s)
	LSLB-IC	Large steam line break inside containment (initial discharge rate >1000 kg/s)
	LSLB-OC	Large steam line break outside containment (initial discharge rate >1000 kg/s)
	SRV	One or more atmospheric steam reject valves spuriously open
	U678SSLB-OC	Unit 6,7 or 8 small steam line break outside containment (initial discharge rate 10-100 kg/s)

Category	Label IE-30-	Description
	U678ISLB-OC	Unit 6,7 or 8 intermediate steam line break outside containment (initial discharge rate 100-1000 kg/s)
	U678LSLB-OC	Unit 6,7 or 8 large steam line break outside containment (initial discharge rate >1000 kg/s)
	IE-44-LSLB-OC IE-44-U1LSLB-OC	Large steam line break outside containment on Pickering NGS ‘A’ Unit 4 Large steam line break outside containment on Pickering NGS ‘A’ Unit 1 These IEs are described, modelled, and quantified as documented in the PARA-L1P study
Loss of Feedwater to Boilers	TLOFW	Total loss of feedwater to all quadrants
	PLOFW	Partial loss of feedwater to all quadrants
	ALOFW	Asymmetric loss of feedwater (no feedwater flow to any single quadrant)
Feedwater Line Break	FLB-IC	Feedline break inside containment
	SFLB-OC	Small feedline break outside containment
	LFLB-OC	Large feedline break outside containment resulting in total loss of feedwater
	FLBCOND	Break in condensate system resulting in total loss of condensate flow to deaerator
	U678LFLB-OC	Unit 6, 7 or 8 large feedwater line break outside of containment
Turbine Trip	TT	All turbine trips not included in other IEs (includes loss of condenser vacuum events)
Loss of Condensate Flow	LOCONDA	Total loss of condensate flow to deaerator (excluding condensate pipe breaks)
	LOCONDB	Loss of main condensate flow to deaerator (excluding condensate pipe breaks)
HP Reheater Drains Line Break to Boilers	RDLB	Breaks in reheater drains line between the boilers and the second check valve
Unplanned Increase in Reactivity	FLOR	Unplanned bulk fast reactivity insertion
	SLOR	Unplanned bulk slow reactivity insertion
	LZCPMPFL	All liquid zone control system pumps fail

Category	Label IE-30-	Description
	URIR	Unplanned regional increase in reactivity
	SORD	Spurious shutoff rod drop resulting in a regional increase in reactivity
Loss of Computer Control	WDTOX	Controlling computer stall Stall of the control computer is an IE when it is combined with failure of the standby computer to assume control. Following WDTOX event, it is expected that the standby computer will assume control of all computer-controlled process outputs. Failure to transfer control is explicitly modelled in the ET / FTs.
	DCCF	Dual computer failure
	DCCUF	Unsafe failure of DCC leading to reactor power increase
	BPCF	Failure 'off' of boiler pressure control program on both computers
	MTCF	Failure 'off' of moderator temperature control program on both computers
	FHCF	Failure 'off' of fuel handling system control program on DCC2
	RRSF	Failure 'off' of reactor power control program on both computers
Loss of LPSW System	LOLPSW	Total loss of low pressure service water
Total Loss of Service Water	TLOSW	Total loss of common and emergency service water (main and emergency screenhouses).
Loss of Common Service Water	LOCSW	Loss of common service water (total loss of main screenhouse)
Partial Loss of Common Service Water	PLOCSW	Partial loss of common service water (partial loss of main screenhouse)
Adverse conditions in the forebay	FOREBAY	In the event tree analysis, events IE-TLOSW, IE-LOCSW, and IE PLOCSW are combined into a single event called IE FOREBAY as all of them are caused by adverse conditions in the forebay.
Loss of HPSW System	LOHPSW	Total loss of high pressure service water
Loss of RCW System	LORCW	Total loss of recirculated cooling water system flow
Loss of Instrument Air	TLOUIA	Total loss of unit instrument air

Category	Label IE-30-	Description
	TLOCIA	Total loss of common instrument air
Loss of Bulk Electricity Supply	LOBES	Loss of bulk electricity supply
Loss of Switchyard	LOSWYD	Loss of switchyard
Loss of Unit Class IV 4.16 kV Bus	LOCL4	Total loss of unit Class IV power
	LOSST	Loss of system service transformer or circuit breakers 5320-CB1A or -CB1C causing loss of power supply to Class IV 4.16 kV buses 5320-BUA or -BUC, respectively
	LO5320BUA LO5320BUB LO5320BUC LO5320BUD	Loss of unit Class IV 4.16 kV bus BUA Loss of unit Class IV 4.16 kV bus BUB Loss of unit Class IV 4.16 kV bus BUC Loss of unit Class IV 4.16 kV bus BUD
Loss of Unit Class IV 600 V Bus	LO5330BUA LO5330BUB LO5330BUC LO5330BUD LO5330BUF	Loss of unit Class IV 600 V bus BUA Loss of unit Class IV 600 V bus BUB Loss of unit Class IV 600 V bus BUC Loss of unit Class IV 600 V bus BUD Loss of unit Class IV 600 V bus BUF
Loss of Unit Class III 4.16 kV Bus	LO5412BUA LO5412BUB	Loss of unit Class III 4.16 kV bus BUA Loss of unit Class III 4.16 kV bus BUB
Loss of Unit Class III 600 V Bus	LO5413BUA LO5413BUB LO5413BUC LO5413BUD LO5413BUE	Loss of unit Class III 600 V bus BUA Loss of unit Class III 600 V bus BUB Loss of unit Class III 600 V bus BUC Loss of unit Class III 600 V bus BUD Loss of unit Class III 600 V bus BUE
Loss of Unit Class II 600 V Bus	LO5423BUA LO5423BUB	Loss of unit Class II 600 V bus BUA Loss of unit Class II 600 V bus BUB

Category	Label IE-30-	Description
Loss of Unit Class II 120 V Bus	LO5424BUA LO5424BUB LO5424BUC LO5424BUD LO5424BUE LO5424BUF LO5424BUG LO5424BU1A LO5424BU1B LO5424BU1C LO5424BU1D LO5424BU1E LO5424BU1F LO5424BU1G LO5424BU2C LO5424BU2D	Loss of unit Class II 120 V ac bus BUA Loss of unit Class II 120 V ac bus BUB Loss of unit Class II 120 V ac bus BUC Loss of unit Class II 120 V ac bus BUD Loss of unit Class II 120 V ac bus BUE Loss of unit Class II 120 V ac bus BUF Loss of unit Class II 120 V ac bus BUG Loss of unit Class II 120 V ac bus BU1A Loss of unit Class II 120 V ac bus BU1B Loss of unit Class II 120 V ac bus BU1C Loss of unit Class II 120 V ac bus BU1D Loss of unit Class II 120 V ac bus BU1E Loss of unit Class II 120 V ac bus BU1F Loss of unit Class II 120 V ac bus BU1G Loss of unit Class II 120 V ac bus BU2C Loss of unit Class II 120 V ac bus BU2D
Loss of Unit Class II 48 V dc Bus	LO5425BU1 to LO5425BU23 LO5425BU31 to LO5425BU52	Loss of unit Class II 48 V dc bus BU1 to bus BU23 Loss of unit Class II 48 V dc bus BU31 bus BU52
Loss of Unit Class I 250 V dc	LO250	Loss of unit Class I 250 V dc buses BUA and BUB
Heat Transport Flow Diversions	SDCMV	Spurious opening of the shutdown cooling isolation valves in one or more quadrants.
Powerhouse Freezing	PHFREEZE	Spurious opening of powerhouse venting during an extreme cold outside condition
ECI Blowback	IE-ECIBB See Appendix B26	ECI Blowback (Spurious Valve Opening Events, and Test Events)

Table 12: PBRA Fuel Damage Categories

FDC*	Definition	Typical Events in FDC
FDC1	Rapid loss of core structural integrity.	Positive reactivity transient and failure to shutdown the reactor.
FDC2	Slow loss of core structural integrity.	LOCA with failure of HTS inventory makeup and failure of moderator heat sink.
FDC3	Moderator required as heat sink in the short term (< 1 hr after reactor trip).	LOCAs of LOCA2 size or greater and failure of HTS makeup before one hour, and successful moderator heat removal.
FDC4	Moderator required as heat sink in the intermediate term (1 to 24 hr after reactor trip).	LOCAs of LOCA2 size or greater and failure of HTS makeup on demand or during mission before 24 hours, and successful moderator heat removal. A loss of all heat sinks leading to breaks in the HTS, with successful HTS inventory makeup.
FDC5	Moderator required as heat sink in the long term (> 24 hr after reactor trip).	LOCAs and failures of HTS makeup after 24 hours, with successful moderator heat removal.
FDC6	Temporary loss of cooling to fuel in many channels.	LOCA4 with successful ECI.
FDC7	Single channel fuel failure with sufficient release of steam or radioactivity to initiate automatic containment button-up.	End-fitting LOCA2 and fuel ejection with successful ECI. LOCA2 stagnation feeder break or large flow blockage, with successful ECI. In-core LOCA2 and end fitting release with successful ECI.
FDC8	Single channel fuel failure with insufficient release of steam or radiation activity to initiate automatic containment button-up.	LOCA1 stagnation feeder break, with successful D_2O makeup or ECI.
FDC9	LOCAs with no fuel failure (ECIS successful); potential for significant economic impact.	LOCAs of size LOCA2A, LOCA2B or LOCA3 with successful ECI. LOCAs of size LOCA1 and failure of D_2Omakeup, with successful ECI and a heat sink.
S	Success plant state. No fuel failure, ECIS not required.	LOCA of size LOCA1 with successful D_2O makeup and long term heat sink. No LOCA events with a successful heat sink.

* End-states representing accident sequences with containment bypass include suffix "-OC" (Outside Containment)

Table 13: List of Systems Modelled by Fault Trees

System Name	L1 At-Power	L1 Outage	L2 At-Power
Heat Transport Feed, Bleed and Relief and D_2O Storage and Transfer System	Y	Y	*
Heat Transport D_2O Recovery System	Y	Y	*
Heat Transport Pump Gland Seal Supply and Gland Seal LOCA	Y	Y	*
Heat Transport Shutdown Cooling System	Y	Y	*
Moderator System	Y	Y	*
Boiler Feedwater System	Y	Y	*
Boiler Emergency Cooling Supply	Y	Y	*
Steam Relief System	Y	Y	*
Class IV Power Supply System	Y	Y	*
Class III Power Supply System	Y	Y	*
Class II Power Supply System	Y	Y	*
Class I Power Supply System	Y	Y	*
Low Pressure Service Water System	Y	Y	*
Recirculated Cooling Water System	Y	Y	*
High Pressure Service Water System	Y	Y	*
Unit Instrument Air System	Y	Y	*
Common Instrument Air System	Y	Y	*
Emergency Coolant Injection System	Y	Y	*
Emergency Water Supply System	Y	Y	*
Standby Generator Fuel Oil System	Y	Y	*
Hostile Environment Events	Y	Y	*
Shutdown System No. 1	Y	N	*
Shutdown System No. 2	Y	N	*
Annulus Gas System	Y	Y	*
Digital Control Computer	Y	Y	*
Emergency Power Supply System	Y	Y	*
Cooling and Ventilation System (UPS, EPS, SG rooms)	Y	Y	*
Reactivity Control System	Y	N	*
Condensate System	Y	Y	*
Emergency Mitigating Equipment	Y	Y	*
Shutdown Heat Sinks	N	Y	N/A
Pressure Relief Valves	N	N	Y
Containment Isolation, Airlocks and Hydrogen Ignition System	N	N	Y
Containment In-Leakage	N	N	Y
Boiler Room and Fuelling Machine Vault Air Cooling Units	N	Y	Y
Pressure Relief Panel System	N	N	Y
Filtered Air Discharge System	N	N	Y

System Name	L1 At-Power	L1 Outage	L2 At-Power
Emergency Coolant Injection Blowback System	Y	N	N

* Included in Level 2 At-Power Model through integration with Level 1 At-Power Model

Note: Fire, seismic and flooding risk is calculated through modifications or interrogations based on the integrated severe core damage model from the Internal Events At-Power Level 1 PSA, and do not include specific FT models for the individual plant systems.

