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Supplementary Information

Written Submission from Ontario Power Generation

In the matter of

Ontario Power Generation

Application to refurbish Pickering Nuclear
Generating Station and to renew licences
for the Pickering Nuclear Generating
Station and Waste Management Facility

**Commission Public Hearing
Part 2**

October 6-9, 2026

Renseignements supplémentaires

Mémoire d'Ontario Power Generation

À l'égard de

Ontario Power Generation

Demande concernant la réfection de la
centrale nucléaire de Pickering et le
renouvellement de ses permis pour la
centrale et l'installation de gestion des
déchets

**Audience publique de la Commission
Partie 2**

6-9 octobre 2026

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

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Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

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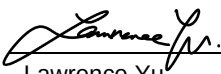
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Prepared by: Staff of Kinectrics Inc.

Reviewed by:  09-MAY-2024
Alex Brittain Boisvert Date
Manager
Projects, Reactor Safety
PNGS Refurbishment

Reviewed by:  09-May-2024
Lawrence Yu Date
Senior Manager
Reactor Safety
PNGS Refurbishment

Reviewed by:  10-MAY-2024
Ricky Aroda Date
Director
Station Engineering
Pickering

Emily Tarle for R. Bakardien
Approved by:  17-May-2024
Riedewaan Bakardien Date
Senior Vice President
Pickering

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 2 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

Table of Contents

	Page
List of Tables and Figures.....	4
Revision Summary.....	5
1.0 INTRODUCTION.....	6
1.1 Purpose.....	6
1.2 Operating Strategy of the Facility	7
1.3 Current Licensing Basis.....	7
2.0 SCOPE OF PICKERING PSR3.....	9
2.1 PSR3 Objectives	9
2.2 General Requirements of PSR3	9
2.3 Previous PSR – D-PSR, PSR2, D-ISR, & PB-ISR Descriptions.....	11
2.4 PSR3 Strategy	12
2.5 Structures, Systems and Components within the Scope of the PSR3 Review	13
2.5.1 Major Component Replacements	13
2.6 PSR3 Assessment Basis.....	13
2.6.1 Safety Factor Review Tasks.....	14
2.6.2 Modern Laws, Regulations, Codes, and Standards Applicable to PSR3.....	15
3.0 PSR3 METHODOLOGY.....	16
3.1 PSR3 Process Overview	16
3.2 Safety Factor Reviews.....	16
3.2.1 Safety Factor Review Process.....	16
3.2.2 Reviews of Modern Laws, Regulations, Codes, and Standards.....	17
3.2.3 Safety Factor Results and Reports.....	19
3.3 Global Assessment	20
3.3.1 Global Assessment Process.....	21
3.3.2 Global Assessment Team	22
3.3.3 Global Assessment Methodology	22
3.3.4 Global Assessment Report.....	26
3.4 Integrated Implementation Plan.....	26
4.0 PSR3 GOVERNANCE	27
5.0 ACRONYMS	28

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 3 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

6.0	REFERENCES.....	30
Appendix A	PSR3 Interface with Previous PSRs.....	32
Appendix B	Structures, Systems and Components within the Scope of PSR3	33
Appendix C	Safety Factor Review Tasks.....	38
Appendix D	Modern Laws, Regulations, Codes, and Standards applicable to PSR3.....	58

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 4 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

List of Tables and Figures

Table 1: PNGS PSR3 Safety Factors.....	14
Table 2: Safety Significance Level and Impact on Nuclear Safety.....	23
Table B1: Pickering NGS 5-8 SOE / SIS Systems	35
Table B2: Pickering NGS 5-8 Civil Structures	37
Table D1: List of LRCSS in the PSR3 Assessment Basis.....	63
 Figure 1: PNGS PSR3 Process Flowchart	 10

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 5 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

Revision Summary

Revision Number	Date	Comments
R000	2024-02-01	Initial issue.
R001	2024-05-08	Revised to incorporate CNSC review comments (P-CORR-00531-23661) and the OPG Protocol with the CNSC (P-CORR-00531-23552, e-Doc 7249270). Revisions made to Sections 1.1, 1.2, 2.3, 2.4, 3.2.1, 4.0, Appendix B Table B2, Appendix C Safety Factor 12, and Appendix D Table D1.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 6 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

1.0 INTRODUCTION

1.1 Purpose

This document outlines the basis for performing the Periodic Safety Review (PSR) for the Pickering Nuclear Generation Station (PNGS). The Pickering PSR (P-PSR3) is a comprehensive PSR and is subsequent to the previous Pickering PSR2 (P-PSR2) performed in support of the evaluation of continued operation of the Pickering NGS units beyond the year 2020. The PSR Basis Document defines the scope and methodology for the conduct of PSR3. It is intended to align with the requirements for a PSR basis document as outlined in Section 3 of REGDOC-2.3.3, Periodic Safety Reviews [1], which was developed to be consistent with the International Atomic Energy Agency (IAEA) Specific Safety Guide, SSG-25, Periodic Safety Reviews for Nuclear Plants [2]. Further guidance is taken from Canadian Standards Association (CSA) N290.18-17, Periodic Safety Review for Nuclear Power Plants [3].

It addresses:

- 1) The current licensing basis for PNGS.
- 2) The proposed operating strategy of the facility.
- 3) The scope of PSR3 to be addressed:
 - a. The Safety Factors to be reviewed and their associated review tasks,
 - b. The Laws, Regulations, Codes, and Standards (LRCS) to be reviewed as established in the Assessment Basis,
 - c. The Structures, Systems and Components list.
- 4) The methodology for conducting the elements of PSR3:
 - a. The strategy for the safety factor reviews,
 - b. The process for categorizing, prioritizing, tracking, and resolving Gaps arising from the Safety Factor reviews,
 - c. Conduct of the Global Assessment,
 - d. The process for developing the Integrated Implementation Plan (IIP).
- 5) The major milestones.
- 6) The project management and quality management processes to be followed in carrying out PSR3.

As per the Pickering Licence Condition Handbook (LCH), Pickering is licensed to operate to August 31, 2028 [4]. PSR3 is being performed to support the PNGS Refurbishment for the life extension of PNGS 058 for an additional 30 years of operating life. As noted, PSR3 is a subsequent PSR, an update building upon the review basis of earlier OPG ISR/PSR work and revisiting the following for applicability to PSR3:

- The Pickering B Integrated Safety Review (PB-ISR), performed in support of refurbishment and continued operation for another 30 years of the Pickering 5-8 units,
- The Darlington ISR (D-ISR), performed in support of refurbishment and continued operation of the Darlington units [5],

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 7 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

- The P-PSR2 performed in support of the evaluation of continued operation of the PNGS units beyond the year 2020 to 2024 [6] and PSR2-B performed in support of continued operation of PNGS units until 2026,
- The Darlington PSR (D-PSR), performed in support of licence renewal (10 years of continued operation) [7].

1.2 Operating Strategy of the Facility

Pickering NGS is located on the shore of Lake Ontario in the City of Pickering, in the Regional Municipality of Durham. The station has 6 operating nuclear reactors and 2 reactors (Units 2 and 3) that have been removed from operations and placed in the safe storage and surveillance state. PNGS was built in two phases. Construction of Units 1 through 4 started in 1966, with Unit 1 becoming operational in 1971 and Unit 4 becoming operational in 1973. The construction of Units 5 through 8 started in 1974 and the units entered operation between 1983 and 1986.

For the purposes of PSR3, the period of operation of PNGS Units 5-8 will be extended for 30 years. PSR3 does not include Pickering 1,4 as the units will be in a safe storage state when Pickering 5-8 are being refurbished. Only Pickering A SSCs required for the continued safe operation of Pickering B are included in the scope of PSR3.

1.3 Current Licensing Basis

The Licensing Basis is defined in the Canadian Nuclear Safety Commission (CNSC) document REGDOC-3.5.3, "Regulatory Fundamentals" [8] as follows:

The licensing basis sets the boundary conditions for a regulated activity, and establishes the basis for the CNSC's compliance program for that regulated activity.

All licensees are required to conduct their activities in accordance with the licensing basis, which is defined as a set of requirements and documents for a regulated activity comprising the following:

- *The regulatory requirements set out in the applicable laws and regulations*
- *The conditions and safety and control measures described in the licence, and the documents directly referenced in that licence*
- *The safety and control measures described in the licence application and the documents needed to support that licence application*

The following demonstrates how this definition of Licensing Basis is established for PNGS:

Item (i): The primary applicable Act and Regulations are the Nuclear Safety and Control Act and the Regulations made under this Act. Other Laws and Regulations applicable to the facility are listed on the CNSC website.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 8 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Item (ii): This item refers to licence conditions and documents directly referenced in the Power Reactor Operating Licence (PROL). The PNGS PROL in effect at the time of the PSR3 documentation freeze date (defined in Section 2.4 of this report) was PROL 48.00/2028 [4]. The CSA and CNSC documents directly referenced in the PROL are listed in the associated revision of the LCH, LCH-PR-48.00/2028-R005 [4], in Appendix D of the LCH.

Item (iii): CNSC document CNSC REGDOC-3.5.3 [8] establishes that documents needed to support the licence application are those documents that demonstrate that the applicant is qualified to carry out the licensed activity, and that appropriate provisions are in place to protect worker and public health and safety, to protect the environment, and to maintain national security and measures required to implement international obligations to which Canada has agreed.

Appendix C of the LCH [4] identifies the OPG governance documents that describe OPG's safety and control measures, taken from OPG's licence application for PNGS. These are the OPG documents that require written notification to the CNSC when revised.

The OPG documents associated with Item (iii) above will be used to support the reviews in PSR3. Additional OPG documents will also be used, as required, to support the reviews.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 9 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

2.0 SCOPE OF PICKERING PSR3

2.1 PSR3 Objectives

CNSC REGDOC-2.3.3 [1] identifies the objectives of a PSR as follows:

The objectives of a PSR are to determine:

- 1. The extent to which the facility conforms to modern codes, standards and practices.*
- 2. The extent to which the licensing basis remains valid for the next licensing period.*
- 3. The adequacy and effectiveness of the programs and the structures, systems and components in place to ensure plant safety until the next PSR or, where appropriate, until the end of commercial operation.*
- 4. The improvements to be implemented to resolve any gaps identified in the review and timelines for their implementation.*

Through the PSR3 work, OPG will assess the design, condition, and operation of PNGS in support of continued safe operation for the period of PSR3. The PSR3 will also determine if there are reasonable and practical enhancements that may be adopted to improve plant safety.

2.2 General Requirements of PSR3

CNSC REGDOC-2.3.3 [1] provides regulatory expectations for the conduct and content of a PSR based on guidance from the IAEA SSG-25, Periodic Safety Review for Nuclear Power Plants [2]. Guidance from CSA N290.18-17 [3] will also be applied in the conduct of PSR3.

In accordance with REGDOC-2.3.3, the PSR3 shall be conducted according to the following four phases:

1. Preparation of a PSR3 Basis Document (this document).
2. Conduct of Safety Factor reviews and identification of Gaps and Strengths.
3. Analysis of the Gaps and identification of potential safety enhancements for PNGS in the Global Assessment process.
4. Preparation of a plan for the implementation of safety enhancements (Integrated Implementation Plan).

The elements of PSR3 are shown in a process flowchart in Figure 1. The steps associated with this process flowchart are further described in Section 3.0 of this document.

Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	10 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

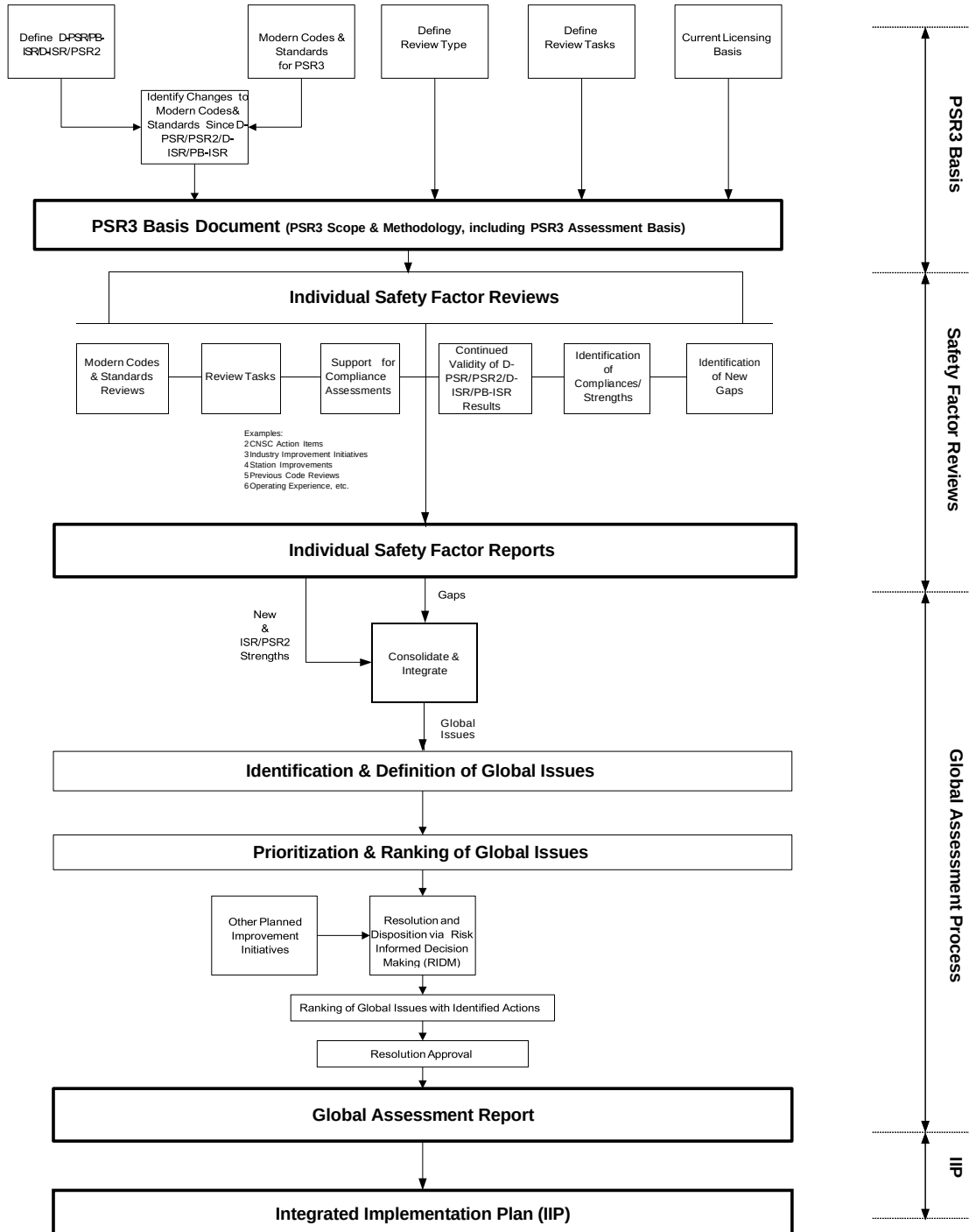


Figure 1: PNGS PSR3 Process Flowchart

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 11 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

2.3 Previous PSR – D-PSR, PSR2, D-ISR, & PB-ISR Descriptions

PSR3 is a comprehensive and subsequent PSR, as defined in CNSC REGDOC-2.3.3 [1] and IAEA SSG-25 [2]. It builds upon on the earlier OPG PSR work and other associated assessments (e.g., PSR2 and PB-ISR at Pickering and the D-PSR and D-ISR). Per REGDOC-2.3.3, Section 2 [1]:

It is expected that the effort necessary to carry out a second (or subsequent) PSR of an NPP will often be considerably less than for the first. In general, subsequent PSRs will focus on changes in requirements, facility conditions, operating experience and new information, rather than repeating the activities of previous reviews.