Table 14: PBRA-L1O Plant Operational State Definition

Input Parameter	Plant Operational State (POS)				
	A	B	C	D	E
GSS	OPGSS	DGSS, or RBGSS with drained moderator	OPGSS	OPGSS	OPGSS
HTS Inventory Level	Full	Full	LLDS	Full	Full
HTS Boundary Configuration	Closed	Closed	Open	Closed	Closed
Typical HTS Temp	38°C	<90°C	According to NK30-OP-33000-0014 – 0016	<70°C	<90°C
Typical HTS Pressure (ROH)	≤200 kPa(g)	≤200 kPa(g)	0 kPa(g)	≤200 kPa(g)	≥2.7 MPa(g)
Typical Primary Heat Sink (Circulation)	SDC pumps (Even / Odd)	SDC pumps (Even / Odd)	Convection	SDC pumps (Even / Odd)	HTS pumps
Typical Primary Heat Sink (Heat Removal)	SDC heat exchangers (Even / Odd)	SDC heat exchangers (Even / Odd)	ACU+ESC+ Moderator	Feedwater + Boiler blowdown	SDC heat exchangers (Even / Odd)
Typical Backup Heat Sink (Circulation)	SDC pumps (Odd / Even), Convection	SDC pumps (Odd / Even), Convection	SDC pumps	SDC pumps (Odd / Even)	SDC pumps
Typical Backup Heat Sink (Heat Removal)	SDC heat exchangers (Odd / Even)	SDC heat exchangers (Odd / Even), Boiler blowdown, ACU+ESC	SDC heat exchangers (Odd / Even)	Boiler blowdown (reheater drains pump)	SDC heat exchangers (Odd / Even), boiler blowdown[2]
Emergency Heat Sink	EWS[1]	EWS[1]	EWS[1]	EWS[1]	EWS[1]
POS Time Fraction Per Reactor-Year	0.0724	0.0154	0.0032	0.0066	0.0046

Note 1: EWS heat sink may include (depending on the configuration):

1. EWS supply to at least two boilers in each loop with heat reject through at least three large SRVs.
2. EWS supply to HT.
3. EWS makeup to moderator (not available in DGSS).
4. EWS supply to Boiler Room and FM vault ACUs.

Note 2: Boiler blowdown (HS#8/8RH) cannot be used when main circulating pumps are operating.

Table 15: Initiating Events for Pickering B Level 1 Outage PSA

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
Initiating Events Related to Intrinsic System Failures for Primary Heat Sink								
1	PHS-POSE-HS2	Failure of Primary Heat Sink #2 (Main HT pumps and Boiler Blowdown)	N	N	N	N	Y	This event represents the group of events leading to the intrinsic failure of the heat sink#2 (main HTS pumps for circulation and boiler blowdown for heat rejection). This includes combinations of equipment failures and failed human actions that cause circulation in the HTS to fall below that required for sustained decay heat removal or failure of the heat rejection process.
2	PHS-POSE-HS4	Failure of Primary Heat Sink #4 (Main HT pumps and SDC HXs)	N	N	N	N	Y	This event represents the group of events leading to the intrinsic failure of the heat sink#4 (main HTS pumps for circulation and SDC HXs for heat rejection). This includes combinations of equipment failures and failed human actions that cause circulation in the HTS to fall below that required for sustained decay heat removal or failure of the heat rejection process.
3	PHS-POSA-HS5 PHS-POSB-HS5 PHS-POSE-HS5	Failure of Primary Heat Sink #5 (SDC pumps and SDC HXs)	Y	Y	N	N	Y	This event represents the group of events leading to the intrinsic failure of the heat sink#5 (SDC pumps for circulation and SDC HXs for heat rejection). The grouped events include intrinsic equipment failures as well as human induced failures such as loss of cooling water to SDC HXs (LOCOOL-SDC), SDC forced flow (LOCIRC-SDC), and spurious closure of any SDC isolation MV (SDC-MV in DARA outage IEs).
4	PHS-POSE-HS7	Failure of Primary Heat Sink #7 (SDC pumps and Bleed Cooler)	N	N	N	N	Y	This event represents the group of events leading to the intrinsic failure of the heat sink#7 (SDC pumps for circulation and bleed cooler for heat rejection). Bleed cooler is supported by service water (RCW and LPSW). The grouped events include intrinsic

	Outage IE Label	IE Definition	POS Applicability A	B	C	D	E	Discussion
								equipment failures as well as human induced failures of the SDC forced flow (LOCIRC-SDC), spurious closure of HT pump discharge MV (HTMV), spurious opening of the SDC isolation MVs in a SDC loop not in service (SDCMV), spurious closure of any SDC isolation MV (SDC-MV in DARA outage IEs), total loss of LPSW (LOLPSW), loss of RCW (LORCW), and failure of the feed and bleed system.
5	PHS-POSA-HS8 PHS-POSB-HS8 PHS-POSD-HS8 PHS-POSE-HS8	Failure of Primary Heat Sink #8 (SDC pumps, MBFP, Condensate Pumps and Boiler Blowdown)	Y	Y	N	Y	Y	This event represents the group of events leading to the intrinsic failure of the heat sink#8 (SDC pumps for circulation and boiler blowdown for heat rejection). Boiler blowdown is supported by intermittent supply of feedwater by a main / auxiliary feedwater pump. The grouped events include combinations of equipment failures and failed human actions that cause circulation in the SDC to fall below that required for sustained decay heat removal or failure of the heat rejection process.
6	PHS-POSA-HS8RH PHS-POSB-HS8RH PHS-POSD-HS8RH	Failure of Primary Heat Sink #8RH (SDC pumps and Boiler Blowdown using Re-heater Drains Pump)	Y	Y	N	Y	N	This event represents the group of events leading to the intrinsic failure of the heat sink#8RH (SDC pumps for circulation and boiler blowdown using re-heater drains pump for heat rejection). Boiler blowdown is supported by demineralized water supply. The grouped events include intrinsic equipment failures as well as human induced failures of the SDC forced flow (LOCIRC-SDC), spurious closure of HT pump discharge MV (HTMV), spurious opening of the SDC isolation MVs in a SDC loop not in service (SDCMV), spurious closure of any SDC isolation MV (SDC-MV in DARA outage IEs).
7	PHS-POSB-HS9a	Failure of Primary Heat Sink #9a (Convection and	N	Y	N	N	N	This event represents the group of events leading to the intrinsic failure of the heat sink#9a (convection for

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
		ACUs and ESC)						circulation and ACU and ESC for heat rejection). The grouped events include intrinsic equipment failures as well as human induced failures of the end shield flow (LOESF), loss of end shield inventory (LOESI), and spurious closure of any SDC isolation MV (SDC-MV in DARA outage IEs).
8	PHS-POSA-HS9b PHS-POSC-HS9b	Failure of Primary Heat Sink #9b (Convection and ACUs, Moderator, and ESC)	Y	N	Y	N	N	This event represents the group of events leading to the intrinsic failure of the heat sink#9b (convection for circulation and ACU, moderator, and ESC for heat rejection). The grouped events include intrinsic equipment failures as well as human induced failures of the moderator heat sink (LOMHS), loss of moderator flow (LOMF), loss of moderator inventory (LOMI), loss of end shied cooling (LOESHS), loss of end shield flow (LOESF), loss of end shield inventory (LOESI), and spurious closure of any SDC isolation MV (SDC-MV in DARA outage IEs).
Initiating Events Related to HT System Boundary								
9	LEAK	Non-isolatable HTS leak due to maintenance induced causes or single ice plug failure (within the capacity of two D_2O feed pumps)	Y	Y	Y	Y	N	The LEAK IEs represent non-isolatable failures of the HTS that occur when the primary HTS is initially depressurized. Mitigating system requirements (e.g., D_2O recovery, ECI) are based on the discharge rates and break locations, and do not depend on the cause of the initial failure (e.g., single channel failure caused by a fuelling machine, versus failure of a feeder ice plug). The IE applies to POSs A, B, C and D, where the HTS is initially depressurized. This event represents the outage HTS leaks (LK1A/B/C) identified in DARA outage assessment failure of a single ice plug (ICE-PLUG), CIGAR event from the PBRA 2007

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
								outage assessment, and very small LOCA (VSLOCA) identified in PBRA 2007 outage.
10	LLEAK	Non-isolatable HTS large leak due to load drop or feeder damage from inadvertent fuelling machine movement (beyond the capacity of D_2O Recovery)	Y	Y	Y	Y	N	The large leak (LLEAK) IEs represent non-isolatable failures of the HTS that occur when the primary HTS is initially depressurized. The leak is beyond the capacity of the D_2O Recovery system. The most likely mechanism is inadvertent movement of the fuelling machine (EFL2), which can be experienced both at-power and during plant outages. The IE applies to POSs A, B, C and D, where the HTS is initially depressurized.
11	LOCA1	Non-isolatable rupture within the capacity of two D_2O feed pumps (initial discharge rate 1-40 kg/s)	N	N	N	N	Y	The LOCA1 IE consists of non-isolatable small breaks of pressure-retaining components (e.g., piping) in the HTS that occur when the primary HTS is initially pressurized. During GSS, ECI must be manually initiated in all cases, if required. This IE only applies to POS E, which represent states where the HTS is initially pressurized. The LOCA1 IE represents LOCA1 size and stagnation feeder break in LOCA1 range (LOCA1-SF) from the PBRA At-Power IEs as well as break inside and outside annulus gas bellows in LOCA1 range (EFL1WAGA and EFL1OAGA) and break involving fuelling machine in size of LOCA1 (EFL1FMIA) from the DARA At-Power IEs.
12	LLOCA	Non-isolatable breaks inside containment from a pressurized HTS, beyond the capacity of two D_2O feed pumps	N	N	N	N	Y	The LLOCA IE consists of large failures of pressure-retaining components in the HTS that occur when the system is initially pressurized. The LLOCA IE represents a group of LOCAs (i.e., LOCA2A/B, LOCA2-SF, EFL2, LOCA3

	Outage IE Label	IE Definition	POS Applicability A	B	C	D	E	Discussion
		(initial discharge rate >40 kg/s)						and LOCA4) from the at-power IEs: Given that the outage LLOCA IE represents non-isolatable breaks inside containment, there is no need to further differentiate between break locations based on the plant response (e.g., core voiding, power pulse, etc.). The mitigating requirements are also similar in all cases. The initial discharge rate might be in either the LOCA2 (>40 kg/s) or LOCA3/4 ranges (>1000 kg/s), but since the unit is in the GSS (i.e., minimal driving force from fuel energy) the HTS would rapidly depressurize. This IE only applies to POS E, where the HTS is initially pressurized.
13	ICEPLUGS	Failure of liquid nitrogen supply to all ice plugs	Y	Y	N	Y	N	The ICEPLUGS IE represents a failure of the liquid nitrogen supply to all ice plugs in use for the outage unit. Outage PBRA included an ICE-PLUG event, but for the current outage PBRA these single failures are included by the leak IEs (LEAK or LLEAK) from a depressurized HTS, as applicable given the size and location of the specific ice plug. The common mode ICEPLUGS IE would result not only in failure of all HTS ice plugs (i.e., resulting in a loss of HTS inventory) but would also cause failure of any ice plugs in other potential mitigating systems such as the moderator. The ICEPLUGS IE only applies to POSs A, B and D, since HTS ice plugs are only used when the system is full and depressurized. Note that a single failure of an ice plug in systems other than the primary HTS would be captured by other IEs (e.g., LOMI etc.).
Initiating Events Related to Pressure Tube Failure								
14	PTF	Pressure tube	N	N	N	N	Y	Pressure tube failures from a