Further, per REGDOC-2.3.3 Section 3.3 [1], this PSR will explicitly consider if the conclusions of the PSR2 remain valid. This includes revisiting previous assessment gaps dispositioned and gaps that may have been closed based on the planned end-of-commercial operation of Pickering Units 5-8.

These previous reviews and their interfaces with the PSR3 are described below:

Interface between Darlington PSR and Pickering PSR3

The D-PSR Safety Factor reviews were completed in 2021 and the IIP is complete. Much of the compliance assessments and evaluation for Safety Factors for D-PSR were based on programs and practices that are common to both Pickering and Darlington and apply to all of OPG's nuclear operations. As a result, D-PSR programmatic conclusions are applicable to the Pickering PSR3 for situations where Pickering follows the same nuclear programs and practices that were assessed for Darlington. Similarly, D-PSR was built on previous findings from the Pickering PSR2.

Interface between Pickering PSR2 and Pickering PSR3

The most recent Pickering PSR performed by OPG prior to PSR3 was PSR2, which was completed in 2018 [9]. Much of the compliance assessment and evaluation for Safety Factors for PSR2 was based on programs and practices that are common to both Pickering and Darlington and apply to all of OPG's nuclear operations. PSR2 was built on previous findings from the D-ISR and PB-ISR. As an extension to PSR2, Pickering PSR2-B has been completed to support continued operation of Pickering 5-8 until 2026.

Interface between Darlington ISR and Pickering PSR3

The D-ISR was completed in 2013 [5], with code refresh reviews [10] completed in 2014. The D-ISR was performed in support of refurbishment and life extension of the Darlington units. Extensive reviews of Codes and Standards were completed as well as component condition assessments for Structures, Systems and Components (SSCs) within the ISR scope. Many of the findings from this review are being addressed through the Darlington Refurbishment IIP [11], which is due to be completed in 2028. Much of the compliance assessment and

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 12 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

evaluation of the Safety Factors for the D-ISR are based on programs and practices that apply across OPG's nuclear operations. As a result, D-ISR programmatic conclusions were applicable to the Pickering PSR2, and subsequently to the Pickering PSR3, for situations where Pickering is confirmed to follow the same nuclear programs and practices that were assessed for Darlington.

Interface between Pickering PB-ISR and Pickering PSR3

The PB-ISR, which included a comprehensive review of Codes and Standards, was completed in 2009 to support refurbishment and continued operation of the Pickering 5-8 units [12]. At that time, OPG was considering the option of refurbishment of the reactor components in each unit in order to continue to operate the Pickering 5 to 8 reactors for another 30 years. OPG later decided not to proceed with the refurbishment, for economic reasons, and decided to pursue instead the alternative of extended operations of all of the units to the end of 2020 without the replacement of the reactor components. In support of this approach, safety enhancements were identified based on the results of the ISR. These actions were documented in the Continued Operations Plan (COP) and have been completed. As the PB-ISR was performed in support of Pickering 5-8 refurbishment and continued operations for 30 years, PB-ISR conclusions may be applicable to PSR3 in situations where requirements have not changed, and where the Pickering design, condition, and operation is confirmed to be the same.

2.4 PSR3 Strategy

The scope of the PNGS PSR3 is to conduct a Periodic Safety Review of PNGS that meets the requirements of CNSC REGDOC-2.3.3 [1] and IAEA SSG-25 [2], and also incorporates the elements of CSA N290.18 [3].

PSR3 is a comprehensive and subsequent PSR assessing against the most modern codes and standards and building on the review basis of the D-PSR, PSR2, PB-ISR, and D-ISR. CNSC REGDOC-2.3.3 and IAEA SSG-25 identify that subsequent PSRs should focus on changes in requirements, facility conditions, operating experience, and new information, rather than repeating activities conducted in previous reviews. As such, PSR3 will be forward looking, focusing on changes to requirements since the last assessment, confirmation that the condition of PNGS will support continued operation for the extended lifetime of 30 years. PSR3 will be used to identify practicable incremental enhancements to safety above those that have already been implemented or that are already committed.

Appendix A depicts the interface between the elements of PSR2, PB-ISR, D-ISR, and D-PSR, and the process by which PSR2 and PB-ISR findings are to be applied to PSR3. PSR3 will evaluate applicability of the findings from the PSR2, PB-ISR, D-ISR, and D-PSR, for the operation period and will consider the service life of the 30 years post refurbishment. In particular, a PSR2, PB-ISR, D-ISR, and D-PSR gap is applicable to PSR3 if the gap is still open or if resolution of a closed gap could be affected for extended operation. Such gaps will be carried forward into PSR3 for consideration in the Global Assessment.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 13 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

The evaluation of Safety Factors for D-PSR and PSR2 was largely based on programs and practices that apply across OPG's nuclear operations. As a result, D-PSR programmatic conclusions are applicable to the PSR3 for situations where Pickering follows the same nuclear programs and practices that were assessed for Darlington. An effectiveness review (effectiveness at PNGS) of OPG programs used to demonstrate compliance with the PSR3 Assessment Basis will be conducted, using recent audit and self-assessment results.

PSR3 Freeze Date

To establish a basis for conducting the review, and in order to reflect the interrelationships of OPG internal documents and external documents (e.g., LRCSSs), a date of December 31, 2023 was selected as the freeze date for PSR3.

PSR3 Period of Validity

The current licensing basis for Pickering NGS covers operation of Pickering NGS units until the end of August 2028. An update to the PSR is required every 10 years per REGDOC-2.3.3 [1].

2.5 Structures, Systems and Components within the Scope of the PSR3 Review

The scope of PSR3 is applicable to the facilities regulated under the PNGS PROL. The review of the SSCs within the scope of the PSR3 will focus on the Pickering Systems Important to Safety (SIS) and the Safe Operating Envelope (SOE) systems [13]. The list of SSCs within PSR3 scope as well as the systematic process to establish this list are detailed in Appendix B.

Only Pickering A SSCs required for the continued safe operation of Pickering B are included in the scope of PSR3. Facilities regulated under other operating licences (i.e., the Pickering Waste Management Facility) are not considered within the PSR3 scope.

2.5.1 Major Component Replacements

Major components to be replaced during Pickering Refurbishment will include the Fuel Channels, Feeders, and Boilers. Any additional components replacements identified as a result of system condition and safety improvement assessments will be added. In addition, changes required to separate interconnected systems with Pickering 014 station will be made.

2.6 PSR3 Assessment Basis

The PSR3 Assessment Basis consists of:

1. Safety Factor Review Tasks derived from CNSC REGDOC-2.3.3 [1], IAEA SSG-25 [2], and CSA N290.18 [3] taking into consideration the Review Tasks used in the D-PSR, PSR2, D-ISR, and PB-ISR.
2. A set of modern LRCSSs, determined to be applicable for PSR3 review.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 14 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

These components of the PSR3 Assessment Basis are described further below.

2.6.1 Safety Factor Review Tasks

Safety Factors cover all aspects important to the safety of an operating nuclear power plant (NPP). There are 15 Safety Factors used in the PSR3 review; 14 are identified in IAEA SSG-25, grouped into five subject areas, and one additional Safety Factor (Radiation Protection) is identified in CNSC REGDOC-2.3.3. These 15 Safety Factors are shown in Table 1.

Table 1: PNGS PSR3 Safety Factors

Subject Area	Safety Factor	
The Plant	1	Plant Design
	2	Actual Condition of Structures, Systems and Components Important to Safety
	3	Equipment Qualification (environmental and seismic)
	4	Aging
Safety Analysis	5	Deterministic Safety Analysis
	6	Probabilistic Safety Assessment
	7	Hazard Analysis
Performance and Feedback from Operating Experience	8	Safety Performance
	9	Use of Experience from other Plants and Research Findings
Management	10	Organization, the Management System and Safety Culture
	11	Procedures
	12	Human Factors
	13	Emergency Planning
Environment	14	Radiological Impact on the Environment
Radiation Protection	15	Radiation Protection

The PSR3 Safety Factor Objectives and Review Tasks are presented in Appendix C. Review Tasks for Safety Factors 1 to 14 are consistent with the requirements and guidance set out in IAEA SSG-25 [2]. The Review Tasks have been chosen to fulfil the requirements of IAEA SSG-25 while maintaining consistency with Review Tasks used in previous ISRs and PSRs performed for Pickering and Darlington. CNSC REGDOC-2.3.3 encompasses all of the PSR

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 15 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

Safety Factors recommended by the IAEA in SSG-25 and, as noted, introduces Safety Factor 15, Radiation Protection.

Given that the PSR2 and D-PSR Review Tasks had been accepted by the CNSC in the PSR2 Basis Document [6] and the D-PSR Basis Document [7], respectively, and that both IAEA SSG-25 and REGDOC-2.3.3 have not been revised since Pickering PSR2, the Safety Factor (SF) review scope as required by these documents has not changed for PSR3. As a result, to develop the Review Tasks for PSR3, the accepted D-PSR and PSR2 Review Tasks were re-assessed to confirm validity and applicability. The D-PSR and PSR2 Review Tasks were compared to the previous PB-ISR Review Tasks as well as the Section 5 clauses in IAEA SSG-25 and recommendations were made to modify the Review Tasks for PSR3 to be more applicable to PNGS and more representative of a subsequent PSR. Review of CSA N290.18-17, which was not issued prior to the execution of PSR2, identified an additional review task for Safety Factor 5 and this was included as a review task for PSR3 as well.

Per SSG-25, some of the Safety Factor descriptions include discussion of reviews of codes and standards, and effectiveness reviews of programs that are used to demonstrate compliance with Review Tasks or codes and standards. In such cases, these discussions were not translated into specific PSR3 Review Tasks. Codes and standards reviews are performed separately for each Safety Factor, and thus having a stand-alone Review Task for this work would be redundant. Refer to Section 3.2.1 for additional detail.

2.6.2 Modern Laws, Regulations, Codes, and Standards Applicable to PSR3

The process to identify the modern LRCSs that are applicable to the PSR3 Assessment Basis involved first creating a broad list from multiple sources (potential candidate LRCSs) and then screening them to identify those that are applicable to the PSR3 scope. The identification and selection criteria are detailed in Appendix D.

The result of the identification and selection process was a set of modern LRCSs that became part of the PSR3 Assessment Basis. These documents are listed in Table D1 in Appendix D. This table also identifies the modern version / date of the LRCS, the type of review that will be completed in PSR3, and applicable Safety Factors. The types of review to be used in PSR3 is explained in greater detail in Section 3.2.2.

For the purpose of the performance of PSR3, OPG has established a freeze date for modern LRCSs to be December 31, 2023 (refer to Section 2.4).

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 16 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

3.0 PSR3 METHODOLOGY

3.1 PSR3 Process Overview

Figure 1 shows the general process overview of PSR3. The PSR3 Basis Document (this report) defines the PSR3 Assessment Basis, scope, and methodology. Safety Factor reviews are conducted in accordance with the PSR3 Assessment Basis, to identify Gaps and Strengths. The Gaps and Strengths are consolidated into Global Issues in the Global Assessment process. The Global Issues are prioritized, assessed, and ranked, and proposed resolutions (enhancements) undergo review and receive approvals. The Global Assessment outputs are documented in a Global Assessment Report (GAR). The Integrated Implementation Plan documents the proposed enhancements from the GAR and provides implementation timelines for the enhancements. Further details of these steps are provided in the sections that follow.

3.2 Safety Factor Reviews

3.2.1 Safety Factor Review Process

Safety Factor reviews will be conducted for the 15 Safety Factors included in PSR3. Each review will address the associated Review Tasks for the Safety Factors (Appendix C), and will conduct reviews against applicable modern LRCs from the PSR3 Assessment Basis (Appendix D). An effectiveness review of OPG programs used to demonstrate compliance with the PSR3 Assessment Basis will be conducted, using recent audit and self-assessment results. A Safety Factor report will be completed for each of the Safety Factors.

Safety Factor compliance assessments will incorporate information from:

- OPG programs and procedures listed in the LCH, and any other programs and procedures which support the compliance arguments;
- Commitments previously made to the CNSC, open CNSC action items, and exemptions granted by CNSC since the current operating licence was issued (safety significant issues, per the Pickering LCH) to determine if there are any impacts associated with extended Pickering operation;
- Fukushima actions that were closed based on previous end-of-commercial-operation dates;
- Previously identified PSR2, PB-ISR, D-ISR, and D-PSR gaps related to each Safety Factor and the status of OPG's improvement plans or other dispositions to address these;
- Assessments and reviews performed since the PSR2, PB-ISR, D-ISR, and D-PSR documents were completed;
- Audits and other Self Assessments to review the effectiveness of OPG Programs, Policies, and Procedures.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 17 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

As a subsequent PSR, the PSR3 Safety Factor reviews will focus on changes in requirements (LRCSS), updated plant conditions, operating experience, and information from research, rather than repeating the activities of previous reviews. The methodology for performing the Safety Factor reviews will consider the results of the safety assessments and LRCS compliance work previously completed by OPG. It will focus on areas where incremental impacts since previous assessments may exist. This approach is in accordance with the guidance provided by the CNSC in REGDOC-2.3.3.

The following will provide guidance for performing the PSR3 Safety Factor reviews:

- Consider existing D-PSR results to the extent that the programmatic aspects can be confirmed to be applicable to Pickering^{1,2}.
- Focus attention on requirements that are new or that have changed in relation to the requirements that were used as the basis for Pickering PSR2 and PB-ISR, so that their impact on PNGS can be assessed (e.g., new or revised LRCSS, as well as incremental guidance contained in IAEA SSG-25).
- Apply information provided in Pickering PROL renewal applications. Documents specifically listed in the PROL have generally been in place for many years, and therefore have been subject to prior assessments as well as ongoing compliance verification activities by both OPG in the form of audits and self-assessments, and by the CNSC in the form of inspections and audits.

Identification of Interdependencies

The identification of interdependencies is managed through the Safety Factor process. The LRCS list identifies the expected applicable Safety Factors for each LRCS. This list identifies the initial consideration for interdependencies as LRCSS may apply to multiple Safety Factors.

All findings (Gaps and Strengths) that are related to other Safety Factors will be identified and provided to the reviewers of the relevant Safety Factors. Table 1 from Appendix I of IAEA SSG-25 will be referenced for potential interfaces between the different safety factors [2]. Each Safety Factor report will have a section “Interfaces with other Safety Factors” where gaps or strengths that are applicable to other Safety Factors are documented.

3.2.2 Reviews of Modern Laws, Regulations, Codes, and Standards

CNSC REGDOC-2.3.3 requires that the PSR3 PSR Basis document identify:

- The modern LRCSS that will be used in the Safety Factor reviews, including their

¹ Areas where the PSR3 assessment cannot be judged to be applicable or sufficient to address Pickering will be identified.

² Compliance assessments for PSR3 Review Tasks will be prepared independently of D-PSR, PSR2, PB-ISR, and D-ISR findings. That is, compliance for each Review Task will be assessed based on current OPG Governance, Programs, Policies, Procedures, Audits, and Self-Assessments as required, and will only mention the findings of D-PSR, PSR2, PB-ISR, or D-ISR if there are pertinent gaps which need to be addressed within the context of PSR3.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 18 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

effective dates, and the criteria for their selection.

- The PSR documentation freeze date beyond which documentation changes and new information will not be considered.
- The type of review to be performed (Clause-by-Clause, High Level, or Incremental).
- The selection process for the modern LRCS list, which is described in Appendix D of this document and in more detail in [14].

The PSR3 will conduct reviews against a baseline of the previous PSR2 and PB-ISR, and applicable D-PSR and D-ISR work. It is important to note that OPG conducts regular reviews of new and revised Codes and Standards, so a large amount of information is already available to assist in the Safety Factor LRCS reviews. As a result, many of the updated codes and standards issued since the previous D-PSR and PSR2 have already had gap assessments performed, to varying degrees of detail, which will be utilized and cited in the present PSR. Further, compliance with some of the listed codes and standards is a requirement of the PROL and is communicated directly to the regulator.

Since this assessment is a subsequent PSR, its focus will be on identifying safety significant differences between Codes and Standards that were previously assessed and what is now different within the current PSR Assessment Basis. In general, these differences relate to:

- More recent (new or revised) versions than what was previously assessed³
- Safety significant differences between OPG NGSS, if one specific station is the basis for the earlier assessment.

As noted above, CNSC REGDOC-2.3.3 identifies three types of reviews that are appropriate for a PSR: Clause-by-Clause, High Level, and Alternative. For the reasons given above that align with the goals of a subsequent PSR, OPG will include an alternate review type called “Incremental”, as defined below and as per CSA N290.18-17 [3]. Thus, the following three review types will be applied for the PSR3:

- Clause-by-Clause review: Any new LRCSs will be subjected to a clause-by-clause type review. In a clause-by-clause review, conformance with individual clauses is demonstrated by supporting evidence stating whether the requirements stipulated in the requirement document are met;
- High Level review: In a high level review, the degree of conformance with clauses or groups of clauses in the LRCS is demonstrated by supporting evidence stating whether the intent of the requirements stipulated in the requirement document is met; and

³ “New” refers to a LRCS that was not previously considered in the context of earlier assessments. “Revised” refers to an updated version of a LRCS that was previously considered in the context of earlier assessments. Where a document has a new number/type, but has not changed fundamentally, it will be treated as a revision, and not a new, document (e.g., in cases where a REGDOC has replaced a CNSC G or RD document, but the content is generally the same). In these cases, a confirmatory review will be done.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 19 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

- Incremental review: For LRCSS that have been reviewed in a previous ISR or PSR but have been subsequently revised. For these cases, the incremental review will focus on the changes made in the revision(s) since the previous review and will be performed in the method applied previously (i.e., Clause-by-Clause or High Level). If a previous review of a document was completed as part of the D-PSR and is applicable to both sites (e.g., programmatic), the previous assessment will be reviewed to assess compliance for PNGS.

The reviews of LRCSS in the PSR Assessment Basis will be grouped and documented in reports to support the Safety Factor reviews and Global Assessment.

Table D1 in Appendix D provides the set of modern LRCS established as the Assessment Basis for the PSR3. This table also includes the type of review to be applied and the expected applicable Safety Factors for each. As the Safety Factor review process progresses, the applicable Safety Factors may change due to the identification of interdependencies or the emergence of other information. The final interdependencies will be captured in the applicable Safety Factor reports.

3.2.3 Safety Factor Results and Reports

The Safety Factor reviews will identify Compliances, Strengths, and Gaps with respect to the review elements in the PSR3 Assessment Basis. These are further defined as:

Compliance:

- For Clause-by-Clause reviews of modern LRCSS, Compliance indicates that the safety requirement is met.
- Where a High Level review has been performed, Compliance indicates that the intent of the safety requirement is met.
- Where an Incremental review has been performed, Compliance indicates that the change in the safety requirement, per the topical review, is met.
- For reviews of Safety Factor Review Tasks, Compliance indicates that either the safety requirement or the intent of the Review Task is met.

Indirect Compliance:

- For Non-PROL Codes and Standards, the review demonstrates that either the safety requirement or the intent of a clause (or set of clauses) is met.
- For a Review Task the assessment of the Review Task demonstrates that it meets either the safety requirement or the intent of the Review Task.

Strength:

- For Clause-by-Clause reviews of modern LRCSS, Strength indicates that the safety requirement is exceeded.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 20 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

- Where a High Level review has been performed, Strength indicates that the intent of the safety requirement is exceeded.
- Where an Incremental review has been performed, Strength indicates that the change in the safety requirement, per the topical review, is exceeded.
- For reviews of Safety Factor Review Tasks, Strength indicates that either the safety requirement or the intent of the Review Task is exceeded.

Gap:

- For Clause-by-Clause reviews of modern LRCs, a Gap indicates that the safety requirement is not met.
- Where a High Level review has been performed, a Gap indicates that the intent of the safety requirement is not met.
- Where an Incremental review has been performed, a Gap indicates that the change in the safety requirement, per the topical review, is not met.
- For reviews of Safety Factor Review Tasks, a Gap indicates that the intent of the Review Task is not met.

Compliances that are equivalent to, or surpass PSR3 Assessment Basis requirements or practices, will be forwarded into the Global Assessment process for consideration as Strengths. Gaps will be evaluated by the Global Assessment methodology to identify Global Issues and, with justification, Acceptable Deviations.

The results of the Safety Factor reviews will be documented in Safety Factor reports, which will be submitted to CNSC staff for review. These reports will include:

- The scope of the review,
- Applicable elements of the PSR3 Assessment Basis (Review Tasks and applicable LRCs),
- Review methodology,
- Assessment of compliance with Review Tasks,
- Effectiveness review of OPG programs supporting compliance assessments,
- Review findings (Strengths and Gaps),
- Impacts on other Safety Factor reviews (interdependencies),
- Overall assessment of the Safety Factor.

3.3 Global Assessment

The objective of the Global Assessment is to provide an overall assessment of the safety of the plant, and to assess the acceptability of PNGS for continued operation over the periods of

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 21 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

PSR3, including an assessment of the defence-in-depth capability. The Global Assessment takes into account the Gaps identified during the Safety Factor reviews, as well as any findings from CNSC staff reviews of the Safety Factor Reports. The Global Assessment includes consideration of the five levels of defence-in-depth, including consideration of Strengths, the enhancements proposed⁴ through Global Issue Resolution Plans, and assessment of the aggregate impact of Acceptable Deviations, in order to make a conclusion on the overall acceptability of operation of the plant over the period considered for PSR3.

Preparation of the Integrated Implementation Plan involves transforming the proposed Resolution Plans resulting from the Global Assessment into actions with corresponding schedules for implementation during the next licensing period.

The major inputs to the Global Assessment are the Safety Factor Reports, which include the inputs as described in Section 3.2.1.

3.3.1 Global Assessment Process

The above information is used to commence the Global Assessment process and consists of the following elements:

1. Identification of Gaps and Strengths from the Safety Factor Reports, and any additional Gaps or Strengths that may be identified during the Global Assessment. Input arising from CNSC feedback on the PSR3 deliverables is also considered.
2. Development of Global Issues by integrating and consolidating the Gaps. Each Global Issue is comprised of Gaps related to a common theme.
3. Assessment of interfaces between the various Safety Factors, as well as the aggregate impact of Global Issues.
4. Prioritization of Global Issues and Gaps.
5. Development of proposed Resolution Plans with consideration of safety benefit, practicability, and the interfaces between the Gaps and Global Issues.
6. Assessment of Defence-in-Depth, including Strengths, and the aggregate impact of Acceptable Deviations.
7. Ranking of Global Issues with Resolution Statements.
8. Third Party/Expert Panel and OPG senior management review of proposed Resolution Statements.
9. Assessment of overall acceptability of operation of the plant over the period considered in PSR3.
10. Preparation of the GAR to summarize the assessments and document the Global Assessment.

⁴ At the Global Assessment stage, enhancements are identified as “proposed” through Global Issue resolutions. Enhancements are finalized at the Integrated Implementation Plan stage of the PSR process.

Report	OPG Proprietary		
	Document Number: P-REP-03680-00054		Usage Classification: N/A
	Sheet Number: N/A	Revision Number: R001	Page: 22 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document			

These elements are described further in Section 3.3.3.

3.3.2 Global Assessment Team

The Global Assessment will be conducted by an interdisciplinary team, with appropriate expertise in operations, design, and safety at the plant, including both participants from the Safety Factor reviews, and members who are independent from the Safety Factor review teams, consistent with the requirements in [2].

3.3.3 Global Assessment Methodology

Details of the elements involved in executing the Global Assessment are provided below.

a) *Identification and Consolidation of Strengths and Gaps from the Safety Factor Reports:*

The Strengths and Gaps from the 15 individual Safety Factor Reports will be consolidated and grouped by topic area to support the Global Assessment.

b) *Development of Global Issues:*

The consolidation of Gaps into Global Issues will provide a means to assemble Gaps of a common nature, facilitating the assessment of safety impact and identifying and assessing practical and effective resolutions. The Global Issues will be tabularized, tracking sources of the issues, to facilitate further review and assessment.

c) *Assessment of the interfaces Between the Various Safety Factors, Aggregate Impact of Global Issues:*

With the assembly of Global Issues and Strengths, the aggregate impact of the Global Issues will be assessed. In this way, the interaction between issues will be identified. New Global Issues may be identified as part of this consolidation review. This will support the prioritization and ranking of Global Issues as described below.

d) *Prioritization of Global Issues and Gaps*

PSR3 Global Issues and associated gaps will be prioritized with respect to their importance to Nuclear Safety to determine the Safety Significance level associated with each Global Issue. This will support the resolution evaluation method and the outcome of the resolution process. This methodology is consistent with OPG prioritization processes used in previous ISRs, PSRs, and industry practice.

The Safety Significance level will consider deterministic and probabilistic safety analysis impact, as appropriate. Probability levels selected for delineation between categories are based on significance, as applied in previous ISRs and PSRs. These values account for overall safety impact and align, where appropriate, with requirements and limits in relevant safety standards.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 23 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

The relationship between Safety Significance Level and Impact on Nuclear Safety is shown in Table 2 below:

Table 2: Safety Significance Level and Impact on Nuclear Safety

Safety Significance Level	Impact on Nuclear Safety
1	High
2	Medium
3	Low
4	Very Low

e) *Development of Resolutions / Dispositions of Global Issues (and Gaps)*

Resolution options will be developed and assessed using risk informed decision making techniques. The development of the resolution will utilize the following strategy:

- In assessing potential dispositions, defence-in-depth [15] elements will be considered.
- In developing the resolutions, consideration of overall safety significance will guide the resolution process.
- For Global Issue resolution – the process will be:
 - a. Evaluate the Global Issue to understand the safety basis, and intent of requirement.
 - b. Consider possible options for resolution/mitigation. Consider safety significance and defence-in-depth elements.
 - c. Evaluate options with respect to effectiveness, cost, schedule, and practicality. For potential plant modifications, this may require an evaluation of the safety impact, both deterministic and probabilistic. If it is not practicable to fully resolve a Global Issue, other mitigation options will be considered for enhancements.
 - d. Practicality of a proposed resolution will be evaluated in terms of cost, resources, schedule, and considered in relation to the overall safety impact.
 - e. Propose recommended resolution/mitigation.
 - f. Document the decision making process.
- Items of High or Medium Impact on Nuclear Safety (Safety Significance levels 1 and 2) will require more in-depth analysis to fully understand the issue and

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 24 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

potential impact, and to develop the proposed recommended resolution/mitigation. This may require deterministic and/or probabilistic assessments to measure nuclear safety impact of modifications and more detailed evaluation of the cost/practicality of proposed resolutions. Insights from available site Probabilistic Safety Analyses may be used in evaluating the benefit/practicality of potential options. This will be done concurrent with the development of the IIP.