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
		failure resulting in an initial discharge rate in excess of 1 kg/s						pressurized HTS (POS E) potentially result in consequential calandria tube failure and possible end fitting ejection. For the outage PSA, this also groups the large flow blockage event (LFB), since the mitigating actions would be the same in both cases.
15	PTL	Pressure tube failure resulting in an initial discharge rate of less than 1 kg/s	Y	Y	Y	Y	Y	Pressure tube leaks from a pressurized HTS (POS E), combined with failure to detect the leak using the annulus gas system, potentially result in consequential calandria tube failure and end fitting ejection. Pressure tube leaks from a depressurized HTS (POSs A, B, C and D) would result in inventory losses to the annulus gas system tank (34980-TK1). A postulated pressure tube leak from the depressurized HTS, but where the annulus gas bellows does fail, is captured by the LEAK IE for a small non-isolatable HTS leak inside containment.
Initiating Event Related to Boiler Tube Rupture								
16	SGTB	Boiler tube break	N	N	N	N	Y	Boiler tube ruptures are postulated for POS E when the HTS is full and pressurized. Boilers are not normally the primary heat sink in POS E. The SDC HX(s) are in service. But flow path for this heat sink is split between SDCHX and boilers so at 2.7 MPa there could be a single boiler tube leak. Also, this may be a concern in POS A, B, and D when HS#8 is in service. Then primary side fluid from the SDC is being circulated through boilers u-tubes. However, failures of boiler tubes are assumed incredible in depressurized plant operational states (POS A, B, C and D) due to the design pressure of boiler tubes (10.1 MPa). The pressure differential across the boiler tubes is about 4.8 MPa when a unit is at power or about 2.7 MPa when a unit is in a

	Outage IE Label	IE Definition	POS Applicability A	B	C	D	E	Discussion
								pressurized outage state (POS E). In addition, depressurized plant operational states, the boiler tubes are either empty (POS C) or full and only slightly pressurized by the D_2O Storage Tank cover gas (POS A, B, D). The shell side of the boilers may be either drained or full and depressurized. In all possible combinations, the maximum estimated pressure differential across boiler tubes cannot be more than 100 kPa at the bottom of the boiler (e.g., in configuration when a boiler is drained and the primary side is pressurized by the D_2O Storage Tank cover gas).
Initiating Event Related to SDC Heat Exchanger Tube Breaks								
17	SDCHX	SDC HX tube break within the capacity of two D_2O feed pumps	Y	Y	Y	Y	Y	The SDCHX IE represents failures of single or multiple tube(s) in the SDC heat exchangers. The break size does not impact the accident progression and credited systems, and therefore, the event trees model a single SDCHX event independent of break size.
Initiating Event Related to Moderator Loss of Inventory								
18	LOMI	Loss of Moderator Inventory	Y	N	Y	Y	Y	This event represents an inadvertent loss of moderator inventory due to a rupture in the moderator system that leads to a drained calandria. It is assumed that the rupture in the moderator system is such that it cannot be isolated and the lost inventory cannot be recovered using the moderator collection system.
Initiating Events Related to SDC System Boundary								
19	LEAK-SDC	Isolatable leak in piping within the SDC system	Y	Y	Y	Y	N	The LEAK-SDC event represents small leaks from the SDC system that discharge into containment but that can be isolated by closing the shutdown

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
								cooling MVs. The sustained discharge would only be a concern in cases where the operators failed to isolate the break. This IE applies to all POSs when the HTs is depressurized including the Low Level Drained State (LLDS), where both SDC loop isolating MVs should be open in one SDC loop per HT loop.
20	LOCA1-SDC	Isolatable break in piping within the SDC system within the capacity of D_2O feed pumps	N	N	N	N	Y	The LOCA1-SDC event represents pipe ruptures (i.e., LOCA1 size) from the SDC system that discharge into containment but that can be isolated by closing the shutdown cooling MVs. The sustained discharge would only be a concern in cases where the operators failed to isolate the break. This IE applies to POS E when HTS is pressurized.
21	LLOCA-SDC	Isolatable large break in piping within the SDC system beyond the capacity of two D_2O feed pumps	N	N	N	N	Y	The LLOCA-SDC event represents large pipe ruptures (i.e., LOCA2/3/4 size) from the SDC system that discharge into containment but that can be isolated by closing the shutdown cooling MVs. The sustained discharge would only be a concern in cases where the operators failed to isolate the break. This IE applies to POS E when the HTS is pressurized.
Initiating Events Related to Adjacent Unit Secondary Side Line Break Events								
22	U678LSSLB-OC	Unit 6, 7 or 8 large secondary side line break outside containment (initial discharge rate >1000 kg/s)	Y	Y	Y	Y	Y	This event postulates a large secondary side line break (initial discharge >1000 kg/s) occurring on a main steam line or feedwater line at one of the sister units (i.e., Unit 6, 7 or 8) of the outage unit (i.e., Unit 5). The secondary side line break is postulated to occur inside the powerhouse, hence resulting in a steam environment in the powerhouse which may impact heat sink availability for the outage unit (i.e., Unit 5). An adjacent unit steam line break may impact on components and systems that support

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
								the outage heat sink due to a harsh environment. The event is independent of the POSs for the outage unit.
23	U678ISSLB-OC	Unit 6,7 or 8 intermediate steam line break outside containment (initial discharge rate 100-1000 kg/s)	Y	Y	Y	Y	Y	See U678LSSB-OC above.
24	U678SSSLB-OC	Unit 6,7 or 8 small secondary side line break outside containment (initial discharge rate 10-100 kg/s)	Y	Y	Y	Y	Y	See U678LSSLB-OC above.
Initiating Events Related to Loss of Heat Transport Pressure and Inventory Control System (Leading to HTS High Pressure)								
25	BVFC	Any HTS bleed valve fails closed	N	N	N	N	Y	Failures of HTS pressure and inventory control which lead to high pressure in the HTS are of interest as they may lead to opening of the HT LRVs and Bleed Condenser Relief Valves resulting in a LOCA. Accidents induced by failures in pressure and inventory control may occur only in POS E where the HTS is pressurized. IE BVFC is defined as spurious closing of any HTS bleed valve. This event leads to increase of the HTS pressure up to the Liquid Relief Valve (LRV) setpoint in the affected loop.
26	FVFO	Any D_2O feed valve fails open	N	N	N	N	Y	IE FVFO is defined as spurious opening of any HTS feed valve. This event may lead to increase of the HTS pressure.
27	BCLCVFC	Bleed condenser level control valves fail closed	N	N	N	N	Y	IE BCLCVFC is defined as spurious closing of both bleed condenser level control valves (LCV). This event will lead to increase of HTS pressure.

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
Initiating Events Related to Loss of Heat Transport Pressure and Inventory Control System (Leading to HTS Low Pressure)								
28	2LBVO	Spurious opening of both HTS liquid bleed valves	N	N	N	N	Y	Failures of PIC which lead to low pressure in the HTS are of interest as they may impair the operation of the primary or back-up heat sink during an outage. Accidents induced by failures in pressure and inventory control may occur only in POS E where the HTS is pressurized. IE 2LBVO is defined as spurious opening of two HTS liquid bleed valves. This event will lead to depressurization of the HTS.
29	LBVO	Spurious opening of one HTS liquid bleed valve	N	N	N	N	Y	IE LBVO is defined as spurious opening of one HTS liquid bleed valve. This event may lead to depressurization of the HTS.
30	FPFO	Operating D_2O feed pump fails	N	N	N	N	Y	IE FPFO is defined as failure of the operating D_2O feed pump. This event may lead to depressurization of the HTS.
31	FVFC	Any D_2O feed valve fails closed	N	N	N	N	Y	IE FVFC is defined as spurious closing of any D_2O feed valve. This event may lead to depressurization of the HTS.
32	XSPR	Bleed condenser spray valve fails open	N	N	N	N	Y	IE XSPR is defined as spurious opening of bleed condenser spray valve 33320-CV113. This event will lead to tripping of the pressuring pump when D_2O storage tank empties.
Pipe Breaks In the Pressure and Inventory Control System								
33	D2OFDL	Pipe break in D_2O feed system upstream of check valve 3331-NV1 or -NV2	N	N	N	N	Y	IE D2OFDL is defined as a pipe break in the D_2O feed system upstream of check valve 3331-NV1 or -NV2. This event will lead to depressurization of the HTS and a LOCA if not isolated.
Initiating Events Related to Electrical System Failures								

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
34	LOBES	Loss of bulk electricity supply	Y	Y	Y	Y	Y	The loss of Bulk Electrical System (BES) IE is defined as a grid instability event that leads to 230 kV line under-frequency or over-frequency being sensed in the PNGS-B switchyard ring. This results in automatic disconnection of the PNGS-B units from the grid at the 230 kV line breakers. This disconnection prevents the PNGS-B units' generator output (via main transformers) from supplying the grid and also prevents the grid from supplying power to PNGS-B via the system service transformers (SST).
35	LOSWYD	Loss of switchyard	Y	Y	Y	Y	Y	The loss of switchyard IE (LOSWYD) is defined as all events that lead to all of the Pickering NGS B switchyard buses becoming de-energized. Possible causes of the LOSWYD event may be a severe ice storm, or component failure of switchgear (circuit breakers, disconnect switches and busses) and failure to isolate or inadvertent operator error. This event is applicable to all the plant outage states.
36	LOCL4	Total loss of unit Class IV power	Y	Y	Y	Y	Y	The loss of Class IV power event (LOCL4 and LOSST) is defined as a loss of power on all four Class IV buses (53200-BUA, -BUB, -BUC and -BUD) of Unit 5. This may be caused by random or common mode switchgear failures. These events cause loss of power supply to Class IV 4.16 kV buses 53200-BUA and – BUC which feed 53200-BUB or – BUD, respectively. It is postulated that power to 4 kV Class IV buses via the 4 kV SES buses supplied from another unit's SST cannot be restored.
37	LO5320BUA	Loss of unit Class IV 4.16 kV bus BUA	Y	Y	Y	Y	Y	Electrical bus failures can occur in all POSs. The impact on outage unit heat sinks may depend on POS-specific

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
38	LO5320BUB	Loss of unit Class IV 4.16 kV bus BUB	Y	Y	Y	Y	Y	plant configuration (e.g., maintenance activities, undetected failures) at the time of the IE.
39	LO5320BUC	Loss of unit Class IV 4.16 kV bus BUC	Y	Y	Y	Y	Y	
40	LO5320BUD	Loss of unit Class IV 4.16 kV bus BUD	Y	Y	Y	Y	Y	
41	LO5330BUA	Loss of unit Class IV 600 V bus BUA	Y	Y	Y	Y	Y	See loss of unit Class IV 4.16 kV above.
42	LO5330BUB	Loss of unit Class IV 600 V bus BUB	Y	Y	Y	Y	Y	
43	LO5330BUC	Loss of unit Class IV 600 V bus BUC	Y	Y	Y	Y	Y	
44	LO5330BUD	Loss of unit Class IV 600 V bus BUD	Y	Y	Y	Y	Y	
45	LO5330BUF	Loss of unit Class IV 600 V bus BUF	Y	Y	Y	Y	Y	
46	LO5412BUA	Loss of unit Class III 4.16 kV bus BUA	Y	Y	Y	Y	Y	See loss of unit Class IV 4.16 kV above.
47	LO5412BUB	Loss of unit Class III 4.16 kV bus BUB	Y	Y	Y	Y	Y	
48	LO5413BUA	Loss of unit Class III 600 V bus BUA	Y	Y	Y	Y	Y	See loss of unit Class IV 4.16 kV above.
49	LO5413BUB	Loss of unit Class III 600 V bus BUB	Y	Y	Y	Y	Y	
50	LO5413BUC	Loss of unit Class III 600 V bus BUC	Y	Y	Y	Y	Y	
51	LO5413BUD	Loss of unit Class III 600 V bus BUD	Y	Y	Y	Y	Y	