- Items of Very Low Impact on Nuclear Safety (Safety Significance level 4) will generally be deemed as Acceptable Deviations within the context of PSR3 (with the rationale provided), and while these items will not be tracked beyond the Global Assessment, they will be shared with the accountable organizations for consideration as potential enhancement initiatives for their future work program planning purposes. This will allow the organizations to prioritize the initiatives as part of their integrated programs to ensure the focus is on the right overall priorities. A similar treatment will be applied for items of Low Impact on Nuclear Safety (Safety Significance level 3) for which a practicable solution is not readily evident.
- Proposed resolutions will be categorized as i) Programmatic (changes to procedures and programs), ii) Engineering (plant modifications or maintenance), or iii) Analytical (e.g., safety analysis, hazard analysis), to facilitate binning of potential work. In some cases, the proposed resolutions may entail work from more than one of these categories.
- In some cases, the development of resolutions/dispositions to the Global Issues will be part of an OPG or industry initiative underway or planned. Alternatively, the resolution and development of options may require more detailed analysis and assessment, extending beyond the timelines for submission of PSR3. In these instances, the status of the initiative and plans will be included in the disposition. The work will be included in the Global Assessment to facilitate continued tracking.
- The results of previous Global Assessments for OPG stations will be considered in the review.
- If in the assessment it is determined that a Global Issue / Gap has been closed, due to work done in the interim or for other reasons, the rationale will be documented, and the Global Issue / Gap will be set to Resolved and Closed.
- At the recommendation of the senior management team, an alternate process / resolution may be utilized for a particular Global Issue / Gap.

f) *Assessment of Defence-in-Depth and Aggregate Impact of Residual Global Issues / Acceptable Deviations:*

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 25 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

An important element of the development of proposed recommendations will be to assess the overall defence-in-depth and aggregate impact of the residual Global Issues / Acceptable Deviations. After evaluating a range of resolutions for Global Issues, and determining a recommended resolution to be selected, the impact on defence-in-depth [15], considering both deterministic and probabilistic elements, will be evaluated to assess the aggregate impact on overall safety. It may be necessary to refine the proposed resolutions based on the results of this review. This overall assessment will be an important element in supporting the enhancement plans and the planned operational strategy over the period of PSR3.

g) Ranking of Global Issues

All Global Issues whose resolution involves identified actions will be ranked from 1 through N, where N is the total number, in accordance with overall safety significance. The ranking process will consider factors such as the priority previously determined (Safety Significance Level), the contribution to defence-in-depth, the source of the issue and the degree of non-compliance with the PSR3 Assessment Basis. The ranking process will also account for the extent of impact on multiple safety factors or areas.

The Global Issue resolution actions and ranking will be confirmed by OPG subject matter experts.

h) Third Party/Expert Panel and OPG senior management review of proposed Resolution Statements:

The results of the Global Assessment will be reviewed by a panel of industry experts independent of the Global Assessment Team. The enhancements identified in the PSR3 GAR, with their priority and safety basis, will then be presented to OPG Senior Management for approval. This review will ensure alignment with the resolutions proposed, their basis and context, and will be the means to obtain concurrence that the proposed enhancements are practicable and effective. This will also allow the senior management team to consider potential realignment of overall priorities based on the insights from PSR3. Consistent with OPG Project Management processes, additional approval gates will be required as the resolution development continues towards full implementation.

i) Assessment of Overall Acceptability of Operation of the Plant Over the Period Considered in PSR3:

As a final step in the assessment process, the team will assess the overall acceptability of operation of the plant over the period considered in PSR3. This will entail a review of the results of the Safety Factor Reviews, a consideration of enhancements planned (both newly identified in PSR3 and from other station plans), and a consideration of plant performance and initiatives underway.

j) Preparation of the Global Assessment Report

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 26 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

Preparation of the GAR will be conducted to summarize the assessments and document the Global Assessment, as detailed in Section 3.3.4 below.

3.3.4 Global Assessment Report

The results of the Global Assessment will be documented in a GAR, presenting the results, assessing the overall defence-in-depth of the plant, and documenting the conclusions, corrective actions, and enhancements to be considered. The GAR will include a ranked list of those Global Issues with identified actions, with rationale for the ranking using an established Decision Support Methodology. This will be done concurrent with the development of the Integrated Implementation Plan.

Residual Global Issues and Acceptable Deviations will be noted in the report, summarizing the assessed aggregate impact on safe operations. These items will be conveyed to the accountable organizations for their consideration as potential enhancement initiatives for their future work program planning purposes. These initiatives will be weighed against other important program and plant modifications as part of the base and project work within these organizations. These items will not be tracked further beyond the GAR or carried forward into the Integrated Implementation Plan.

The GAR will include a statement of OPG's assessment of the overall acceptability of operation of the plant. Reviews and approval of the report will be conducted as required under the OPG Management System. The GAR will be submitted to CNSC staff for review.

3.4 Integrated Implementation Plan

The proposed enhancements resulting from the GAR will be documented in the IIP. The IIP will provide the proposed timeline for the implementation of the safety enhancements. It will include a tabularized listing of the safety enhancement initiatives, and their planned schedule. The enhancements summarized in the IIP will be mapped to the CNSC Safety and Control Areas (SCAs) (as per Appendix B of CNSC REGDOC-2.3.3 so as to facilitate CNSC review).

The past PSR2 IIP [16] documented the scope and schedule for the implementation of safety improvements which had been identified through the PSR2. PNGS has also established regular reporting on the progress of IIP commitments to the CNSC. Furthermore, OPG has developed instructions for managing non-intent changes to the IIP including completion and closeout documentation requirements. This process was developed in the Pickering PSR2 IIP Plan Administration document, P-INS-03680-00001 [17].

The IIP for PSR3 will be developed based on the lessons learned from the completion and close out of previous IIP items in the OPG fleet, to ensure alignment with already established procedures, and to benefit from the process efficiency improvements.

The IIP will be submitted to the CNSC for acceptance as per the requirements of REGDOC-2.3.3 [1].

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 27 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

4.0 PSR3 GOVERNANCE

Quality Management

PSR3 work will be conducted under OPG's quality management program (compliant with CSA N286-12). Where external contractors are engaged in performing portions of PSR3, they will either work under OPG's quality program, or under a quality program that has been accepted by OPG as meeting the requirements of N286-12. It will be confirmed that these contractors have appropriate prior experience in order to conduct the OPG PSR3 work. This approach is consistent with OPG's management system.

Teams of reviewers composed of qualified and experienced technical staff in the subject matter of the reviews have been established or will be established for future tasks. The results of the reviews will be evaluated by OPG to confirm that the reviewers had access to all of the information relevant to the performance of the review. OPG will accept the contracted deliverables for OPG use.

Project Management

OPG project management principles will be applied in performing PSR3. The key dates for completion, and submission to the CNSC, of PSR3 deliverables were established via a protocol between OPG and the CNSC [18] and are as follows:

Date	Item
June 05, 2024	PSR3 Basis Document accepted by CNSC
May 19, 2025	Safety Factor Reviews Submitted by OPG
November 19, 2025	Safety Factor Reviews Reviewed by CNSC
August 08, 2026	Global Assessment Report Submitted by OPG
January 08, 2027	Global Assessment Report Reviewed by CNSC
August 31, 2027	Integrated Implementation Plan Submitted by OPG
November 30, 2027	Integrated Implementation Plan Reviewed and Accepted by CNSC

Project Communications

Effective communications practices will be used throughout the performance of PSR3. Interfaces between the work of different external contractors and OPG staff will be managed to ensure consistency between related deliverables and accuracy of information.

Urgent issues will be communicated to the project manager promptly, including any emerging observations that suggest a non-compliance with the current PROL.

Communication with the CNSC will take place through Pickering Regulatory Affairs, consistent with established OPG processes and OPG-CNSC Protocol.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 28 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

5.0 ACRONYMS

ALARA	As Low As Reasonably Achievable
BDBA	Beyond Design Basis Accident
BPVC	Boiler and Pressure Vessel Code
CANDU	Canada Deuterium Uranium
CBC	Clause-by-Clause
CEOF	Central Emergency Operations Facility
CNSC	Canadian Nuclear Safety Commission
COP	Continued Operations Plan
CSA	Canadian Standards Association
DBA	Design Basis Accident
D-ISR	Darlington Integrated Safety Review
D-PSR	Darlington Periodic Safety Review
DSA	Deterministic Safety Analysis
ECI	Emergency Coolant Injection
EME	Emergency Mitigating Equipment
EOC	Emergency Operations Center
EPRI	Electric Power Research Institute
EPS	Emergency Power System
EWS	Emergency Water Supply
FAD	Filtered Air Discharge
FADS	Filtered Air Discharge System
GAR	Global Assessment Report
HL	High Level
HPECI	High Pressure Emergency Coolant Injection
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
IC	Incremental
IFB	Irradiated Fuel Bay
IIP	Integrated Implementation Plan
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
LRCS	Laws, Regulations, Codes, and Standards
NGS	Nuclear Generating Station
NPC	Negative Pressure Containment
NPP	Nuclear Power Plant
OPG	Ontario Power Generation
OSR	Operational Safety Requirements
PAR	Passive Autocatalytic Recombiner
PARMS	Post-Accident Radiological Monitoring System
PB-ISR	Pickering B Integrated Safety Review
PNERP	Provincial Nuclear Emergency Response Plan
PNGS	Pickering Nuclear Generating Station
PROL	Power Reactor Operating Licence
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 29 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

P-PSR2	Pickering Periodic Safety Review 2
P-PSR3	Pickering Periodic Safety Review 3
REGDOC	CNSC Regulatory Document
RIDM	Risk Informed Decision Making
SCA	Safety and Control Area
SF	Safety Factor
SG	Steam Generator
SIS	Systems Important to Safety
SMC	Site Management Center
SOE	Safe Operating Envelope
SSC	Structures, Systems and Components
SSG	Specific Safety Guide
UECC	Unit Emergency Control Centre
UPS	Uninterrupted Power Supply
USI	Universal System Index
VB	Vacuum Building

Report	OPG Proprietary		
	Document Number: P-REP-03680-00054		Usage Classification: N/A
	Sheet Number: N/A	Revision Number: R001	Page: 30 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document			

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Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 31 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

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Report

OPG Proprietary

Document Number:

P-REP-03680-00054

Usage Classification:

N/A

Sheet Number:

N/A

Revision Number:

R001

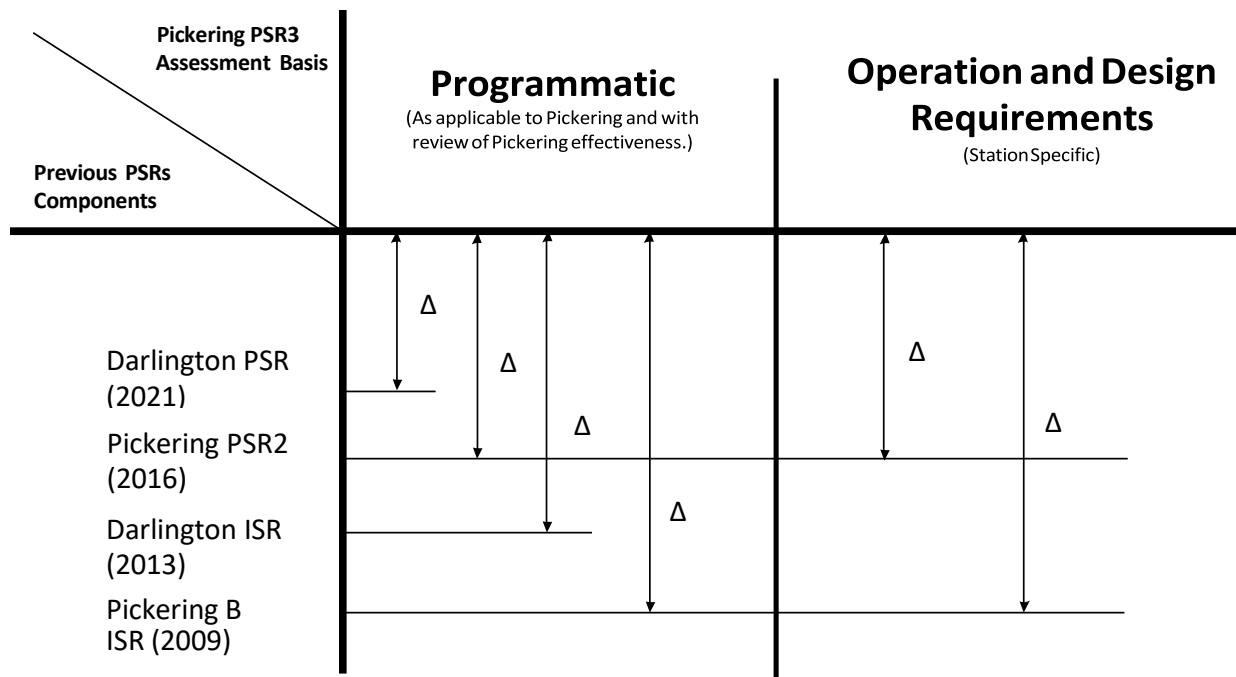
Page:

32 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Appendix A PSR3 Interface with Previous PSRs



The diagram is a schematic depiction of the primary interfaces between the elements of the D-PSR, PSR2, PB-ISR, and D-ISR with the PSR3, and the process by which the D-PSR, PSR2, and D-ISR findings are to be applied to PSR3. The PSR3 is a subsequent PSR and builds on the assessments of the D-PSR, PSR2, PB-ISR, and D-ISR to the extent that they are applicable. The focus is on changes between the PSR2 and PB-ISR and the PSR3 Assessment Bases and applicable results.

Programmatic aspects of the D-PSR, PSR2, PB-ISR, and D-ISR may be (or remain) applicable to PNGS and will be updated to reflect the latest OPG Governance where required (with a review of the effectiveness at Pickering for the related OPG Programs). Requirements related to station-specific aspects of the PSR3 Assessment Basis will draw on aspects of the PSR2 and PB-ISR. In addition, some station-specific gaps from the D-PSR and D-ISR may be applied to PNGS if it is determined that they also apply to PNGS.

Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	33 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

Appendix B Structures, Systems and Components within the Scope of PSR3

The scope of the SSCs within the PSR3 review will encompass those SSCs required for the continued safe operation of Pickering 058, based on Pickering Safety Related Systems [19] with a focus on Pickering SIS [20] and SOE Systems [21]. Only Pickering A SSCs required for the continued safe operation of Pickering B will be included in the scope of PSR3.

OPG defines Safety Related Systems as those systems, and the components and structures thereof, which, by virtue of failure to perform in accordance with the design intent, have the potential to impact on the radiological safety of the public or plant personnel from operation of the NPP [19]. Safety Related Systems are associated with the provision of safety related functions including regulation (and shutdown and start up) (control), and cooling of the reactor core (cool), and limiting the release of radioactive material and exposure of plant personnel and/or the public during normal operation and accident conditions (contain).

SIS are defined following OPG governance [22], considering the risk importance of systems and utilizing expert review panels to select these systems. The identification of SIS is consistent with the requirements of CNSC REGDOC-2.6.1 [23].

SOE systems are identified per OPG governance [13], and include systems, and their associated critical components and structures, for which operational safety requirements are specified to conform with the Safety Report and hence, the Safety Analysis. OPG has prepared formal Operational Safety Requirements (OSR) documents in support of the SOE systems. The SOE systems and associated OSRs are listed in OPG governance [21].

The Pickering NGS 5-8 SOE/SIS systems are listed in Table B1.

Critical Structures (e.g., Reactor Buildings, Pressure Relief Duct, Vacuum Building) will be considered in the review. The civil structures are listed in Table B2.

Additional systems may be considered in the reviews conducted as part of PSR3 if required to demonstrate compliance with specific modern codes and standards.

B.1.0 SSCS IMPORTANT TO EMERGENCY RESPONSE ANDbdba ACCIDENT RESPONSE

In addition, to those SSCs identified above PNGS has several SSCs that are credited for Beyond Design Basis Accident (bdba) accident management. These SSCs are included in PSR3 scope because they are credited with managing residual safety risk and several SF Review Tasks identify bdba and accident management requirements. The requirements for the related SSCs are provided in [24] and [25], while [26] identifies additional SSCs and facilities credited for accident management.

The control rooms (i.e., the Main Control Room and Unit Emergency Control Centres (UECCs) and the Emergency Power System (EPS)/Emergency Water Supply (EWS) Building,

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 34 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

which are all important to accident management, and the structures in the PSR3 scope are in Table B2.

The additional facilities [25] credited with accident management, including BDBA and emergency response are:

- Emergency Operations Center (EOC)
- Site Management Center (SMC)
- Chemistry Lab
- Emergency Mitigating Equipment (EME) storage buildings
- Central Emergency Operations Facility (CEOF)

The SSCs that are only credited for BDBA accident mitigation are identified below based on [26]. A complete list of SSCs is provided in Appendix C of [26] and is not repeated in totality below. In some cases, SSCs credited for Design Basis Accident (DBA) mitigation can also be used or recovered and used for BDBA mitigation. The BDBA SSCs are:

a) EME and Tie-ins to the existing systems. These include:

- Portable power supplies for Group 2 indication and control
- Diesel generators to energize and supply EPS loads.
- Large EME diesel generators to power EPS loads and large EPS supplied motors.
- Diesel powered EME pumps.
- EME Tie-ins to the: Screenhouse, EPS, Feedwater, EWS (Steam Generators (SGs), Heat Transport System (HTS), Moderator), End Shield Cooling, Hydrogen Igniters, and the Irradiated Fuel Bays (IFBs).

b) Passive Autocatalytic Recombiners (PARs) - 34620

c) Post-Accident Radiological Monitoring System (PARMS)

d) Radiological Source Term Monitoring

e) Flood Barriers (various System Classification Indices (USI))

These SSCs and facilities will be considered when assessing Codes and Standards and SF Review Tasks relating to BDBA accident mitigation and emergency response.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 35 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Table B1: Pickering NGS 5-8 SOE / SIS Systems

#	SOE/SIS Systems Required for Pickering 5-8 Operation*	SOE**	SIS***
1	Emergency Coolant Injection System	✓	✓
2	Shutdown System One	✓	✓
3	Shutdown System Two	✓	✓
4	Negative Pressure Containment Systems (e.g., Reactor Buildings, Pressure Relief Duct and Vacuum Building)	✓	✓
5	Powerhouse Emergency Venting System	✓	✓
6	Reactor Regulating System	✓	
7	Service Water Systems	✓	✓ Limited to Class III Low and High Pressure Service Water
8	Moderator System	✓	-
9	Group 1 Electrical Power System	✓	✓ (Limited to Standby Class III Power / Class II Power and also includes Class II Uninterrupted Power Supply (UPS) Room Ventilation)
10	Emergency Water Supply System	✓	✓ (Limited to Emergency Water Supply to Boilers, Heat Transport, Moderator and ACUs)
11	Heat Transport System+	✓	-
12	Shutdown Cooling System	✓	✓
13	Boiler Emergency Cooling System	✓	-

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 36 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

#	SOE/SIS Systems Required for Pickering 5-8 Operation*	SOE**	SIS***
14	Feedwater System	✓	✓ (Limited to Auxiliary Boiler Feedwater and Auxiliary Condensate Systems)
15	Main Steam Supply System	✓	-
16	Fuel and Reactor Physics	✓	-
17	Emergency Power Supply	✓	✓
18	Fuel Handling & Irradiated Fuel Bays	✓	-
19	High Pressure Emergency Coolant Injection (HPECI) Power Supplies	✓	-
20	Annulus Gas System	✓	-
21	Critical Safety Parameter Monitoring Instrumentation	✓	-
22	Shield Cooling System	✓	-

* Includes required critical components and structures

**Specific Universal System Indices (USIs) are provided in associated system OSRs listed in [21] and [19]. Also includes elements of specified support systems (e.g., Instrument Air) where required for credited design basis functions. (This is generally reflected in criticality coding.)

***Specific USIs and required functional elements are detailed in [19].

+Fuel Channels and Feeders will also be included in this review.

The civil structures listed in Table B2 were derived from Table 1 in the PNGS Passive SSC Assessment Non-Containment Civil Structures Screening Report [27]. This screening report was also used in the development of the Pickering Safety Related Structures Periodic Inspection Program [28].

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 37 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Table B2: Pickering NGS 5-8 Civil Structures

#	Pickering 5-8 Civil Structures
1	Turbine Hall
2	Turbine Auxiliary Building
3	Reactor Auxiliary Building
4	Irradiated Fuel Bay (IFB) ⁵
5	SG Oil Pumphouse
6	HPECI System Pumphouse, including Emergency Coolant Injection (ECI) Riser
7	HPECI Auxiliary Building
8	058 Standby Generator Building
9	ECI Recovery Riser
10	Emergency Water & Power Building
11	Off-Gas Building
12	Filtered Air Discharge System (FADS) Building & Filtered Air Discharge (FAD) Stack Enclosure (Including Purge Valve Enclosure)
13	Vacuum Building (VB) Basement
14	Service Wing & Extension
15	Screenhouses
16	Auxiliary Power Supply Buildings
17	Negative Pressure Containment (NPC) Pressure Relief Manifold Enclosure
18	Yard Structures

⁵ IFB B and the Auxiliary IFB are in PSR3 scope. IFB A is not in scope because it is not required for the continued operation of Pickering 058. The AIFB is only included in-scope to the extent that it is used for Pickering 058 Cobalt processing, i.e., not for Pickering 1,4 used fuel storage.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 38 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

Appendix C Safety Factor Review Tasks

Objectives and Review Tasks for the 15 Safety Factor reviews are shown in the tables that follow.

As a subsequent PSR, Review Tasks for PSR3 have been developed from work in the Pickering PSR2 and subsequently in the D-PSR. Review Tasks for Safety Factors 1 to 14 are developed from the requirements, guidance and methodologies set out in IAEA SSG-25 [2] and CSA N290.18 [3]. The Review Tasks have been developed to fulfil the requirements of IAEA SSG-25 and REGDOC-2.3.3 while maximizing consistency with Review Tasks applied in D-PSR and PSR2. CNSC REGDOC-2.3.3 encompasses all of the PSR Safety Factors recommended by the IAEA in SSG-25 and expands upon it by adding Safety Factor 15, Radiation Protection. As a result, PSR3 Review Tasks for Safety Factor 15 are derived from REGDOC-2.3.3 Appendix A.

Additional detail regarding the derivation and application of Review Tasks is provided in Sections 2.6.1 and 3.2.1 as well as [29] and [30]. The Intent and Objectives statements of Safety Factors 1 – 14 are as referenced in IAEA SSG-25 [2]. CNSC REGDOC-2.3.3 [1] does not provide an intent statement for Safety Factor 15.

Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	39 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 1 – Plant Design

Intent	Plant SSCs important to safety should be appropriately designed and configured in such a way that there is a high degree of confidence that they will meet the requirements for safe operation and for performance in compliance with design characteristics, including the prevention and mitigation of events that could jeopardize safety (i.e., fulfilment of their safety functions). Adequate design information, including information on the design basis, should be made available to provide for the safe operation and maintenance of the plant and to facilitate plant modifications.
Objective	Objective of the review of the design of the NPP is to determine the adequacy of the design and its documentation in an assessment against current national and international standards and practices.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Confirm adequacy of the detailed description of the plant design, documenting the Design Basis, supported by layout, systems, and equipment drawings. (2) Assess the adequacy of design documentation. (3) Confirm the adequacy of the SSCs important to safety (list of these systems identified in Appendix B of this document) (4) Perform a review of the application of defence in depth. This includes an examination of: - The degree of independence of the levels of defence in depth; - The adequacy of delivery of preventive and mitigatory safety functions; - Redundancy, separation, and diversity of SSCs important to safety; - Defence in depth in the design of structures (for example: • The degree of independence of the levels of defence in depth; • The adequacy of delivery of preventive and mitigatory safety functions; • Redundancy, separation, and diversity of SSCs important to safety; • Review of the integrity of fuel, cooling circuit and containment building). (5) Perform a review to assess the adequacy of the arrangement for providing radiological protection since the last relevant PSR/ISR assessment. (6) Where the plant has undergone a significant number of modifications over its lifetime, examine the cumulative effects of the modifications on the design. (7) Confirm the adequacy of the processes to ensure that the plant SSCs are compliant with design specifications and consistent with the design documentation.

Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	40 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 2 – Actual Condition of SSCs Important to Safety

Intent	The actual condition of SSCs important to safety within the NPP is an important factor in any review of the safety of the plant design. Hence, it is important to document thoroughly the condition of each SSC important to safety. Additionally, knowledge of any existing or anticipated obsolescence of plant systems and equipment should be considered part of this safety factor.
Objective	The objective of the review is to determine the actual condition of SSCs important to safety and whether it is adequate for them to meet their design requirements. In addition, the review should confirm that the condition of SSCs is properly documented.
Review Tasks	The following Review Tasks are intended to support the Objectives above: (1) Assess and document present conditions of the SSCs important to safety and confirm appropriate measures to address any significant existing or anticipated aging degradation. Any major difference between operating units with respect to aging degradation mechanisms, present condition, or recommended actions shall also be presented. (2) Confirm resources and facilities (on and off site) are available for ongoing plant maintenance. (3) After determining the actual condition of the SSCs important to safety, each of these SSCs will be assessed against the current design basis to confirm that design basis assumptions have not been significantly challenged and will remain that way throughout the period of PSR3. (4) Review the condition and operation of spent fuel storage facilities and their effect on the spent fuel storage strategy for PNGS. (5) Assess dependence on obsolescent equipment for which no direct substitute is available. (6) Assess dependence on essential services and/or supplies external to the plant.

Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	41 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 3 – Equipment Qualification (Environmental and Seismic)

Intent	Plant equipment important to safety should be properly qualified to ensure its capability to perform its safety functions under all relevant operational states and accident conditions, including those arising from internal and external events and accidents (such as loss of coolant accidents, high energy line breaks and seismic events or other vibration conditions). The qualification should adopt a graded approach consistent with the safety classification of the SSC and should be an ongoing activity.
Objective	Objective of the review is to determine whether equipment important to safety is qualified to perform its designated safety function throughout its installed service life.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Confirm the adequacy and effectiveness of the suite of Engineering programs or processes to ensure equipment qualification requirements are met and documented. (2) Confirm equipment qualification has been adequately established for all service conditions expected during normal operation, anticipated operational occurrences and accident conditions. These service conditions are subdivided into environmental conditions and operational conditions. Environmental conditions include ambient temperature, pressure, humidity/steam, radiation, water/chemical sprays, fluid submergence, fire, and seismic vibration. Operational conditions include process related conditions such as vibration, load cycling, electrical loading parameters, electromagnetic interference, mechanical loads, and process fluid conditions. (3) Confirm that the installed equipment is qualified to perform its Design Basis function for its operational life and that effective programs are adequate to monitor for timely maintenance or replacement, as required. (4) Confirm the adequacy and effectiveness of the process for ensuring compliance with these programs and of documented previous qualification measures taken to ensure qualification throughout the equipment's installed life (i.e., prescribed testing, calibration, maintenance, and parts replacement). (5) Confirm the adequacy and effectiveness of the surveillance program and a feedback procedure to ensure aging degradation of qualified equipment remains insignificant. (6) Review and confirm the previous work from Pickering PSR2 (if available), the adequacy and effectiveness of the monitoring of actual environmental conditions, and identification of 'hot spots' of high activity or temperature. (7) Confirm the adequacy and effectiveness of the assessment that determines the effects of equipment failures on equipment qualification and appropriate corrective actions and/or safety improvements to maintain equipment qualification. (8) Confirm there is protection and/or adequate and effective separation of equipment important to safety from adverse environmental conditions.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 42 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

	(9) Confirm physical condition and functionality capability of qualified equipment has been checked by walkdowns. (10) Confirm that changes to equipment classification have occurred, as required, as a result of major design modifications.
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Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	43 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 4 – Aging

Intent	All SSCs important to the safety of NPPs are subject to some form of physical change caused by aging, which could eventually impair their safety functions and service lives. It is important that these change processes are understood, monitored, and managed.
Objective	Objective of the review is to determine whether aging in a NPP is being effectively managed so that required safety functions are maintained, and whether an effective Aging Management Program is in place for future plant operation.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Confirm that there is a documented method and criteria for identifying safety related SSCs covered by the Aging Management Program. (2) Confirm an effective Aging Management Program and dedicated organization with clearly defined roles and responsibilities with sufficient resources to continually assess aging effects in safety related SSCs. (3) Establish a list of SSCs covered by the aging management program and records that provide information in support of the management of aging. (4) Evaluate and document impact of potential aging degradation of safety related SSCs. (5) Confirm understanding of dominant aging mechanisms of safety related SSCs are still valid and determine if there are new aging mechanisms that needs to be considered. (6) Confirm adequacy of predictive maintenance program. (7) Confirm the existence and adequacy of programs for timely detection and mitigation of aging mechanisms and/or aging effects of any SSCs important to safety, including obsolescence of technology used in the plant or obsolescence of services or supplies external to the plant. (8) Confirm the adequacy of acceptance criteria and required safety margin for safety related SSCs for the period of PSR3 through reliability and risk assessments. (9) Confirm adequacy of management of the effects of aging on those parts of the plant that will be required for safety when the reactor has ceased operation, for example the spent fuel storage facilities. (10) Confirm the models used to predict the evolution and advancement of aging degradation are properly supported in accordance with current accepted practices pertaining to aging degradation.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 44 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 5 – Deterministic Safety Analysis