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
52	LO5413BUE	Loss of unit Class III 600 V bus BUE	Y	Y	Y	Y	Y	
53	LO5423BUA	Loss of unit Class II 600 V bus BUA	Y	Y	Y	Y	Y	See loss of unit Class IV 4.16 kV above.
54	LO5423BUB	Loss of unit Class II 600 V bus BUB	Y	Y	Y	Y	Y	
55	LO5424BUA	Loss of unit Class II 120 V ac bus BUA	Y	Y	Y	Y	Y	See loss of unit Class IV 4.16 kV above.
56	LO5424BUB	Loss of unit Class II 120 V ac bus BUB	Y	Y	Y	Y	Y	
57	LO5424BUC	Loss of unit Class II 120 V ac bus BUC	Y	Y	Y	Y	Y	
58	LO5424BUD	Loss of unit Class II 120 V ac bus BUD	Y	Y	Y	Y	Y	
59	LO5424BUE	Loss of unit Class II 120 V ac bus BUE	Y	Y	Y	Y	Y	
60	LO5424BUF	Loss of unit Class II 120 V ac bus BUF	Y	Y	Y	Y	Y	
61	LO5424BUG	Loss of unit Class II 120 V ac bus BUG	Y	Y	Y	Y	Y	
62	LO5424BU1A	Loss of unit Class II 120 V ac bus BU1A	Y	Y	Y	Y	Y	
63	LO5424BU1B	Loss of unit Class II 120 V ac bus BU1B	Y	Y	Y	Y	Y	
64	LO5424BU1C	Loss of unit Class	Y	Y	Y	Y	Y	

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
		II 120 V ac bus BU1C						
65	LO5424BU1D	Loss of unit Class II 120 V ac bus BU1D	Y	Y	Y	Y	Y	
66	LO5424BU1E	Loss of unit Class II 120 V ac bus BU1E	Y	Y	Y	Y	Y	
67	LO5424BU1F	Loss of unit Class II 120 V ac bus BU1F	Y	Y	Y	Y	Y	
68	LO5424BU1G	Loss of unit Class II 120 V ac bus BU1G	Y	Y	Y	Y	Y	
69	LO5424BU2C	Loss of unit Class II 120 V ac bus BU2C	Y	Y	Y	Y	Y	
70	LO5424BU2D	Loss of unit Class II 120 V ac bus BU2D	Y	Y	Y	Y	Y	
71	LO5425BU1	Loss of unit Class II 48 V dc bus LO5425BU1	Y	Y	Y	Y	Y	See loss of unit Class IV 4.16 kV above.
72	LO5425BU2	Loss of unit Class II 48 V dc bus LO5425BU2	Y	Y	Y	Y	Y	
73	LO5425BU3	Loss of unit Class II 48 V dc bus LO5425BU3	Y	Y	Y	Y	Y	
74	LO5425BU4	Loss of unit Class II 48 V dc bus LO5425BU4	Y	Y	Y	Y	Y	
75	LO5425BU5	Loss of unit Class II 48 V dc bus LO5425BU5	Y	Y	Y	Y	Y	

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
76	LO5425BU6	Loss of unit Class II 48 V dc bus LO5425BU6	Y	Y	Y	Y	Y	
77	LO5425BU7	Loss of unit Class II 48 V dc bus LO5425BU7	Y	Y	Y	Y	Y	
78	LO5425BU8	Loss of unit Class II 48 V dc bus LO5425BU8	Y	Y	Y	Y	Y	
79	LO5425BU9	Loss of unit Class II 48 V dc bus LO5425BU9	Y	Y	Y	Y	Y	
80	LO5425BU10	Loss of unit Class II 48 V dc bus LO5425BU10	Y	Y	Y	Y	Y	
81	LO5425BU11	Loss of unit Class II 48 V dc bus LO5425BU11	Y	Y	Y	Y	Y	
82	LO5425BU12	Loss of unit Class II 48 V dc bus LO5425BU12	Y	Y	Y	Y	Y	
83	LO5425BU13	Loss of unit Class II 48 V dc bus LO5425BU13	Y	Y	Y	Y	Y	
84	LO5425BU14	Loss of unit Class II 48 V dc bus LO5425BU14	Y	Y	Y	Y	Y	
85	LO5425BU15	Loss of unit Class II 48 V dc bus LO5425BU15	Y	Y	Y	Y	Y	
86	LO5425BU16	Loss of unit Class II 48 V dc bus LO5425BU16	Y	Y	Y	Y	Y	
87	LO5425BU17	Loss of unit Class II 48 V dc bus	Y	Y	Y	Y	Y	

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
		LO5425BU17						
88	LO5425BU18	Loss of unit Class II 48 V dc bus LO5425BU18	Y	Y	Y	Y	Y	
89	LO5425BU19	Loss of unit Class II 48 V dc bus LO5425BU19	Y	Y	Y	Y	Y	
90	LO5425BU20	Loss of unit Class II 48 V dc bus LO5425BU20	Y	Y	Y	Y	Y	
91	LO5425BU21	Loss of unit Class II 48 V dc bus LO5425BU21	Y	Y	Y	Y	Y	
92	LO5425BU22	Loss of unit Class II 48 V dc bus LO5425BU22	Y	Y	Y	Y	Y	
93	LO5425BU23	Loss of unit Class II 48 V dc bus LO5425BU23	Y	Y	Y	Y	Y	
94	LO5425BU31	Loss of unit Class II 48 V dc bus LO5425BU31	Y	Y	Y	Y	Y	
95	LO5425BU32	Loss of unit Class II 48 V dc bus LO5425BU32	Y	Y	Y	Y	Y	
96	LO5425BU33	Loss of unit Class II 48 V dc bus LO5425BU33	Y	Y	Y	Y	Y	
97	LO5425BU34	Loss of unit Class II 48 V dc bus LO5425BU34	Y	Y	Y	Y	Y	
98	LO5425BU35	Loss of unit Class II 48 V dc bus LO5425BU35	Y	Y	Y	Y	Y	

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
99	LO5425BU36	Loss of unit Class II 48 V dc bus LO5425BU36	Y	Y	Y	Y	Y	
100	LO5425BU37	Loss of unit Class II 48 V dc bus LO5425BU37	Y	Y	Y	Y	Y	
101	LO5425BU38	Loss of unit Class II 48 V dc bus LO5425BU38	Y	Y	Y	Y	Y	
102	LO5425BU39	Loss of unit Class II 48 V dc bus LO5425BU39	Y	Y	Y	Y	Y	
103	LO5425BU40	Loss of unit Class II 48 V dc bus LO5425BU40	Y	Y	Y	Y	Y	
104	LO5425BU41	Loss of unit Class II 48 V dc bus LO5425BU41	Y	Y	Y	Y	Y	
105	LO5425BU42	Loss of unit Class II 48 V dc bus LO5425BU42	Y	Y	Y	Y	Y	
106	LO5425BU43	Loss of unit Class II 48 V dc bus LO5425BU43	Y	Y	Y	Y	Y	
107	LO5425BU44	Loss of unit Class II 48 V dc bus LO5425BU44	Y	Y	Y	Y	Y	
108	LO5425BU45	Loss of unit Class II 48 V dc bus LO5425BU45	Y	Y	Y	Y	Y	
109	LO5425BU46	Loss of unit Class II 48 V dc bus LO5425BU46	Y	Y	Y	Y	Y	
110	LO5425BU47	Loss of unit Class II 48 V dc bus	Y	Y	Y	Y	Y	

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
		LO5425BU47						
111	LO5425BU48	Loss of unit Class II 48 V dc bus LO5425BU48	Y	Y	Y	Y	Y	
112	LO5425BU49	Loss of unit Class II 48 V dc bus LO5425BU49	Y	Y	Y	Y	Y	
113	LO5425BU50	Loss of unit Class II 48 V dc bus LO5425BU50	Y	Y	Y	Y	Y	
114	LO5425BU51	Loss of unit Class II 48 V dc bus LO5425BU51	Y	Y	Y	Y	Y	
115	LO5425BU52	Loss of unit Class II 48 V dc bus LO5425BU52	Y	Y	Y	Y	Y	
116	LO250	Loss of unit Class I 250 V dc buses (odd and even)	Y	Y	Y	Y	Y	See loss of unit Class IV 4.16 kV above.
Initiating Events Related to Failures of Support Systems								
117	FOREBAY	Adverse forebay conditions	Y	Y	Y	Y	Y	The FOREBAY IE is defined as the presence of adverse conditions in the forebay, which may result in a degradation of the common (CCW and LPSW) and/or emergency (EWS) water systems. Such an event may be caused by frazil ice, algae runs, fish runs or excessive zebra mussel accumulation and may lead to various degrees of plugging of the main screenhouse and/or the EWS pumphouse.
118	LOLPSW	Total loss of low pressure service water	Y	Y	Y	N	Y	This IE may result from the failure of all LPSW pumps, check valves or strainers. The failure of the LPSW impacts several plant systems. The specific

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
								impact of the loss of the LPSW is modelled in mitigating system FTs and may result in either failure or reduced reliability of heat sinks (primary, back-up and emergency), HTS pressure and inventory control, or increased probability of an induced LOCA. The event is not applicable to POS D when the service water is unavailable due to scheduled maintenance.
119	LOHPSW	Total loss of high pressure service water	Y	Y	Y	N	Y	This IE may result from the failure of all HPSW pumps or check valves. The loss of HPSW affects a number of systems that for example rely on service water to provide cooling flow through heat exchangers. Effects of significance during the outage (POSs A/B/C/E) are main HTS pump stator cooling and loss of cooling to SDC heat exchangers or ACUs (used in HS#9). The event is not applicable to POS D when the service water is unavailable due to scheduled maintenance.
120	LORCW	Total loss of recirculated cooling water system flow	Y	Y	Y	N	Y	This IE may result from the failure of all RCW pumps or check valves. The total loss of recirculated cooling water event results in overheating the HTS pump motor bearing and seal housing and loss of cooling to gland recirculation HXs. The event is not applicable to POS D when the service water is unavailable due to scheduled maintenance.

	Outage IE Label	IE Definition	POS Applicability					Discussion
			A	B	C	D	E	
121	TLOUIA	Total loss of unit instrument air	Y	Y	Y	Y	Y	The IE represents failure of the compressed air supply to provide sufficient quantity of air to the required pneumatic loads at the necessary minimum pressure. This IE may be caused by failure of instrument air compressors or breaks in the air distribution headers and will result in various air operated control valves and motorized valves failing to their default position potentially challenging the HTS boundary or effectiveness of declared heat sinks while the reactor is in GSS.
Initiating Event Related to ECI Blowback								
122	ECIBB	Emergency Coolant Injection Blowback	Y	Y	Y	Y	Y	The ECIBB event is defined as inadvertent opening of various valves in the Emergency Coolant Injection (ECI) system establishing a flow path between any one of the four system quadrants and the low pressure portion of the ECI piping.
Initiating Event Related to Power House Freeze								
123	PHFREEZE	Powerhouse Freezing during an Extreme Cold Outside Condition	Y	Y	Y	Y	Y	This event represents the situation following a spurious opening of the powerhouse panels during an extreme cold outside condition. This could result in freezing of standing water inside the powerhouse, hence, a potential impact on operating and standby mitigating systems.

Table 16: Summary of Fuel Damage Categories for PBRA-L1O

FDC	Definition	Typical Events in FDC
FDC1-SD	Rapid loss of core structural integrity.	Positive reactivity transient during shutdown and failure to terminate the event.
FDC2-SD	Slow loss of core structural integrity.	LOCA with failure of HTS inventory makeup and failure of moderator heat sink.
FDC5-SD	Moderator required as heat sink in the long term (> 24 hr after reactor shutdown).	LOCA1 and failures of D2O make up and Emergency Coolant Recovery (ECR).
FDC7-SD	Single channel fuel failure with sufficient release of steam or radioactivity to initiate automatic containment button-up.	In-core LOCA and fuel ejection. Large flow blockage. LOCA1 stagnation feeder break.
FDC9-SD	LOCAs with no fuel failure (ECIS successful); potential for significant economic impact.	LOCA1 with no D2O makeup.

Table 17: Seismic Hazard Bins

BIN	Bin Seismic Range (g)	Bin Average Acceleration (g)	Seismic Bin Frequency
1	0.01 – 0.05	0.02	3.4E-03
2	0.05 – 0.10	0.07	3.3E-04
3	0.10 – 0.20	0.14	1.4E-04
4	0.20 – 0.30	0.24	3.9E-05
5	0.30 – 0.50	0.39	2.7E-05
6	0.50 – 0.70	0.59	9.4E-06
7	0.70 – 1.00	0.84	5.9E-06
8	1.00 – 2.00	1.41	5.0E-06
9	> 2.00	2.00	1.6E-06

Table 18: Summary of Selected Accident Sequences

Plant Damage States	Representative Sequence
PDS1	No representative sequence required.
PDS2A	PTF, with loss of moderator cooling and failure of ECI.
PDS2B	PTF, with loss of moderator cooling and failure of ECI, combined with hydrogen ignition system failure.
PDS2C	PTF, with loss of moderator cooling and failure of ECI, combined with boiler room ACU failure.
PDS2D	PTF, with loss of moderator cooling and failure of ECI, combined with boiler room ACU failure and with hydrogen ignition system failure.
PDS2E	PTF, with loss of moderator cooling and failure of ECI, combined with failure of PRVs.
PDS2F	PTF, with loss of moderator cooling and failure of ECI, combined with failure of PRVs and boiler room ACU failure.
PDS2G	PTF, with loss of moderator cooling and failure of ECI, combined with containment envelope impairment.
PDS2H	PTF, with loss of moderator cooling and failure of ECI, combined with containment envelope impairment and boiler room ACU failure.
PDS3-4U	Main steam line break outside containment, combined with failures causing station blackout, leading to a loss of heat sink and failure of ECI and moderator cooling at four units simultaneously, with interim boiler heat sinks failed at all four units.
PDS3-4U-NO-INTERIM	Main steam line break outside containment, combined with failures causing station blackout, leading to a loss of heat sink and failure of ECI and moderator cooling at four units simultaneously, with interim boiler heat sinks failed at all four units.
PDS3-6	Main steam line break outside containment, with PEVS failure, leading to a loss of heat sink and failure of ECI and moderator cooling at six units simultaneously.
PDS4	No representative sequence required.
PDS5	LOCA2 combined with failure of ECI, with the moderator providing a long term heat sink.
PDS6	Multiple steam generator tube rupture combined with failure of ECI, with the moderator providing a long term heat sink.

Table 19: PNGS-B Release Categorization Scheme

Release Category #	Description	Definition
RC1	Very large release with potential for acute offsite radiation effects and/or widespread contamination	Release containing > 2-3% core inventory of I-131/Cs-137
RC2	Early release in excess of “Large Release” definition	Mixture of fission products containing > 1E14 Bq of Cs-137 but less than RC1 occurring mainly within 24 hours
RC3	Late release in excess of “Large Release” definition	Mixture of fission products containing > 1E14 Bq of Cs-137 but less than RC1 occurring mainly after 24 hours
RC4	Early release in excess of “Small Release” definition	Mixture of fission products containing > 1E15 Bq of I-131 but < 1E14 Bq of Cs-137 occurring mainly within 24 hours
RC5	Late release in excess of “Small Release” definition	Mixture of fission products containing > 1E15 Bq of I-131 but < 1E14 Bq of Cs-137 occurring mainly after 24 hours
RC6	Greater than normal containment leakage below “Small Release” limit	Mixture of fission products containing > 1E14 Bq of I-131 but < 1E15 Bq of I-131
RC7	Normal containment leakage	Leakage across an intact containment envelope or long-term filtered release
RC8	Basemat Melt-through	No release to atmosphere

Table 20: Summary of PBRA Severe Core Damage Frequency and Large Release Frequency Results for Internal Events

Model	SCDF (occurrences per reactor year)	LRF (occurrences per reactor year)
Internal Events At-Power	1.0E-06	8.0E-07
Internal Events Outage	1.0E-06 [1] 3.2E-07 [2]	4.3E-07
OPG Safety Goal	1E-04	1E-05

[1] SCDF for moderator drained GSS with guaranteed hole, or moderator drained RBGSS where outage activities prevent timely emergency restoration of the moderator pressure boundary.

[2] SCDF for RBGSS with drained moderator.

Table 21: PBRA Level 1 At-Power Internal Events Fuel Damage Results

Fuel Damage Category	Predicted Frequency (/year)
FDC1	< 1.0E-09
FDC2	1.0E-06
FDC3	4.5E-05
FDC4	4.6E-05
FDC5	1.9E-06
FDC6	2.0E-06
FDC7	3.2E-03
FDC8	1.3E-03
FDC9	3.0E-02
Severe Core Damage FDC1 + FDC2	**1.0E-06**

Table 22: Plant Damage State Frequency

Plant Damage State	Predicted Frequency (/year)
PDS1	6.9E-10
PDS2	6.5E-07
PDS3-4U	2.2E-07
PDS3-4U-NO-INTERIM	3.2E-08
PDS3-6U	6.1E-07
PDS4	2.3E-11
PDS5	3.2E-03
PDS6	9.3E-07

Table 23: Release Category Frequency for PBRA L2P

Release Category	Frequency (/year)
RC1	8.0E-07
RC2	0
RC3	0
RC4*	0
RC5*	0
RC6	3.1E-07
RC7	5.0E-07
RC8*	0
* The RC results with a zero value occur because only Containment Event Tree sequences with zero probability go to those end states regardless of the truncation level.	

Table 24: Summary of PBRA SCDF and LRF Results for Internal Fire, Seismic, Internal Flooding and High Wind Events for Pickering B Reactors

Model	SCDF (occurrences per reactor year)	LRF (occurrences per reactor year)
Fire At-Power	7.75E-07	4.03E-07
Seismic At-Power	1.3E-07	1.3E-07[1]
Flooding At-Power	2.0E-07	9.6E-08
High Wind At-Power	9.9E-06	5.9E-06

1 The seismically-induced containment failure frequency (SCFF) is estimated to be 4.6E-08/year; however, a Level 2 model has not been developed and, therefore, the LRF is reported as equivalent to SCDF.

Table 25: Summary of PBRA LRF for Non-Reactor Sources Events

Hazard[1]	LRF (occurrences per year)
Internal events	2.6E-07
Seismic events	1.4E-08

1 The results listed in the table above are due to hazards directly affecting the IFB. The hazards associated with the UFDS facility have been screened out.

REQUEST FOR CONFIDENTIALITY

OF MATERIAL SUBMITTED IN RELATION TO CMD 24-H5

In the matter of:

Ontario Power Generation - Request for Authorization to Operate Pickering Nuclear Generating Station Units 5-8 until 2026

This request has been prepared in Canada, in the province of Ontario, in the matter of *Ontario Power Generation - Request for Authorization to Operate Pickering Nuclear Generating Station Units 5-8 until 2026***,** scheduled for consideration in a public hearing, scheduled for June 2024.

I, Riedewaan Bakardien, Senior Vice President of 1675 Montgomery Park Road, Pickering, Ontario L1V 2R5, am an authorized representative of Ontario Power Generation Pickering Nuclear Generating Station. 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 protection 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**
1.	OPG Letter, A Cipolla to R. Richardson, A. Mathai, and N. Greencorn, "OPG - Submission of the 2023 Annual Report for the 2023-2027 CNSC Financial Guarantee", February 27, 2023, CD# N-CORR-00531-23486, e-Doc 6990272	✓ Entire content ☐ Redacted content as shown

This request is made pursuant to the following paragraph(s) of rule 12 of the CNSC Rules of Procedure:

- Rule 12 (1) (b) the information is confidential information of a financial, commercial, scientific, technical, personal or other nature that is treated consistently as confidential and the person affected has not consented to the disclosure.

Further,

1. The above-noted material should be protected for the following reasons:
   - Rule 12 (1) (b) – The correspondence *OPG - Submission of the 2023 Annual Report for the 2023-2027 CNSC Financial Guarantee* contains information of a financial nature that is treated consistently as confidential and OPG has not consented to its disclosure.
2. I attest that the above-noted material is not available through any public sources.
3. 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.

4. 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).

5. 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.

6. 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:

- OPG Letter, A. Cipolla to R. Richardson, A. Mathai, and N. Greencorn, "OPG - Submission of the 2023 Annual Report for the 2023-2027 CNSC Financial Guarantee", February 27, 2023, CD# N-CORR-00531-23486, e-Doc 6990272.
- Non-Confidential Summary: OPG Report, "Documentary Information Summary 2023 – 2027 CNSC Financial Guarantee", April 29, 2022, CD# W-REP-00400-10048.

Authorized signature:

Riedewaan Bakardien,
Senior Vice President,
Pickering Nuclear Generating Station

2024/03/28
Date

CD# W-REP-00400-10048

DOCUMENTARY INFORMATION SUMMARY
2023 – 2027 CNSC FINANCIAL GUARANTEE

A Submission to the Canadian Nuclear Safety Commission

In Support of

Licence Conditions for the Purpose of Decommissioning Plans and Financial Guarantee for Nuclear Generating Stations and Waste Management Facilities Owned by OPG

April 2022

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Ontario Power Generation, Inc.

Approved by: April 29, 2022
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Ontario Power Generation, Inc.

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1.0 EXECUTIVE SUMMARY

Ontario Power Generation Inc. ("OPG"), owner of nuclear facilities in Ontario, is required to have in place a financial guarantee acceptable to the Canadian Nuclear Safety Commission ("CNSC") in support of these nuclear facilities' licence conditions. The financial guarantee is required to be updated by OPG every five years and reflects all nuclear waste produced to date and the decommissioning required for all OPG nuclear facilities. OPG's consolidated CNSC financial guarantee currently in place is valid until December 31, 2022. This Documentary Information Summary ("2022 DIS") report summarizes OPG's proposed consolidated financial guarantee for its nuclear facilities for the next five years, 2023 to 2027 ("2023 – 2027 CNSC Financial Guarantee").

The total CNSC financial guarantee requirement ("Total CNSC Requirement") is based on the present value of cost estimates for nuclear waste management and decommissioning as of year-end in any given year, aggregated in respect of all OPG nuclear facilities. The proposed Total CNSC Requirement for 2023 is $20,480 million (January 1, 2023 present value dollars) and increases to $22,303 million (January 1, 2027 present value dollars) in 2027.

It is proposed that the Total CNSC Requirement would continue to be satisfied, collectively, by the federally mandated Ontario Nuclear Fuel Waste Act Trust ("Ontario NFWA Trust") and by providing the CNSC with access to the two segregated funds governed by the Ontario Nuclear Funds Agreement ("ONFA") between OPG and the Province of Ontario. The Ontario NFWA Trust and the segregated funds (collectively, the "Nuclear Funds") are projected to have a fair market value in excess of the proposed Total CNSC Requirement throughout the period. The fair market value of the Nuclear Funds is projected to be $25,148 million on January 1, 2023 increasing to $28,250 million by January 1, 2027.

Further details are provided in the body of this document. OPG will continue to provide an annual status report to CNSC staff updating nuclear waste management and decommissioning plans and cost estimates and indicating how the Total CNSC Requirement will be satisfied.

2.0 INTRODUCTION

The 2023 – 2027 CNSC Financial Guarantee associated with nuclear waste management and decommissioning is in support of licence conditions for the nuclear generating stations and nuclear waste management and other nuclear facilities owned by OPG. The nuclear generating stations are Pickering and Darlington, operated by OPG, and Bruce, leased to and operated by Bruce Power L.P ("BP"). The other nuclear facilities are Pickering, Western, and Darlington Waste Management Facilities, the Radioactive Waste Operations Site 1 ("RWOS1"), the Central Storage Facility ("CSF")[1], and the Central Maintenance Facility ("CMF"). The CSF is a newly constructed facility by BP to store standard shipping containers containing contaminated tooling, equipment, and components in preparation for Major Component Replacement outages at the Bruce nuclear generating stations. The Spent Solvent Treatment Facility ("SSTF"), which was included in the latest Documentary Information Submission ("2017 DIS") that covered the 2018 to 2022 period ("2018 – 2022 CNSC Financial Guarantee"), has been decommissioned and therefore is not included in the proposed 2023 – 2027 CNSC Financial Guarantee.