Intent	Deterministic Safety Analysis (DSA) should be conducted for each NPP, in order to confirm the design basis for SSCs important to safety and to evaluate the plant behaviour for postulated initiating events.
Objective	Objective of the review of the DSA is to determine to what extent the existing DSA remains valid when the following aspects have been taken into account: actual plant design; the actual condition of SSCs and their predicted state at the end of the period covered by PSR3; current deterministic methods; and current safety standards and knowledge. In addition, the review should also identify any weaknesses relating to the application of the defence in depth concept.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Perform review to confirm the adequacy of current deterministic safety analyses and the assumptions used to perform these analyses since the last relevant PSR/ISR assessment. (2) Perform review to evaluate any changes to the documentation and processes for defining, implementing, and maintaining the SOE since the last relevant PSR/ISR assessment. (3) Review will determine if the postulated events, event sequences, and event combinations covered by the existing analysis are sufficient when compared against those for a modern NPP in accordance with the methodology in CNSC REGDOC-2.4.1, "Deterministic Safety Analysis". (4) Perform review of any changes to the documented guidelines for DSA. (5) Evaluate the supporting analyses for design extension conditions to confirm that the arrangements aimed at preventing or mitigating severe core damage meet regulatory requirements. (6) Confirm that the impact of equipment failures and human errors, as well as the engineering and administrative measures to prevent and mitigate accidents, have been analyzed and documented. (7) Confirm that the capabilities of the plant in its current state, and where relevant with account taken of planned safety improvements, have been demonstrated to be within regulatory requirements and expectations for both normal operation and accident conditions. In addition, confirm that plans are in place to ensure that forecast operational conditions of the plant will meet acceptance criteria for the design basis, including adequacy of safety margins, throughout the period of PSR3.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 45 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

(8) Confirm the following has been addressed by the DSA in accordance with the DSA related requirements in CNSC REGDOC-2.3.2, REGDOC-2.4.1, REGDOC-2.4.2, and REGDOC-2.5.2:

- a) prediction of expected environment conditions due to BDBAs, including severe accidents;
- b) demonstration of the adequacy of the provisions for protection against severe accidents (e.g., performance expectations for containment, biological shielding, and re-criticality);
- c) demonstration of the use of the DSA in establishing and validating severe accident management and guidelines (e.g., severe accident management guidelines);
- d) confirmation that identification of events includes at-power and shutdown states, as well as common-cause events affecting multiple reactor units on a site; and
- e) evaluation of the effectiveness of keeping the safety analysis report up to date in light of subsequent modifications and changes to operation, testing, maintenance, and other practices.

Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	46 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 6 – Probabilistic Safety Assessment

Intent	A review of the Probabilistic Safety Assessment (PSA) should be conducted to identify weaknesses in the design and operation of the plant and, as part of the global assessment, to evaluate and compare proposed safety improvements.
Objective	Objective of the review of the PSA is to determine to what extent the existing PSA remains valid as a representative model of the plant when the following aspects have been taken into account: changes in the design and operation of the plant; new technical information, current methods; and new operational data.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Confirm adequacy of PSA and the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case. (2) Confirm adequacy of processes to assess the impact of changes in plant design, operation, and plant specific failure data and update the PSA to reflect the current plant status as required. (3) Confirm adequacy of guidelines to account for operator actions, common cause events, cross-link effects, redundancy, and diversity. (4) Confirm that the accident management programs for accident conditions (design basis accident conditions and design extension conditions) are consistent with PSA models and results. (5) Confirm that the results of the PSA show that risks are sufficiently low and well balanced for all postulated initiating events and operational states, and meet relevant probabilistic safety criteria. (6) Review the extent to which hazards are represented in the PSA to verify that omissions are based on site specific justifications and that these omissions do not weaken the overall risk assessment for the plant.

Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	47 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 7 – Hazard Analysis

Intent	To ensure the delivery of required safety functions and operator actions, SSCs important to safety, including the main control room and the secondary control area, should be protected against relevant internal and external hazards.
Objective	Objective of the review of hazard analysis is to determine the adequacy of protection of the NPP against internal and external hazards with account taken of the actual plant design, actual site characteristics, the actual condition of SSCs and their predicted state at the end of the period covered by the PSR, and current analytical methods, safety standards, and knowledge.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Perform an assessment of the Deterministic and Probabilistic analyses to confirm adequacy of hazard analyses for internal and external hazards that are not screened out as part of hazard screening analysis. (2) Confirm that the analyses and/or methods take into account the plant design and the condition of SSCs important to safety (both at present and predicted for the end of the period covered by PSR3). (3) For each relevant hazard, verify, by means of current analytical techniques and data, that the frequency of occurrence and/or the consequences of the hazard are sufficiently low so that either no specific protective measures are necessary, or the preventive and mitigatory measures in place are adequate.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 48 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 8 – Safety Performance

Intent	Safety performance is determined from assessment of operating experience, including safety related events, and records of the unavailability of safety systems, radiation doses, and the generation of radioactive waste and discharges of radioactive effluents.
Objective	Objective of the review of Safety Performance is to determine the safety performance of the NPP and its trends from records of operating experience.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Confirm adequacy of system for identifying, classifying, and recording safety related incidents and operating experience including: • Safety related incidents, low level events and near misses; • Safety related operational data; • Maintenance, inspection, and testing; • Replacements of SSCs important to safety owing to failure or obsolescence; • Modifications, either temporary or permanent, to SSCs important to safety; • Unavailability of safety systems; • Radiation doses (to workers, including contractors); • Off-site contamination and radiation levels; • Discharges of radioactive effluents; • Generation of radioactive waste; • Non-radioactive effluents, including hazardous substances. (2) Confirm that safety related incidents are investigated using root cause analysis and that lessons learned from investigation of these incidents are fed back into the conduct of Operations and Maintenance. (3) Confirm that the results of the root cause analysis are used to minimize the chances of the same incident reoccurring. (4) Confirm that information from trend analysis of safety related incidents is fed back into the conduct of Operations and/or Maintenance. (5) Confirm there is an adequate set of safety performance indicators that provides a systematic and comprehensive method to record, trend and analyze safety related data including the major system parameters, and maintenance and inspection records. Performance indicators may include: • Frequency of unplanned trips while the reactor is critical • Satisfactory performance of safety system tests within required limits • Special Safety System unavailability

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 49 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

- Reliability of SIS
- Collective annual radiation dose of plant staff
- Amount of gaseous and liquid radioactive release relative to permitted limits
- Heavy water escape and loss rates
- Fuel reliability
- Chemistry index
- Volume of Low Level radioactive waste
- Change control index
- Maintenance backlog
- Training
- Environment Index
- Non-radioactive effluents, including hazardous substances
- Non-radioactive wastes
- Spills

(6) Confirm that for cases where performance indicators show an unsatisfactory trend, corrective action is taken.

(7) Review the adequacy of:

- Records of the integrity of physical barriers for the containment of radioactive material.
- Records of radiation doses to persons on the site.
- Records of data from off-site radiation monitoring and records of the quantities of radioactive effluents.
- Records of non-radioactive effluents, including hazardous substances.
- Records of radioactive and non-radioactive waste.
- Records of spills.
- Records of other environmental impacts.

(8) Consider the effects of any changes in operation at the plant on safety performance. In particular, confirm that current indicators and other safety performance methods continue to be relevant in the context of current and future operations, and confirm that only relevant data and records are used.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 50 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 9 – Use of Experience from Other NPPs and Research Findings

Intent	Experience from other NPPs, and sometimes from non-nuclear facilities, together with research findings, can reveal previously unknown safety weaknesses or can help in solving existing problems. IAEA Specific Safety Requirements SSR-2/2, requirement 24 [31] requires the operating organization to obtain and evaluate information on operating experience at other plants and to derive lessons for its own operations. This should include information from other plants for which the operating organization is responsible and wider experience in Canada and in other countries, including relevant information from non-nuclear facilities.
Objective	Objective of the review of experience from other plants and research findings is to determine whether there is adequate feedback of safety experience from other NPPs and of the findings of research.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Confirm adequacy of the program for the sending and receiving of experience relevant to safety to and from other NPPs and relevant nonnuclear plants. (2) Confirm adequacy of the program for receiving of information on the findings of relevant research programs. (3) Confirm the adequacy of the process for assessing the significance of operating experience from other plants and incorporating the lessons learned into improving safety performance at the station. (4) Confirm the adequacy of the process for assessing the significance of research findings and technology developments and for incorporating relevant improvements into the station's design and operation. (5) Review adequacy and effectiveness of the feedback arrangements and timely implementation of assessment findings. (Assess program audit results.) (6) List the major operating experience events and resulting plant changes.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 51 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 10 – Organization, Management System and Safety Culture

Intent	The operating organization is required to have in place a management system that ensures that policies and objectives are implemented in a safe, efficient, and effective manner. Similarly, the organization should have a strong safety culture so that all individuals carry out duties important to safety correctly, with alertness, due thought, full knowledge, sound judgement, and a proper sense of accountability.
Objective	Objective of the review of organization and administration is to determine whether the organization and management system and safety culture are adequate and effective for ensuring the safe operation.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Review organization and administrative procedures to ensure they play a significant role in defining safety culture and evaluate the adequacy of safety culture indicators. (2) Confirm adequacy of the safety policy to ensure that safety takes precedence over production where a conflict between these two requirements exists. (3) Identify the method for setting performance targets and confirm that these targets are regularly and systematically reviewed. Confirm that appropriate actions are initiated if safety targets are not met. (4) Confirm that the published Nuclear organization clearly defines the roles and responsibilities of all individuals and work groups who are involved in activities that could influence the safe operation of the station. Ensure that this organization is understood, and that adequate and effective procedures are in place to ensure the availability of these resources and control changes to this organization. (5) Establish that mechanisms for maintaining configuration control of the plant and its documentation are effective and up to date. (6) Confirm the adequacy of the formal arrangements for employing external technical, maintenance or other specialist staff, and confirm that the contracting procedures ensure that contract employees are qualified to do the work assigned to them. (7) Confirm the adequacy of the approved Quality Assurance program and that regular Quality Assurance audits are conducted involving both internal and independent assessors. (8) Confirm that a program for self-assessment and continuous improvement has been adequately and effectively implemented including feedback of experience relating to organizational and management failures. (9) Confirm the adequacy of the system to ensure that comprehensive, easily retrievable, and auditable records exist of baseline design information, and operational and maintenance history. (10) Confirm the adequacy of the framework to support the management of regulatory affairs.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 52 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

	(11) Confirm that the organization and management system include the processes and supporting information that explain how work is to be specified, prepared, reviewed, performed, recorded, assessed, and improved. (12) Confirm there is control of purchasing of equipment and services where this affects plant safety. (13) Confirm there are comprehensive policies in place for the communication of safety issues. (14) Confirm that a questioning attitude exists, and conservative decision making is undertaken in the organization. (15) Verify that there is a process in place for prioritization of safety issues, with realistic objectives and timescales that ensures that these issues receive proper resources.
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Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	53 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 11 – Procedures

Intent	Procedures important to the safety of the NPP should be comprehensive, validated, formally approved, appropriately distributed, and subject to rigorous management control. In addition, the procedures should be unambiguous and relevant to the actual plant (with modifications taken into account); they should reflect current operating practices and due consideration should be given to human factor aspects (for example, whether they are user friendly).
Objective	Objective of the review of the procedures of a NPP is to determine whether the procedures are of an adequate standard.
Review Tasks	The following Review Tasks are intended to support the Objective above: <i>The review of this safety factor will focus on those procedures that have the highest safety significance.</i> (1) Confirm the adequacy of the process for the development, approval, and documenting of all safety related procedures. (2) Confirm the adequacy of the process for modifying procedures affecting safety, including adequate arrangements for tracking changes. (3) Confirm the adequacy of the program for assessing procedures and performance audits to determine if there is regular review and maintenance of these procedures. (4) Confirm that self-assessments are performed to ensure that the procedures are followed. (5) Establish that there is a means for assessing the adequacy of safety related procedures in comparison with industry good practices. (6) Confirm the adequacy of the operating procedures that apply comprehensively to normal, abnormal, and emergency conditions (including anticipated operational occurrences, design basis accident conditions, post-accident conditions, and design extension conditions). (7) Confirm there is a means for assuring the clarity of procedures taking into account human factors. (8) Evaluate processes to update procedures to allow for changes in the assumptions made and/or the limits and conditions arising from the safety analysis, plant design, and operating experience. (9) Verify that the analysis and justification of the accident management procedures are documented. (10) Verify that an appropriate process is in place for the categorization of procedures in accordance with their significance to safety. (11) Examine whether there is adequate involvement in the development of procedures by the staff who will use them. (12) The distribution process for the control, copying, and removal of obsolete versions of procedures, so that only the last approved edition is used. (13) Evaluate audits, self-assessments, safety performance, and events to determine whether there is adequate understanding and acceptance of these procedures by managers and staff.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 54 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 12 – Human Factors

Intent	Human factors influence all aspects of the safety of an NPP. The review should examine the human factors at the plant and within the operating organization to determine whether these correspond to accepted good practices and to verify that they do not present an unacceptable contribution to risk. In particular, the review should determine whether operator actions claimed to be in support of safety are feasible and properly supported.
Objective	Objective of the review of human factors is to determine the status of the various human factors that may affect the safe operation of the NPP.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Confirm that there are procedures to ensure that a minimum number of qualified staff, appropriate to the operating state of the plant, is available at all times. (2) Confirm that adequate staff training facilities, training staff, and training programs exist. (3) Confirm that the method of selecting staff for new positions and for promotions involves systematic and validated staff selection methods and a method for succession planning. (4) Confirm that there are appropriate programs for certification, initial, refresher, and upgrade training. For operating staff, this should include the use of simulators. (5) Establish that there is training in safety culture, including for management staff, that includes work supervision practices and internal communication practices and expectations. (6) Confirm there are fitness for duty guidelines relating to hours of work, health, and substance abuse. (7) Confirm that the human-machine interface is considered in the design of the control room and other workstations, that analysis of human information requirements and task workload is performed, and that there is linkage to the PSA, DSA, and Hazard Analysis. This review should include a discussion of how CNSC REGDOC-2.5.1 and CSA N290.12 are relevant to the design of Pickering. (8) Confirm the style and clarity of procedures provides an appropriate level of detailed guidance for staff through a review of plant events identifying inadequate procedures as a contributing cause.

Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	55 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 13 – Emergency Planning

Intent	The design and operation of a NPP are required to prevent or otherwise minimize releases of radioactive substances that could give rise to risks to workers or the public or to the environment. Emergency planning for the possibility of such releases is a prudent and necessary action, not only for the operating organization but also for local and national authorities.
Objective	Objective of the review of emergency planning is to determine (a) whether the operating organization has adequate plans, staff, facilities, and equipment for dealing with emergencies and (b) whether the operating organization's arrangements have been adequately coordinated with local and national systems and are regularly exercised.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Confirm the full range of accidents and radiation emergencies have been identified and studied. (2) Confirm the appropriate response and mitigation strategies have been developed and have taken account of major changes at site and around the site (industrial, commercial, residential development). (3) Confirm that the station organization includes dedicated Emergency Response personnel available at all times to handle accident and emergencies within the required time limit. (4) Assess the adequacy of the training program for emergency response personnel including training, emergency exercises and qualification records. (5) Confirm there is a process for notification of staff that will be brought in to assist in the management of the response in the longer term. (6) Determine that there is a classification of accidents to guide the type of response. (7) Confirm there is a mechanism for notifying and informing relevant off-site organizations such as the police, fire departments, hospitals, ambulance services, regulatory bodies, local authorities, government, public welfare authorities and the news media. (8) Confirm the availability of sufficient communications equipment at the plant and at the off-site Emergency Centre to permit effective communications with Emergency Response Teams, both on and off site. (9) Assess adequacy of the emergency response procedures and training and exercises for all site staff. Confirm that integrated and partial emergency exercises have been conducted to check satisfactory function of the emergency organization and its equipment. (10) Confirm that the adequacy of on-site equipment and facilities for emergencies and offsite emergency facilities or locations, including walkdowns of relevant areas on and off the site. (11) Confirm the adequacy of a program for Severe Accident Management.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 56 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 14 – Radiological Impact on the Environment

Intent	The operating organization should have in place an established and effective monitoring program that provides data on the radiological impact of the NPP on its surroundings.
Objective	Objective of the review of the radiological impact of the NPP on the environment is to determine whether the operating organization has an adequate program for surveillance of the radiological impact of the plant on the environment.
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Confirm there are procedures in place to ensure that permitted release limits of radiological substances are not exceeded and, if they are, that appropriate corrective action is taken to minimize the possibility of limits being exceeded in the future. (2) Confirm records of radiological effluent release are maintained in accordance with regulatory requirements. (3) Confirm that a program exists to define the requirements for alarm systems to respond to unplanned effluent releases from on-site facilities. (4) Confirm the environmental data recorded by the station is published and is available on request to the general public. (5) Review the environmental data recorded by the station and compare with the values measured before the plant was put into operation. (6) Confirm there is a process to address changes in the use of land external to the site with respect to the impact on public safety from facility releases. (7) Confirm that the monitoring program is appropriate and sufficiently comprehensive. In particular, confirm that the radiological impact of the plant on the environment is not significant compared with that due to other sources of radiation.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 57 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Safety Factor 15 – Radiation Protection

Objective	The objective of this safety factor is to confirm that Radiation Protection has been adequately accounted for in the design and operation of the reactor facility, that Radiation Protection provisions (including design and equipment) provide adequate protection of persons from the harmful effects of radiation and ensure that contamination and radiation exposures and doses to persons are monitored and controlled, and maintained as low as reasonably achievable (ALARA).
Review Tasks	The following Review Tasks are intended to support the Objective above: (1) Confirm the adequacy of the reactor design features for Radiation Protection. (2) Confirm the adequacy of the Radiation Protection equipment and instrumentation for radiation monitoring. (3) Confirm that adequate provisions are in place to address Radiation Protection of the public and workers during nuclear emergencies. (4) Confirm that the Radiation Protection provisions have been improved as the result of external operating experience. (5) The review will demonstrate that the ALARA principle has been incorporated in any modifications of the reactor design and operational programs and arrangements.

Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	58 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

Appendix D Modern Laws, Regulations, Codes, and Standards applicable to PSR3

Completing a PSR in accordance with CNSC regulatory requirements in REGDOC-2.3.3 [1] and CSA N290.18 [3], consistent with IAEA Safety Guide SSG-25 [2], is one licensing component that will be required for life extension of the Pickering units. The PSR being scoped as part of this work is a subsequent PSR referred to as Pickering PSR3, which builds upon earlier OPG PSR work, including:

- The PB-ISR, performed in support of refurbishment and continued operation for another 30 years of the Pickering 5-8 units,
- The D-ISR, performed in support of refurbishment and continued operation of the Darlington units,
- The P-PSR2, performed in support of the evaluation of continued operation of the Pickering NGS units beyond the year 2020, and,
- The D-PSR, performed in support of licence renewal (10 years of continued operation).

PSR3 differs in scope from the Pickering PSR2 performed in 2018, in that PSR3 is specific to the life extension of PNGS 058 for 30 years, whereas PSR2 was done in support of licence renewal and evaluated continued operation beyond 2020 up to August 31, 2028.

One of the first inputs required for this project is a list of the LRCSs that are required to be included in the PSR3 Assessment Basis. This list of LRCSs will serve as an input to subsequent assessments which will identify potential issues that may result from PSR3 Safety Factor reviews and will require resolution (i.e., PNGS Refurbishment scope items).

The methodology to identify the LRCSs that are applicable to PSR3 involved creating a broad list from multiple sources (candidate LRCSs), and assessing the LRCSs for applicability, to establish a list to be used in the PSR3 Assessment Basis.

The overall process can be summarized as:

1. A potential set of candidate LRCSs were identified.
2. A series of criteria detailed in Section D.1.0 were applied to screen and select the LRCSs that will be included in PSR3 with rationale provided.
3. A review type for each LRCS is determined for the established list of LRCSs according to the rules presented in Section D.1.0. The applicable review types are Clause-by-Clause (CBC), High Level (HL) or Incremental (IC).

The result of the identification and selection process was a comprehensive set of LRCS that will be used in the PSR3 Assessment Basis which is presented in Table D1. This table identifies the modern version/date of the LRCS, the type of review that is proposed for PSR3, and the most

Report

OPG Proprietary		
Document Number:	Usage Classification:	
P-REP-03680-00054	N/A	
Sheet Number:	Revision Number:	Page:
N/A	R001	59 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

applicable Safety Factors. A documentation freeze date of December 31, 2023 is used for the PSR3.

Detailed methodology is provided in Section D.1.0, along with the complete list of candidate LRCSSs and the rationale for their inclusion or exclusion from the PSR3 Assessment Basis.

D.1.0 DETERMINATION OF THE POTENTIAL LRCS IN THE PSR3 ASSESSMENT BASIS

There was a detailed process that was followed to determine the potential LRCSSs in the PSR3 Assessment Basis. The executed steps below are generally based on the process employed in Appendix D of the Pickering PSR2 Basis Document [6] and Appendix D of the D-PSR Basis Document [7], and that the PSR3 will support extended operation for 30 years.

1.0 Identifying Candidate LRCSSs

The following details the process applied to identify the candidate LRCSSs.

- 1.1 Assemble all LRCSSs identified in the PNGS LCH [4] and group the LRCSSs into those that are:
 - a. Directly referenced in the PROL (listed in LCH Appendix D).
 - b. Identified in the LCH as guidance (listed in LCH Appendix E).
 - c. Referenced in the main body of the LCH but not listed in the LCH Appendices.
- 1.2 Identify CNSC Regulations from the CNSC Acts and Regulations webpage [32] published on or prior to the PSR3 freeze date. The regulations listed on this CNSC website consist of the following categories:
 - a. Those in Section 2.0 – SCAs. SCAs are the technical topics used by the CNSC to assess, review, verify, and report on regulatory requirements and performance across all regulated facilities and activities. These documents provide requirements and guidance and are most relevant to the scope of the PSR. These documents are presented into two sub-categories:
 - i. those that are currently issued or under development.
 - ii. those that are superseded by the documents in subcategory i.
 - b. Those in Section 1.0 – Regulated Facilities and Activities. Documents in this category provide guidance to applicants about the information required for licence applications and to licensees about the requirements for conducting the licensed activity, and generally fall outside of the scope of PSR.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 60 of 72
Title: Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document		

- c. Those in Section 3.0 – Other Regulatory Areas. Documents in this category address the full scope of CNSC regulatory activities but these topics generally fall outside of the scope of PSR.

- 1.3 Confirm that the Regulations listed by the CNSC as noted in Step 1.2 are up-to-date by cross-referencing to the CNSC Regulatory Framework webpage [33].
- 1.4 Identify the LRCSs from the D-PSR Assessment Basis [7] that were not identified above.
- 1.5 Identify the LRCSs from the PSR2 Assessment Basis [6] that were not identified above.
- 1.6 Identify the LRCSs from the D-ISR Assessment Basis [34] that were not identified above.
- 1.7 Identify the LRCSs from the PB-ISR Assessment Basis [35] that were not identified above.
- 1.8 Identify new CNSC Regulatory Documents that are currently in development and may be issued prior to the start of PSR3.
- 1.9 Identify new CSA N-Series Standards that are published but not identified in any of the above lists. Identify new CSA N-Series Standards that are currently in development and may be issued prior to the freeze date of the PSR3.

2.0 Screening Candidate List and Identify the PSR3 Assessment Basis

The following criteria were applied to remove LRCSs from the candidate list:

- 2.1 If the LRCS is identified by more than one source (e.g., if there are duplicate entries from the PROL, PSR2 Basis, D-PSR Basis, etc.).
- 2.2 If the LRCS is not in the PSR2 Basis or D-PSR Basis, not on the CNSC website listings as currently applicable, and not one of the PROL LRCS (LCH Appendix D), then it is not part of the PSR3 Assessment Basis. An exception was made for new REGDOCs and Standards that may not currently be listed on the CNSC website since the code/standard has not been published.
- 2.3 If the LRCS is not listed in the PROL (LCH Appendix D) and is in only one of the PSR2 basis, or D-PSR basis, or the CNSC website listings as currently applicable, then it was assessed for significance on a case-by-case basis and was excluded from the PSR3 Assessment Basis where it covered an area that is not significant to nuclear safety.
- 2.4 LRCSs that are uniquely related to the CNSC Safety Control Areas for “Packaging and Transport”, “Security” and “Safeguards and Non-proliferation” are not part of the PSR3 Assessment Basis since they are not addressed in the Safety Factors (REGDOC-2.3.3). Requirements related to “Ontario Nuclear Funds Agreement”, and “public engagement and information” are not included in the PSR3 Assessment Basis. LRCSs related to Waste Management are in the scope of PSR3 where it relates to activities that are governed by the

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 61 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

PNGS PROL (as opposed to the Pickering Waste Management Facility Waste Facility Operating Licence).

- 2.5 Since the portions of the Pickering Nuclear site that are subject to PSR3 are federally regulated, the National Building Code of Canada (NBCC) and the National Fire Code of Canada (NFCC) are included in the PSR3 Assessment Basis, while the Ontario Building Code and the Ontario Fire Code are not included in the PSR3 Assessment Basis.
- 2.6 Codes and Standards defined as guidelines are not part of the PSR3 Assessment Basis unless they are cited as mandatory within a PROL Code or Standard.
- 2.7 IAEA Codes and Standards are generally not part of the PSR3 Assessment Basis as they may have been used as references in the development of the Codes and Standards listed by the CNSC as being current, or they may duplicate a Canadian Regulatory Document or CSA Standard that addresses the same topic area. However, if an IAEA Code or Standard is identified as mandatory within a PROL Code or Standard, it will be added to the PSR3 Assessment Basis.
- 2.8 Other national regulator requirements and industry organization standards are not part of the PSR3 Assessment Basis. In some instances, these may have been used as references in the development of the Codes and Standards listed by the CNSC as being current, or they may duplicate a Canadian Regulatory Document or CSA Standard that addresses the same topic area.
- 2.9 CNSC Policy documents are excluded from the PSR3 Assessment Basis since they are intended for internal CNSC application.
- 2.10 Laws and Regulations are not considered to be part of the PSR3 Assessment Basis if they are not listed in the PROL (LCH Appendix D).

Following the exclusions in Steps 2.1 to 2.10, the remaining LRCSs represent those that will form part of the PSR3 Assessment Basis, subject to the following additional considerations:

- 2.11 The version of the LRCS as of December 31, 2023 (the PSR3 documentation freeze date) was used.

3.0 Determination of the Review Type

In determining the type of review (CBC, HL, or IC) for a LRCS, consideration is given to the type of review the LRCS was subjected to in the previous PSR/ISR. Although PSR3 will build upon previous ISRs and PSRs, the objective of PSR3 is to support Pickering Refurbishment for an additional 30 years of operation.

The specific review type is a case-by-case determination for each LRCS; however, the following general rules are applied:

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 62 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

- 3.1 If an LRCS was not reviewed previously, a CBC or HL review is recommended based on the applicability of the LRCS.
- 3.2 For LRCSs directly related to the design and condition of SSCs, if the LRCS was previously reviewed in PB-ISR, an incremental review is performed for PSR3. As part of performing the incremental review, it may be determined that the LRCS has been substantially modified since the PB-ISR review or the PB-ISR review is no longer applicable to the scope of PSR3, in which case a HL or CBC review may be warranted. Otherwise, even if the LRCS was previously reviewed in PSR2 or D-PSR, if it was not reviewed in PB-ISR, a CBC review is performed.
- 3.3 For programmatic LRCSs, if the LRCS was reviewed in PSR2, an incremental review is performed, unless the LRCS has been substantially modified in which case a HL or CBC review may be warranted. If the LRCS was not reviewed in PSR2 but was reviewed in D-PSR, the level of review is determined based on the applicability of the D-PSR code review to PSR3.
- 3.4 If the LRCS was reviewed previously and has not been revised since, an incremental review is still performed in order to confirm the applicability of the previous code review(s) in the context of extended operation.