Under agreements with BP, OPG retains responsibility for decommissioning of the Bruce nuclear generating stations and other nuclear facilities leased to BP and management of used nuclear fuel and low and intermediate level waste ("L&ILW") produced by the Bruce nuclear generating stations. OPG is responsible for providing the CNSC financial guarantee required to cover the liability for nuclear waste management and decommissioning related to the Bruce nuclear generating stations and other nuclear facilities leased to BP by OPG.

An initial Documentary Information Summary was completed in July 2003 along with supporting legal agreements. Since the initial submission, there have been subsequent updates, with the latest submission being the 2017 DIS. In addition, annual reports have been submitted to the CNSC to provide the status of the financial guarantee, detailing amounts accumulated in the Nuclear Funds and any material changes in preliminary decommissioning plans and nuclear waste management plans, nuclear waste quantities or cost estimates which might impact the Total CNSC Requirement.

This submission is filed to update projected present value cost estimates of OPG's proposed preliminary decommissioning plans and nuclear waste management plans, and the associated consolidated financial guarantee for the 2023 to 2027 period. This submission would also serve as a baseline for future CNSC financial guarantee reporting as required by OPG's licence conditions. The updated cost estimates in this submission are based on the baseline cost estimates and underlying planning and economic assumptions of the current reference plan under the ONFA.

A listing of supporting documentation for preliminary decommissioning plans, nuclear waste management plans and cost estimates as well as a proposed amendinglegal agreement granting CNSC access to the ONFA segregated fundsis included as Appendix C to this document.

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[1] The Central Storage Facility is also referred to as the Contaminated Tools Storage Facility or the Contaminated Tooling Storage Facility

An analysis of changes in the Total CNSC Requirement from the 2017 DIS is presented in Appendix D.

3.0 NUCLEAR WASTE MANAGEMENT AND DECOMMISSIONING PLANS

Nuclear waste management and decommissioning plans include interim storage and disposal plans for used nuclear fuel and L&ILW arising from the operation of OPG owned facilities, as well as the decommissioning of OPG nuclear generating stations as well as nuclear waste management and other nuclear facilities.

Used Nuclear Fuel Management Plans

OPG's used nuclear fuel management reference plans include interim storage of used fuel at each nuclear generating station site ("Used Fuel Storage Program") until such time that a national long-term management facility is available ("Used Fuel Long-Term Management Program").

The Used Fuel Storage Program encompasses interim storage of used nuclear fuel in dry storage containers in a dry storage facility after initially being stored in wet bays for a minimum of 10 years. At the time of station shutdown, used nuclear fuel would remain in the wet bays to allow the fuel to be adequately cooled before it is either transferred to dry storage or retrieved for shipment to the permanent disposal facility.

The Used Fuel Long-Term Management Program encompasses the retrieval, transportation and permanent emplacement of used nuclear fuel for long-term management, which is based on the Adaptive Phased Management ("APM") concept accepted by the Government of Canada on recommendation of the Nuclear Waste Management Organization ("NWMO") in response to the *Nuclear Fuel Waste Act* (Canada). The NWMO is responsible for the design and implementation of Canada's plan for the safe long-term management of used nuclear fuel. The APM approach includes the isolation and containment of used nuclear fuel in a deep geologic repository ("Used Fuel DGR") after a collaborative process of communication and engagement with Canadians aimed at selecting a suitable geological site with an informed and willing host community. The NWMO is in the process of undertaking a site selection for the Used Fuel DGR. The proposed Total CNSC Requirement is based on the NWMO's bounding site selection scenario for cost estimating purposes and assumes an in-service date of 2043 for the Used Fuel DGR. The Used Fuel Long-Term Management Program includes OPG's portion of NWMO's costs toward the development and implementation of the Used Fuel DGR.

Low and Intermediate Level Waste Management Plans

OPG's L&ILW management reference plans include interim storage of operational L&ILW at the Western Waste Management Facility ("WWMF") situated at the Bruce nuclear site ("L&ILW Operations Program") prior to the permanent emplacement of these wastes into the assumed L&ILW long-term disposal facilities away from the WWMF ("L&ILW Long-Term Management Program").

The L&ILW Operations Program encompasses activities to transport, process, package and interim store operational L&ILW in adherence to the waste acceptance criteria to allow for emplacement into the long-term disposal facilities. The majority of OPG's operational low level waste (“LLW”) is transported to the WWMF and is received at the waste volume reduction facilities where it is sorted and either processed (i.e., incineration or compaction) to achieve volume reduction or interim stored as is (i.e., non-processible waste) in storage buildings. Operational intermediate level waste (“ILW) is also stored centrally at the WWMF within above or in-ground storage structures.

The L&ILW Long-Term Management Program reference plans in the 2017 DIS were based on the retrieval of operational L&ILW for emplacement in OPG’s proposed L&ILW deep geologic repository (“L&ILW DGR”) adjacent to the WWMF. In January 2020, Saugeen Ojibway Nation (“SON”) community members voted not to support this proposed project. OPG upheld its commitment not to proceed with the L&ILW DGR without SON support and has cancelled the project. At this time, OPG is awaiting the results of Natural Resources Canada’s process to modernize Canada’s Radioactive Waste Policy, which includes developing Canada’s integrated strategy for L&ILW. For financial planning purposes, the proposed 2023 – 2027 CNSC Financial Guarantee is based on a conceptual assumption consisting of transportation and emplacement of operational LLW into a near surface disposal facility located away from the WWMF and transportation and emplacement of operational ILW into an expanded Used Fuel DGR.

It should be noted that the above long-term disposal facilities are assumed for cost estimating purposes only, and no project or site selection process has commenced. OPG’s future planning assumptions for the L&ILW Long-Term Management Program will be informed by the outcomes of the Natural Resources Canada’s process to modernize Canada’s Radioactive Waste Policy Framework.

Nuclear Generating Station Decommissioning Plans

Nuclear generating station decommissioning plans are based on a deferred dismantling strategy which assumes a nominal 30-year safe storage period following the shutdown of the final reactor at each station (“Decommissioning Program”). The current nuclear generating stations’ end-of-life assumptions include the refurbishment of Darlington and Bruce and the operation of Pickering to the end of 2025 subject to the CNSC’s regulatory approval, as summarized in Table 1 below. The assumed long-term disposal facilities for operational L&ILW would also accommodate L&ILW arising from station decommissioning activities.

Appendix B shows the assumed nuclear generating station decommissioning timelines from the end of operations through the safe storage period and subsequently through the dismantling period. Also shown are the reference dates for financial planning purposes for the transportation and long-term management of used nuclear fuel.

Table 1: Nuclear Generating Station End of Life Dates Assumed for the Proposed 2023 – 2027 CNSC Consolidated Financial Guarantee

Nuclear Generating Station	Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6	Unit 7	Unit 8
Pickering	2024	2005	2005	2024	2025	2025	2025	2025
Bruce	2043	2043	2061	2062	2062	2058	2063	2063
Darlington	2055	2050	2053	2056	N/A	N/A	N/A	N/A

Nuclear Waste Management and Other Nuclear Facility Decommissioning Plans

The nuclear waste management and other nuclear facilities will be decommissioned after all corresponding nuclear waste and/or used nuclear fuel has been transferred to the long-term disposal facilities. The sites will then be restored and made available for re-use.

4.0 NUCLEAR WASTE MANAGEMENT AND DECOMMISSIONING COST ESTIMATES

In 2021, OPG completed a comprehensive update of the estimate for its obligations for nuclear waste management and nuclear facilities decommissioning which forms the basis of OPG's proposed Total CNSC Requirement in this submission.

As in prior submissions, cost estimates of all nuclear waste management and nuclear facilities decommissioning programs are first prepared in constant dollars, which assume that expenditures occur at the time of estimate preparation. These estimates are then escalated to the scheduled expenditure period using economic forecasts prepared by external experts. Consistent with the 2017 DIS and prior submissions, macroeconomic forecasts from the University of Toronto's Institute of Policy Analysis Economic Forecasting Series ("University of Toronto forecast") continue to be applied as escalators in this submission. The escalation rates applied against the cost estimates in this submission are based on the University of Toronto forecast for Ontario dated February 2021 and are applied to each year of the programs' cost flows. The Ontario forecast is used as the source of the escalation rates given the majority of the work programs is expected to be executed in Ontario. The long-term escalation rate is 3.4% for labour cost, 2.1% for material and equipment costs and 2.0% for other costs. Refer to Appendix E for the annual escalation rates used for each of the three cost categories.

Subsequent to the finalization of the cost estimates, a sensitivity analysis was conducted using the most recent available University of Toronto forecast for Ontario dated February 2022 to confirm that the February 2022 forecast would not result in a higher Total CNSC Requirement.

Consistent with the 2017 DIS, a discount rate of 5.15% is applied to the estimates in escalated dollars in order to determine the present value ("PV") of future costs. This rate is consistent with the discount rate employed under the Ontario Nuclear Funds Agreement and represents the target rate of return on investments held in the Nuclear Funds.

The cost estimates for nuclear waste management and nuclear facilities decommissioning from January 1, 2023 onwards in 2023 constant and present value dollars are summarized in Table 2 below. For nuclear waste management programs, Table 2 reflects OPG's forecasted quantities of operational L&ILW and used nuclear fuel as at December 31, 2023.

Table 2: Proposed 2023 Financial Guarantee Cost Estimate

Program	Cost Estimate	
	2023 Constant M$	Jan. 1, 2023 PV M$
Decommissioning OPG-owned Nuclear Generating Stations	18,066	7,300
Used Fuel Management	24,301	11,073
Low and Intermediate Level Waste Management	3,462	2,002
Decommissioning Pickering Waste Management Facility (PWMF)	39	17
Decommissioning Western Waste Management Facility (WWMF)	158	56
Decommissioning Darlington Waste Management Facility (DWMF)	28	7
Decommissioning RWOS1, CMF and CSF	63	25
Total*	**46,117**	**20,480**

*Details may not add to total due to rounding.

The nuclear generating station decommissioning cost estimates from January 1, 2023 onwards on a station-by-station basis are summarized in Table 3 below in 2023 constant and present value dollars.

Table 3: Proposed 2023 Cost Estimates for Nuclear Generating Station Decommissioning

Nuclear Generating Station	Cost Estimate	
	2023 Constant M$	Jan. 1, 2023 PV M$
Pickering A (Units 1 - 4)	3,401	1,960
Pickering B (Units 5 - 8)	3,410	1,934
Bruce A (Units 1 - 4)	3,594	1,109
Bruce B (Units 5 - 8)	3,542	986
Darlington (Units 1 - 4)	4,119	1,312
Total*	**18,066**	**7,300**

*Details may not add to total due to rounding.

The estimated cost for the Used Fuel Storage Program and the Used Fuel Long-Term Management Program from January 1, 2023 onwards, based on year-end 2023 forecasted 3.0 million used nuclear fuel bundles, are summarized in Table 4 below in 2023 constant and present value dollars.

Table 4: Proposed 2023 Cost Estimates for Used Fuel Management

Used Fuel Management	Cost Estimate	
	2023 Constant M$	Jan. 1, 2023 PV M$
Used Fuel Storage Program	2,449	1,851
Used Fuel Long-Term Management Program	21,852	9,222
Total*	**24,301**	**11,073**

*Details may not add to total due to rounding.

The estimated cost for the L&ILW Operations Program and the L&ILW Long-Term Management Program from January 1, 2023 onwards, based on year-end 2023 forecasted disposal volume of 94,167 m^3 and 11,746 m^3 of LLW and ILW respectively, are summarized in Table 5 below in 2023 constant and present value dollars.

Table 5: Proposed 2023 Cost Estimates for L&ILW Management

Low and Intermediate Level Waste Management	Cost Estimate	
	2023 Constant M$	Jan. 1, 2023 PV M$
L&ILW Operations Program	1,184	898
L&ILW Long-Term Management Program	2,278	1,104
Total*	**3,462**	**2,002**

*Details may not add to total due to rounding.

The nuclear waste management and other nuclear facilities decommissioning cost estimates from January 1, 2023 onwards are summarized in Table 6 below in 2023 constant and present value dollars.

Table 6: Proposed 2023 Cost Estimates for Nuclear Waste Management and Other Nuclear Facility Decommissioning

Nuclear Waste Management Facility	Cost Estimate	
	2023 Constant M$	Jan. 1, 2023 PV M$
PWMF	39	17
WWMF	158	56
DWMF	28	7
RWOS1, CMF and CSF	63	25
Total*	**288**	**105**

*Details may not add to total due to rounding.

The cost estimates for nuclear waste management and nuclear facilities decommissioning from each of January 1, 2024, January 1, 2025, January 1, 2026 and January 1, 2027 onwards in corresponding year constant and present value dollars can be found in the Cost Estimate Summary Report for the 2023 - 2027 CNSC Financial Guarantee.

5.0 CNSC FINANCIAL GUARANTEE

As summarized in Section 4.0 of this report, the proposed Total CNSC Requirement for 2023 is $20,480 million (January 1, 2023 present value). It is proposed to be satisfied by the Nuclear Funds, which are projected to have a fair market value of $25,148 million as at January 1, 2023. The Nuclear Funds consist of the Ontario NFWA Trust and two segregated funds governed by the ONFA.

OPG established the Ontario NFWA Trust on November 15, 2002, in accordance with the requirements of the *Nuclear Fuel Waste Act* (Canada). OPG continues to make annual contributions to the Trust, as required under the Act.

The ONFA, entered into between OPG and the Province of Ontario, governs two segregated funds, the Decommissioning Segregated Fund (“DSF”) and the Used Fuel Segregated Fund (“UFSF”). These segregated funds were established in July 2003 and are held in the custodianship of financial institutions. The DSF was established to pay for costs associated with the Decommissioning Program, the L&ILW Long-Term Management Program, certain costs of the Used Fuel Storage Program incurred after the nuclear generating stations are shut down, and the costs of the L&ILW Operations Program incurred after the nuclear generating stations are shut down. The UFSF pays for the costs of the Used Fuel Long-Term Management Program and certain costs of the Used Fuel Storage Program after the nuclear generating stations are shut down.

CNSC access to segregated funds would continue to be provided in accordance with the CNSC Financial Security and ONFA Access Agreement between the CNSC, OPG and the Province of Ontario. While OPG’s access to the segregated funds under ONFA is limited to the purposes described above for each respective fund, the CNSC has the right to demand, in circumstances and on terms described in the agreement, payment of the balance of both funds up to the applicable Total CNSC Requirement.

The proposed annual Total CNSC Requirements for the 2023 to 2027 period are summarized in Table 7 below. Over the five-year period from 2023 to 2027, the proposed annual Total CNSC Requirement for a given year is based on estimated future program expenditures and includes year-over-year increases in the present value of the cost estimates to reflect the passage of time as well as estimated incremental costs to manage projected additional used nuclear fuel and L&ILW volumes generated by a given year-end.

Table 7: Proposed 2023 – 2027 Total CNSC Requirement

Year	Total CNSC Requirement M$
2023	20,480
2024	21,149
2025	21,764
2026	22,140
2027	22,303

The forecasted fair market value of the Nuclear Funds as of January 1 of each year is summarized in Table 8 below. The forecast is based on the assumed growth in the assets of the Nuclear Funds at the 5.15% target rate of return per annum. There are no planned contributions to the segregated funds during the period.

Table 8: 2023 – 2027 Forecasted Fair Market Value of the Nuclear Funds

Year	Nuclear Funds M$
2023	25,148
2024	26,102
2025	27,011
2026	27,768
2027	28,250

As illustrated in Tables 7 and 8 above, the proposed Total CNSC Requirement for each year of the 2023 to 2027 period is lower than the forecasted fair market value of the Nuclear Funds. As for 2023, it is proposed that the Total CNSC Requirement for years 2024 to 2027 be satisfied by the Nuclear Funds.

6.0 REPORTING

OPG will continue to provide an annual status report to CNSC staff detailing amounts accumulated in the Nuclear Funds. The report will also identify any material changes in preliminary decommissioning plans, nuclear waste management plans, waste quantities and cost estimates which may impact the Total CNSC Requirement and how this requirement is satisfied.

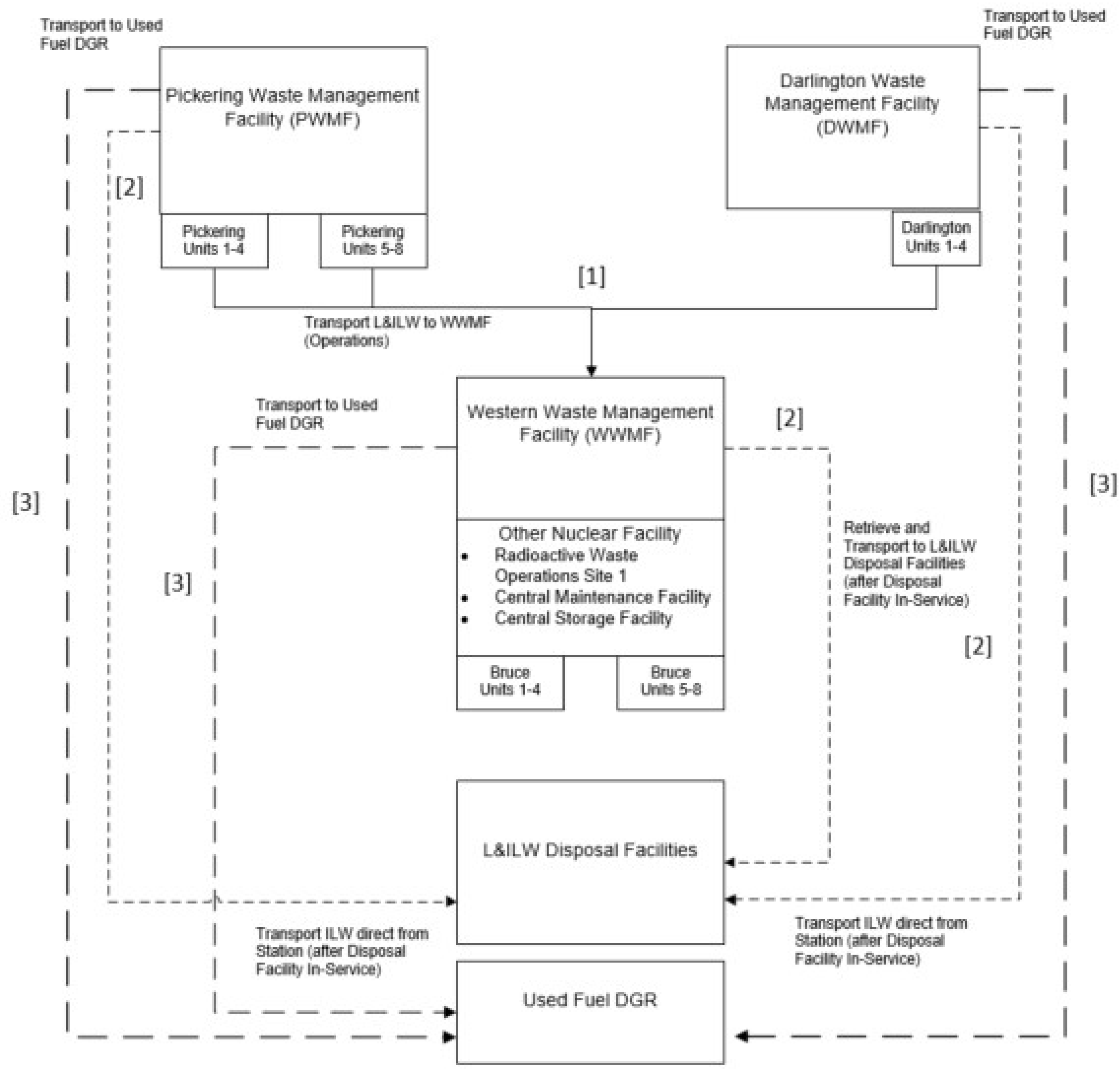
7.0 SUMMARY

OPG requests that the information presented above be considered as the Total CNSC Requirement for Pickering nuclear generating station, Darlington nuclear generating station, Bruce nuclear generating stations, Pickering Waste Management Facility, Western Waste Management Facility, Darlington Waste Management Facility, Radioactive Waste Operations Site 1, Central Storage Facility and Central Maintenance Facility in support of meeting CNSC financial guarantee licence conditions for OPG-owned nuclear facilities for the years 2023 to 2027. OPG proposes that the Total CNSC Requirement be satisfied by the Nuclear Funds pursuant to an amended CNSC Financial Security and ONFA Access Agreement.

8.0 SUPPORTING DOCUMENTATION

Documentation supporting the preliminary decommissioning plans, nuclear waste management plans, cost estimates and a legal agreement for the 2023 – 2027 CNSC Financial Guarantee is listed in Appendix C.

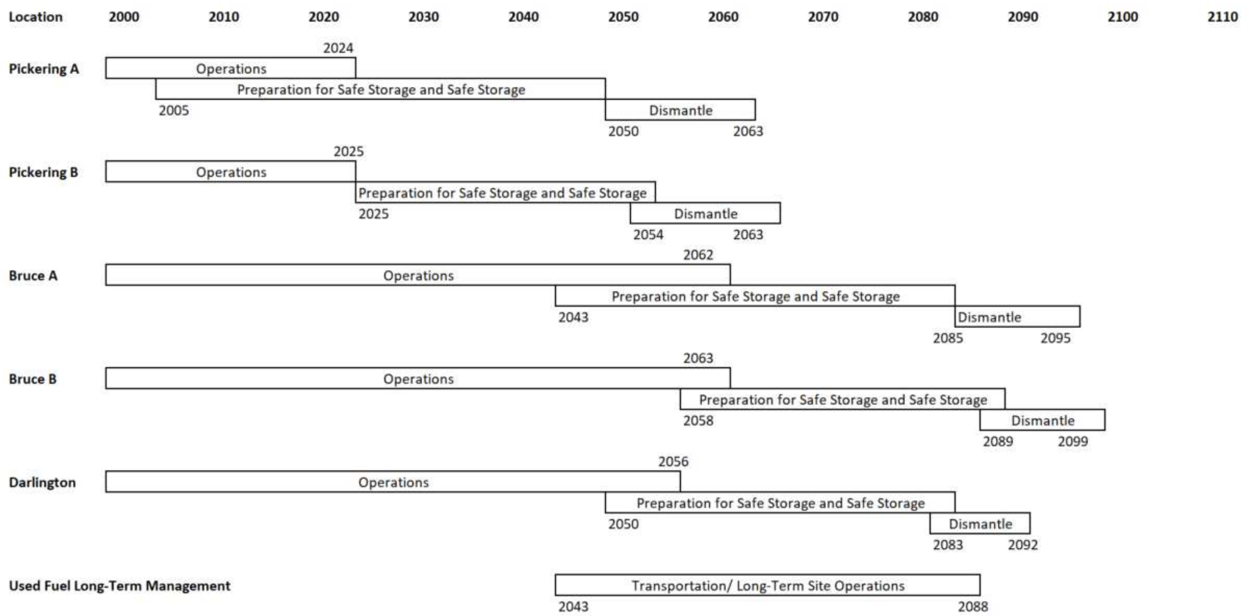
Appendix A: Nuclear Waste Management and Other Nuclear Facilities



Notes

[1] Prior to the in-service of the assumed long-term disposal facilities, low and intermediate level waste (L&ILW) will be transported from the Pickering and Darlington stations to the Western Waste Management Facility (WWMF) for processing (low level waste or LLW) and then interim stored.
[2] Interim storage of L&ILW would occur until the assumed long-term disposal facilities are in service. Subsequently, LLW would continue to be transported to the WWMF for processing but would not be interim stored. ILW would be transported directly from the nuclear stations to the long-term disposal facility, once in service.
[3] Used nuclear fuel stored at the waste management facilities will be transported to the Used Fuel DGR once in service.