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 63 of 72

Title:
Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

Table D1: List of LRCSs in the PSR3 Assessment Basis

#	Document Number	Document Name	Current Version	PSR3 Review Type	Applicable Safety Factors
LRCS from the Pickering LCH (Appendix D - List of Licensing Publications)					
1	CSA N285.0	General requirements for pressure-retaining systems and components in CANDU nuclear power plants	2023	IC	1, 2, 4
2	CSA N285.4	Periodic inspection of CANDU nuclear power plant components	2023	CBC	1, 2, 4
3	CSA N285.5	Periodic inspection of CANDU nuclear power plant containment components	2022	CBC	1, 2, 3, 4
4	CSA N285.6	Material Standards for reactor components for CANDU nuclear power plants	2023	IC	1
5	CSA N285.8	Technical requirements for in-service evaluation of zirconium alloy pressure tubes in CANDU reactors	2023	CBC	2, 4
6	CSA N286	Management system requirements for nuclear facilities	2012 (R2022)	IC	1, 2, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 15
7	CSA N286.7	Quality assurance of analytical, scientific and design computer programs	2016 (R2021)	IC	1, 5, 6, 7, 10
8	CSA N287.1	General requirements for concrete containment structures for nuclear power plants	2014 (R2019)	IC	1

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 64 of 72

Title:
Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

#	Document Number	Document Name	Current Version	PSR3 Review Type	Applicable Safety Factors
9	CSA N287.2	Material requirements for concrete containment structures for CANDU nuclear power plants	2017 (R2022)	IC	1, 2, 3, 4
10	CSA N287.7	In-service examination and testing requirements for concrete containment structures for CANDU nuclear power plants	2017 (R2022)	IC	2, 3, 4
11	CSA N288.1	Guidelines for modelling radionuclide environmental transport, fate, and exposure associated with the normal operation of nuclear facilities	2020	IC	8, 14
12	CSA N288.3.4	Performance testing of nuclear air-cleaning systems at nuclear facilities	2013 (R2022)	CBC	2, 4, 8, 14
13	CSA N288.4	Environmental monitoring programs at nuclear facilities and uranium mines and mills	2019	IC	8, 14
14	CSA N288.5	Effluent and emissions monitoring programs at nuclear facilities	2022	CBC	8, 14
15	CSA N288.6	Environmental risk assessments at nuclear facilities and uranium mines and mills	2022	CBC	8, 14
16	CSA N288.7	Groundwater protection and monitoring programs for nuclear facilities and uranium mines and mills	2023	CBC	14
17	CSA N288.8	Establishing and implementing action levels for releases to the environment from nuclear facilities	2017 with Update No. 1 (R2022)	CBC	8, 14

Report**OPG Proprietary**

Document Number:

P-REP-03680-00054

Usage Classification:

N/A

Sheet Number:

N/A

Revision Number:

R001

Page:

65 of 72**Title:**
Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

#	Document Number	Document Name	Current Version	PSR3 Review Type	Applicable Safety Factors
18	CSA N289.1	General requirements for seismic design and qualification of CANDU nuclear power plants	2023	IC	1, 3
19	CSA N290.12	Human factors in design for nuclear power plants	2023	IC	1, 12
20	CSA N290.13	Environmental qualification of equipment for CANDU nuclear power plants	2018	CBC	3, 4
21	CSA N290.14	Qualification of digital hardware and software for use in instrumentation and control applications for nuclear power plants	2015 (R2020)	CBC	1
22	CSA N290.15	Requirements for the safe operating envelope for nuclear power plants	2019	IC	5, 8
23	CSA N291	Requirements for nuclear safety-related structures for CANDU nuclear power plants	2019	CBC	1, 2, 4
24	CSA N292.3	Management of low and intermediate-level radioactive waste	2014 (R2019)	CBC	8, 10, 14
25	CSA N293	Fire protection for nuclear power plants	2023	IC	1, 7, 13
26	REGDOC-2.1.2	Safety Culture	2018	CBC	10, 11
27	REGDOC-2.2.2	Personnel Training, Version 2	2016	IC	10, 12, 13
28	REGDOC-2.2.3	Personnel Certification Volume III: Certification of Persons Working at Nuclear Power Plants	2023	IC	10, 12
29	REGDOC-2.2.4	Fitness for Duty: Managing Worker Fatigue	2017	CBC	10, 12

Report**OPG Proprietary**

Document Number:

P-REP-03680-00054

Usage Classification:

N/A

Sheet Number:

N/A

Revision Number:

R001

Page:

66 of 72**Title:**
Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

#	Document Number	Document Name	Current Version	PSR3 Review Type	Applicable Safety Factors
30	REGDOC-2.2.4	Fitness for Duty, Volume II: Managing Alcohol and Drug Use, Version 3	2021	CBC	8, 10, 12
31	REGDOC-2.3.2	Accident Management, Version 2	2015	CBC	1, 5, 6, 7, 8, 10, 13
32	REGDOC-2.3.3	Periodic Safety Review	2015	CBC	8
33	REGDOC-2.4.1	Deterministic Safety Analysis	2014	CBC	5, 7
34	REGDOC-2.4.2	Probabilistic Safety Assessment (PSA) for Nuclear Power Plants	2022	CBC	6, 7
35	REGDOC-2.6.1	Reliability Programs for Nuclear Power Plants	2017	CBC	2, 4, 6, 8
36	REGDOC-2.6.2	Maintenance Programs for Nuclear Power Plants	2017	CBC	2, 3, 4, 8
37	REGDOC-2.6.3	Aging Management	2014	CBC	2, 3, 4
38	REGDOC-2.9.1	Environmental Protection: Environmental Principles, Assessments and Protection Measures, Version 1.2	2020	IC	8, 14
39	REGDOC-2.10.1	Nuclear Emergency Preparedness and Response, Version 2	2016	IC	13
40	REGDOC-3.1.1	Reporting Requirements for Nuclear Power Plants, Version 2	2016	IC	8, 10
41	CSA B51	Boiler, pressure vessel, and pressure piping code	2019	IC	1
42	ASME BPVC	Boiler and Pressure Vessel Code, ASME Section III	2021	IC	1

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 67 of 72

Title:
Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

#	Document Number	Document Name	Current Version	PSR3 Review Type	Applicable Safety Factors
43	ASME BPVC	Boiler and Pressure Vessel Code, ASME Section VIII	2021	IC	1
44	ASME B31.1	Power Piping	2018	CBC	1
45	ASME B31.3	Process Piping	2016	HL	1
46	ASME B31.5	Refrigeration Piping and Heat Transfer Component	2016	HL	1
LRCS from the Pickering LCH (Appendix E – List of Guidance Publications)					
47	CSA N285.7	Periodic inspection of CANDU nuclear power plant balance of plant systems and components	2021	CBC	2, 4
48	CSA N286.10	Configuration management for high energy reactor facilities	2016 (R2021)	IC	1, 10
49	CSA N287.3	Design requirements for concrete containment structures for NPPs	2014 with Update No. 1 (R2019)	IC	1, 2, 3, 4
50	CSA N287.4	Construction, fabrication, and installation requirements for concrete containment structures for CANDU nuclear power plants	2019	HL	1
51	CSA N287.5	Examination and testing requirements for concrete containment structures for nuclear power plants	2020	CBC	2, 4
52	CSA N287.8	Aging management for concrete containment structures for nuclear power plants	2015 (R2020)	CBC	2, 4

Report**OPG Proprietary**

Document Number:

P-REP-03680-00054

Usage Classification:

N/A

Sheet Number:

N/A

Revision Number:

R001

Page:

68 of 72**Title:**
Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

#	Document Number	Document Name	Current Version	PSR3 Review Type	Applicable Safety Factors
53	CSA N288.2	Guidelines for calculating the radiological consequences to the public of a release of airborne radioactive material for nuclear reactor accidents	2019 with Update No. 1	IC	5
54	CSA N288.9	Guidance for design of fish impingement and entrainment programs at nuclear facilities	2018 (R2023)	HL	1, 14
55	CSA N289.2	Ground motion determination for seismic qualification of CANDU nuclear power plants	2021	IC	1, 3
56	CSA N289.3	Design procedures for seismic qualification of nuclear power plants	2020	IC	1, 3
57	CSA N289.4	Testing procedures for seismic qualification of nuclear power plant structures, systems, and components	2022	CBC	1, 3
58	CSA N289.5	Seismic instrumentation requirements for nuclear power plants and nuclear facilities	2012 with Update No. 1 (R2022)	IC	1, 3
59	CSA N290.0	General requirements for safety systems of nuclear power plants	2017 (R2022)	CBC	1, 3
60	CSA N290.1	Requirements for the shutdown systems of nuclear power plants	2013 (R2023)	CBC	1, 3
61	CSA N290.2	Requirements for emergency core cooling systems of nuclear power plants	2017 (R2022)	CBC	1, 3
62	CSA N290.3	Requirements for containment system of nuclear power plants	2022	CBC	1, 3

Report**OPG Proprietary**

Document Number:

P-REP-03680-00054

Usage Classification:

N/A

Sheet Number:

N/A

Revision Number:

R001

Page:

69 of 72**Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document**

#	Document Number	Document Name	Current Version	PSR3 Review Type	Applicable Safety Factors
63	CSA N290.4	Requirements for reactor control systems of nuclear power plants	2019	IC	1, 3
64	CSA N290.5	Requirements for electrical power and instrument air systems of CANDU nuclear power plants	2016 (R2021)	IC	1, 3
65	CSA N290.6	Requirements for monitoring and display of nuclear power plant safety functions in the event of an accident	2016 (R2021)	IC	1, 3, 13
66	CSA N290.11	Requirements for reactor heat removal capability during outage of nuclear power plants	2021	CBC	1, 11
67	CSA N290.16	Requirements for beyond design basis accidents	2016 (R2021)	CBC	1, 5, 6, 7, 13
68	CSA N290.17	Probabilistic safety assessment for nuclear power plants	2023	CBC	6, 7
69	CSA N290.18	Periodic safety review for nuclear power plants	2017 (R2022)	CBC	8
70	CSA N292.0	General principles for the management of radioactive waste and irradiated fuel	2019	CBC	8, 10, 14, 15
71	CSA N1600	General requirements for nuclear emergency management programs	2021	IC	13, 15
72	REGDOC-2.1.1	Management System	2019	IC	5, 6, 9, 10, 11
73	REGDOC-2.2.5	Minimum Shift Compliment	2019	CBC	8, 10, 12, 13
74	REGDOC-2.4.3	Nuclear Criticality Safety, Version 1.1	2020	HL	5, 8, 11

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 70 of 72

Title:
Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

#	Document Number	Document Name	Current Version	PSR3 Review Type	Applicable Safety Factors
75	REGDOC-2.5.1	General Design Considerations: Human Factors	2019	IC	1, 12
76	REGDOC-2.5.2	Design of Reactor Facilities: Nuclear Power Plants	2023	CBC	1, 5, 6, 7, 12, 13, 14, 15
77	REGDOC-2.7.1	Radiation Protection	2021	CBC	8, 15
78	REGDOC-2.7.2	Dosimetry, Volume I: Ascertaining Occupational Dose	2021	CBC	8, 15
79	REGDOC-2.11.1	Waste Management, Volume I: Management of Radioactive Waste	2021	CBC	8, 10, 14
LRCS from the Pickering LCH Body of Document					
80	N.S.C.A. 1997, c.9	Nuclear Safety and Control Act	2017	IC	10
81	SOR/2000-202	General Nuclear Safety and Control Regulations	2015	IC	10, 15
82	SOR/2000-203	Radiation Protection Regulations	2021	IC	8, 15
83	SOR/2000-204	Class I Nuclear Facilities Regulations	2017	CBC	10, 15
84	NBCC	National Building Code of Canada	2020	IC	1
85	NFPA 24	Standard for the Installation of Private Fire Service Mains and Their Appurtenances	2022	CBC	1
86	PNERP	Provincial Nuclear Emergency Response Plan (PNERP) Master Plan	2017 (Master Plan) 2019	HL	13

Report

OPG Proprietary		
Document Number: P-REP-03680-00054		Usage Classification: N/A
Sheet Number: N/A	Revision Number: R001	Page: 71 of 72

Title:
Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

#	Document Number	Document Name	Current Version	PSR3 Review Type	Applicable Safety Factors
			(Implementing Plan)		
Additional LRCS on the CNSC Website					
87	CSA N290.8	Technical specification requirements for nuclear power plant components	2015 (R2020) with Update No. 1	IC	1
88	CSA N299.1, N299.2, N299.3, N299.4	Quality assurance program requirements for the supply of items and services for nuclear power plants	2019 with Update No. 1 for CSA N299.1, N299.2, N299.3	IC	1, 10
89	REGDOC-2.3.1	Conduct of Licensed Activities: Construction and Commissioning Programs	2016	CBC	1, 10
Additional LRCS included in D-PSR					
90	CSA N290.9	Reliability and maintenance programs for nuclear power plants	2019	CBC	2, 4, 6, 8
91	NFCC	National Fire Code of Canada	2020	IC	1

Report**OPG Proprietary**

Document Number:

P-REP-03680-00054

Usage Classification:

N/A

Sheet Number:

N/A

Revision Number:

R001

Page:

72 of 72

Title:

Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document

#	Document Number	Document Name	Current Version	PSR3 Review Type	Applicable Safety Factors
Additional LRCS included in PSR2					
92	NFPA 20	Standard for the Installation of Stationary Pumps for Fire Protection	2022	CBC	1
Additional LRCS included in D-ISR					
93	IAEA SSG-50	Operating Experience Feedback for Nuclear Installations	2018	HL	9
New CSA N-Series Standards					
94	CSA N288.0	Environmental management of nuclear facilities: Common requirements of the CSA N288 series of Standards	2022	CBC	14
95	CSA N290.19	Risk Informed Decision Making (RIDM) for Nuclear Reactor Facilities	2018	HL	6, 10
96	CSA N290.20	Aging management requirements for nuclear power plants	2021	CBC	2, 3, 4
97	CSA N292.4	Storage of radioactive waste and irradiated fuel	2023	CBC	8, 10, 14, 15
98	CSA N292.8	Characterization of radioactive waste and irradiated fuel	2021	CBC	8, 10, 14