Appendix B: Nuclear Generating Stations Decommissioning/ Used Fuel Management Timelines



Notes:

1) Operations timeline is to the end of operation of the final unit at each nuclear generating station.
2) Start dates shown for safe storage are for the first unit at each nuclear generating station.
3) Used nuclear fuel will be stored at nuclear generating station sites until the Used Fuel DGR is operational.
4) All dates are nominal and for financial planning purposes only.
5) Timelines represent execution only and do not include pre-planning activities. The dismantle timelines include site restoration.
6) Although Pickering A was shutdown in 1998, preparation for safe storage activities on Units 2 and 3 commenced in 2005 when the decision was made not to restart these units.

Appendix C: Support Documents Relating to Decommissioning Plans, Cost Estimates and CNSC Financial Guarantee

Documents Pertaining to all Licences

- Documentary Information Summary 2023 - 2027 CNSC Financial Guarantee, W-REP-00400-10048
- Cost Estimate Summary Report for the 2023 - 2027 CNSC Financial Guarantee, W-REP-00400-10047
- CNSC Financial Security and ONFA Access Agreement – Proposed Fifth Amending Agreement between the CNSC, the Province of Ontario and OPG, effective January 1, 2023 (Planned for submission in Q2 2022)

1. Darlington Nuclear Generating Station, PROL 13.03/2025 expires November 30, 2025

- Preliminary Decommissioning Plan - Darlington Nuclear Generating Station, Report No. NK38-PLAN-00960-10001-R003

2. Pickering Nuclear Generating Station, PROL 48.01/2028 expires August 31, 2028

- Preliminary Decommissioning Plan - Pickering Nuclear Generating Stations A & B, Report No. P-PLAN-00960-00001-R003

3. Bruce A and B Nuclear Generating Stations, including the Central Maintenance and Laundry Facility PROL 18.02/2028 expires September 30, 2028

- Preliminary Decommissioning Plan - Bruce Nuclear Generating Stations A & B, 06819-PLAN-00960-00001-R003

4. Western Waste Management Facility, WFOL-W4-314.00/2027 expires May 31, 2027

- Preliminary Decommissioning Plan - Western Waste Management Facility, 0125-PLAN-00960-00001-R004

5. Pickering Waste Management Facility, WFOL-W4-350.00/2028 expires August 31, 2028

- Preliminary Decommissioning Plan - Pickering Waste Management Facility, 92896-PLAN-00960-00001-R004

6. Darlington Waste Management Facility, WFOL-W4-355.01/2023 expires April 30, 2023

- Preliminary Decommissioning Plan - Darlington Waste Management Facility, 00044-PLAN-00960-00001-R005

7. Radioactive Waste Operations Site 1, WNSL-W1-320.05/2029 expires October 31, 2029

- Preliminary Decommissioning Plan – RWOS1, CMLF and CSF, W-PLAN-00960-00001-R002

Appendix D: Variance Analysis: 2018 – 2022 CNSC Financial Guarantee Submission to 2023 – 2027 CNSC Financial Guarantee Submission

The 2017 DIS for the accepted 2018 – 2022 CNSC Financial Guarantee submission contained the following annual Total CNSC Requirements as of January 1:

Year	Total CNSC Requirement M$
2018	16,468
2019	17,094
2020	17,722
2021	18,300
2022	18,836

The 2022 DIS for the proposed 2023 – 2027 CNSC Financial Guarantee contains the following proposed annual Total CNSC Requirements as of January 1:

Year	Total CNSC Requirement M$
2023	20,480
2024	21,149
2025	21,764
2026	22,140
2027	22,303

The following table compares the Total CNSC Requirement for year 2022 as contained in the 2017 DIS with the proposed Total CNSC Requirement for year 2023 as contained in the 2022 DIS. For the purposes of this comparison, the Total CNSC Requirement for year 2022 of $18,836 million in January 1, 2022 present value dollars is normalized to January 1, 2023 present value.

Program	2017 DIS 2018 – 2022 CNSC Financial Guarantee Year 2022	2022 DIS 2023 – 2027 CNSC Financial Guarantee Year 2023	Variance
	Jan. 1, 2023	Jan. 1, 2023	Jan. 1, 2023
	M$ PV*	M$ PV	M$ PV
Nuclear Generating Station Decommissioning	6,639	7,300	661
Used Fuel Management	10,755	11,073	318
L&ILW Management	2,085	2,002	(83)
Decommissioning PWMF	19	17	(2)
Decommissioning WWMF	42	56	14
Decommissioning DWMF	7	7	0
Decommissioning RWOS1, CMF and CSF	15	25	10
Total	**19,562**	**20,480**	**919**
Variance Breakdown:			
Changes in Future Economic Assumptions			**(99)**
Changes in Cost Estimates and Planning Assumptions			**1,017**

* To be comparable to the proposed financial guarantee cost estimates for year 2023, financial guarantee cost estimates for year 2022 shown are based on the 2018 - 2022 CNSC Financial Guarantee submission normalized to January 1, 2023 PV dollars and using available actual escalation factors.
**Details may not add to total due to rounding.

Variances for “Future Economic Assumptions” and “Cost Estimates and Planning Assumptions” are due to the following:

a) Future Economic Assumptions

The changes reflect updated University of Toronto Policy and Economic Analysis Program Policy Study forecast. While the long-term escalation rate for labour cost and other costs remains unchanged at 3.4% and 2.0%, respectively, and the long-term escalation rate for material and equipment increased from 2.0% in the 2017 DIS to 2.1% in the 2022 DIS, the net decrease of $99 million in the Total CNSC Requirement is mainly driven by lower near-term escalation rates for labour cost in the 2022 DIS.

b) Cost Estimates and Planning Assumptions

The net increase of $919 million in the Total CNSC Requirement is primarily attributable to changes in the following areas:

i) Nuclear Generating Station Decommissioning – the increase is primarily due to additional assessments for elements such as emptying the fuel bays, long-term management of heavy water and remediation of asbestos at Pickering, as well as a higher risk contingency included as part of the cost estimating process.

ii) Used Fuel Management – the increase primarily reflects NWMO staffing costs including resources to achieve the site selection milestone and transition to the regulatory decision making phase leading to becoming a qualified licencee, updated transportation costs based on the NWMO’s bounding scenario for Used Fuel DGR site selection, and continued development and demonstration of the engineered barrier system design and components for the Used Fuel DGR.

iii) L&ILW Management – the decrease mainly reflects the net impact of the new, conceptual long-term disposal facilities' assumption, longer duration of interim storage required until long-term disposal facilities are available, and expansion of OPG's waste minimization (volume reduction through processing) program.

Appendix E: University of Toronto Policy and Economic Analysis Program Policy Study Forecast – Escalation Factors

	February 2021		
Year	**Labour**	**Material & Equipment**	**Other**
2023	2.50%	2.00%	2.10%
2024	2.80%	2.10%	2.00%
2025	3.30%	2.00%	2.00%
2026	3.40%	2.00%	2.00%
2027	3.40%	2.00%	2.00%
2028	3.40%	2.00%	2.00%
2029	3.40%	2.00%	2.00%
2030	3.40%	2.00%	2.00%
2031	3.40%	2.10%	2.00%
2032	3.40%	2.00%	2.00%
2033	3.40%	2.10%	2.00%
2034	3.40%	2.10%	2.00%
2035	3.40%	2.10%	2.00%
2036	3.40%	2.10%	2.00%
2037	3.40%	2.10%	2.00%
2038	3.40%	2.10%	2.00%
2039	3.40%	2.10%	2.00%
2040	3.40%	2.10%	2.00%
2041	3.40%	2.10%	2.00%
2042	3.40%	2.10%	2.00%
2043	3.40%	2.10%	2.00%
2044	3.40%	2.10%	2.00%
2045	3.40%	2.10%	2.00%
2046	3.40%	2.10%	2.00%
2047	3.40%	2.10%	2.00%
2048	3.40%	2.10%	2.00%
2049	3.40%	2.10%	2.00%
2050	3.40%	2.10%	2.00%

For years beyond the above forecast period, the escalation factors for year 2050 are applied.

REQUEST FOR CONFIDENTIALITY

OF MATERIAL SUBMITTED IN RELATION TO CMD 24-H5

In the matter of:

Ontario Power Generation - Request for Authorization to Operate Pickering Nuclear Generating Station Units 5-8 until 2026

This request has been prepared in Canada, in the province of Ontario, in the matter of *Ontario Power Generation - Request for Authorization to Operate Pickering Nuclear Generating Station Units 5-8 until 2026***,** scheduled for consideration in a public hearing, scheduled for June 2024.

I, Riedewaan Bakardien, Senior Vice President of 1675 Montgomery Park Road, Pickering, Ontario L1V 2R5, am an authorized representative of Ontario Power Generation Pickering Nuclear Generating Station. 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 protection 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**
1.	OPG Letter, J. Franke to R. Richardson, "Pickering NGS Units 5 to 8: Updated Safety Report - Parts 1 and 2: Facility Description", October 20, 2022, CD# NK30-CORR-00531-08586.	✓ Entire content ☐ Redacted content as shown

This request is made pursuant to the following paragraph(s) of rule 12 of the CNSC Rules of Procedure:

- Rule 12 (1) (b) the information is confidential information of a financial, commercial, scientific, technical, personal or other nature that is treated consistently as confidential and the person affected has not consented to the disclosure.

Further,

1. The above-noted material should be protected for the following reasons:
   - Rule 12 (1) (b) – The information in the *Pickering NGS Units 5 to 8: Updated Safety Report - Parts 1 and 2: Facility Description* has been deemed confidential as this report is considered technical in nature and OPG has not consented to its disclosure.
2. I attest that the above-noted material is not available through any public sources.
3. 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.

4. 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).
5. 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.
6. 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:

- OPG Letter, J. Franke to R. Richardson, "Pickering NGS Units 5 to 8: Updated Safety Report - Parts 1 and 2: Facility Description", October 20, 2022, CD# NK30-CORR-00531-08586.
- Enclosure to NK30-CORR-00531-08586: OPG Report, "Pickering B Safety Report – Part 1", October 7, 2022, CD# NK30-SR-01320-00001 R006.
- Enclosure to NK30-CORR-00531-08586: OPG Report, "Pickering B Safety Report – Part 2", October 7, 2022, CD# NK30-SR-01320-00002 R006.
- Attachment #1 (included below): Summary of NK30-SR-01320-00001 R006 and NK30-SR-01320-00002 R006

Authorized signature:

2024/03/28

Riedewaan Bakardien,
Senior Vice President,
Pickering Nuclear Generating Station

Date

Attachment #1: Summary of NK30-SR-01320-00001 R006 and NK30-SR-01320-00002 R006

NK30-SR-01320-00001 R006

NK30-SR-01320-00001, Part 1, Revision 006 issued on October 07, 2022 is a comprehensive document outlining the design philosophy and site characteristics of the Pickering Nuclear Generating Station 'B' (Units 5 to 8) in Pickering, Ontario. Section 1 of the report is an introduction to the station and a general description of critical systems. It covers comparisons with other Ontario nuclear stations, and derived releases limits. Section 2 is a summary of the siting and environmental data applicable to the Pickering Nuclear site. It covers geography, demography, local land use, geology, seismology, hydrology, and meteorology. The report also has many reference documents, which have been used in its development.

NK30-SR-01320-00002 R006

NK30-SR-01320-00002, Part 2, Revision 006, issued on October 07, 2022, is a detailed description of critical systems at the Pickering Nuclear Generating Station 'B', Units 5 to 8. This document outlines the safety design philosophy of CANDU reactors using industry OPEX and by describing critical design criteria, structures, reactor design, process systems, and special safety systems. It also includes information on instrumentation and control systems, electrical systems, turbines and generators, fuel handling, auxiliary systems, radiation protection, and radioactive waste management.