ONTARIOPOWER GENERATION

Report

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DARLINGTON NGS - GLOBAL ASSESSMENT REPORT (GAR)

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Executive Summary

Ontario Power Generation's (OPG) record of safe and reliable operation of the Darlington Nuclear Generating Station (NGS), together with the strong Nuclear Safety Culture of its staff and organization, and the commitment to continuous improvement, are the foundation for extending the life of the station. Building on this foundation, OPG embarked on assessments which considered both Canadian and international modern codes and standards, to demonstrate that completion of the Integrated Implementation Plan (IIP) activities will result in a further 30 years of safe and reliable operation and the continued supply of 3600 MW of clean electrical power to the Province of Ontario.

Since beginning commercial operation, Darlington has demonstrated strong and improving operational performance with its reactor units ranking among the top performing CANDU units in the world. It is an industry leader in plant reliability with a very low forced generation loss rate. More recently, Darlington was recognized by its industry peers for achieving the highest rating of performance excellence of any nuclear power plant worldwide.

Nuclear safety is a core value at OPG. This is reflected in N-POL-0001, "Nuclear Safety Policy", and endorsed by OPG's board of directors. The policy places nuclear safety as the overriding priority above that of cost, schedule or production. It requires that all employees conduct themselves in a manner consistent with the behaviour of a healthy Nuclear Safety Culture by always considering how their everyday activities can impact on the fundamental safety functions of the station.

OPG is committed to a strong safety culture. This was demonstrated when OPG became the first employer in Ontario to be awarded the Infrastructure Health and Safety Association ZeroQuest® Platinum award. This recognition was achieved following five years of sustained safety performance. Furthermore, the strong safety culture was evident when OPG reached a significant conventional safety milestone, achieving more than 12.8 million person-hours worked at Darlington without a lost time accident from May 2008 to March 2012.

Darlington has a strong commitment to the community and the environment. It was selected from over 100 sites across North America to receive the Corporate Habitat of the Year Award from the Wildlife Habitat Council (WHC). OPG was also the recipient of the WHC William W. Howard C.E.O. Award which recognizes a corporation that has a history of striving for excellence in biodiversity, conservation, education, and outreach.

Consistent with best industry practices, OPG has taken into account the Operating Experience (OPEX) from Canada and around the world. The most significant OPEX in recent history was that from the events at Fukushima in 2011, both in terms of equipment and emergency response to very improbable events.

This report, the Global Assessment Report (GAR), is a requirement of the Canadian Nuclear Safety Commission (CNSC) regulatory document RD-360, "Life Extension of Nuclear Power Plants". The GAR is produced in accordance with the OPG Procedure N-PROC-LE-0007, "Nuclear Refurbishment – Global Assessment Report and Integrated Implementation Plan – Darlington", with the objective of providing an overall risk judgment on the acceptability of continued operation.

The Global Assessment is a systematic review of the extent to which Darlington meets the codes and standards of a new nuclear generating station and incorporates the results of two comprehensive assessments. The first is an Environmental Assessment (EA); the second is an Integrated Safety Review (ISR). The mitigating and corrective actions from these assessments are identified in the Integrated Implementation Plan (IIP). The implementation of this plan assures the continued long-term safe operation of the plant.

The EA concluded that the only residual effects of Darlington's Life Extension were minor in nature. There were four design enhancements which OPG has committed to as part of the EA and these are included in the IIP. These modifications will further increase safety margins and reduce plant risk.

The results of the ISR demonstrated that the current state of the plant and its operational performance complies closely with modern codes and standards and utilizes industry best practices. The ISR considered components that would limit safe long-term operation (e.g. fuel channels). These components are scheduled for refurbishment or replacement and are contained within the IIP, along with other activities to enhance the condition of the station.

The Global Assessment included an assessment of Defence in Depth (DID). The DID assessment was performed using the CNSC Regulatory Document RD-337, "Design of New Nuclear Power Plants". Guidance from IAEA Safety Report Series No. 46, "Assessment of Defence in Depth for Nuclear Power Plants" was also applied. This assessment verified the existence of multiple levels of barriers between radioactive materials and the environment, and concluded that no single human error, or single equipment failure, could result in any harmful radiological releases to the public or the environment.

This report presents the significant results of the EA, ISR, an analysis of the condition of the plant safety-related equipment and its required maintenance, and also a positive basis for continued operation. OPEX has been incorporated into the scope of Darlington's IIP. Modifications that provide a significant safety benefit to the plant, programs, and processes will be undertaken so that the safety of the facility will be close to that of a new nuclear power plant.

All aspects of CNSC RD-360 were evaluated. The results demonstrate that Darlington is a safe and reliable nuclear power plant today. The results also demonstrate there are opportunities for further improvements. Implementation of these improvements, as identified within the IIP, will result in Darlington being an even safer and more reliable supplier of clean electrical power to the Province of Ontario for another 30 years.

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1.0 INTRODUCTION

The Darlington Nuclear Generating Station (NGS), also referred to here as Darlington, is a four-unit generating station that includes a separately housed Tritium Removal Facility (TRF). The site is located in the Municipality of Clarington, Regional Municipality of Durham, in the Province of Ontario. Darlington has been operated successfully since the early 1990s and a Life Extension program is now underway to enable the station to be operated for up to 30 additional years.

The justification for the continued operation of Darlington following life extension is documented in accordance with the CNSC Regulatory Document RD-360, "Life Extension of Nuclear Power Plants" [R-1]. That document, hereafter referred to as RD-360, requires the licensee to demonstrate that continued station operation poses no unreasonable risk to health, safety, security of the public or the environment, and will continue to conform to international obligations. The requirements for producing the Global Assessment Report (GAR) as outlined in RD-360 have been implemented in accordance with OPG procedure N-PROC-LE-0007, "Nuclear Refurbishment – Global Assessment Report and Integrated Implementation Plan – Darlington" [R-2]. This procedure reflects CNSC expectations [R-3].

Three principal activities are considered to systematically identify the environmental and safety enhancements that will assure ongoing safe operation for the proposed Life Extension:

1. Environmental Assessment (EA)

   The EA is a comprehensive assessment of the potential impacts of future operation on the natural environment including public safety and socio-economic considerations. The EA is focused on the impacts beyond the plant boundary. The EA determines the mitigation that is necessary to ensure that the plant will not have significant adverse environmental impacts. The results of the EA are contained in the Environmental Impact Statement [R-4] including the technical support documents, and the CNSC decision as documented in the CNSC's Record of Proceedings [R-5].

2. Integrated Safety Review (ISR)

   The ISR is a systematic and comprehensive assessment of the plant design, actual condition, and of the management system used to operate and maintain the nuclear plant. The ISR enables determination of the reasonable and practical modifications that should be made to the plant design or the management system to enhance future safe operation. The results of the ISR are documented in a series of reports based on established Safety Factor review topics [R-6]. A Safety Factor is a topic required by the CNSC for inclusion in an ISR as listed in IAEA Safety Standards Series, Safety Guide No. NS-G-2.10, "Periodic Safety Reviews of Nuclear Power Plants" [R-7]. Some Safety Factors beyond those identified by IAEA were added to ensure that CNSC safety areas and programs in RD-360 are addressed.

3. Global Assessment (GA)

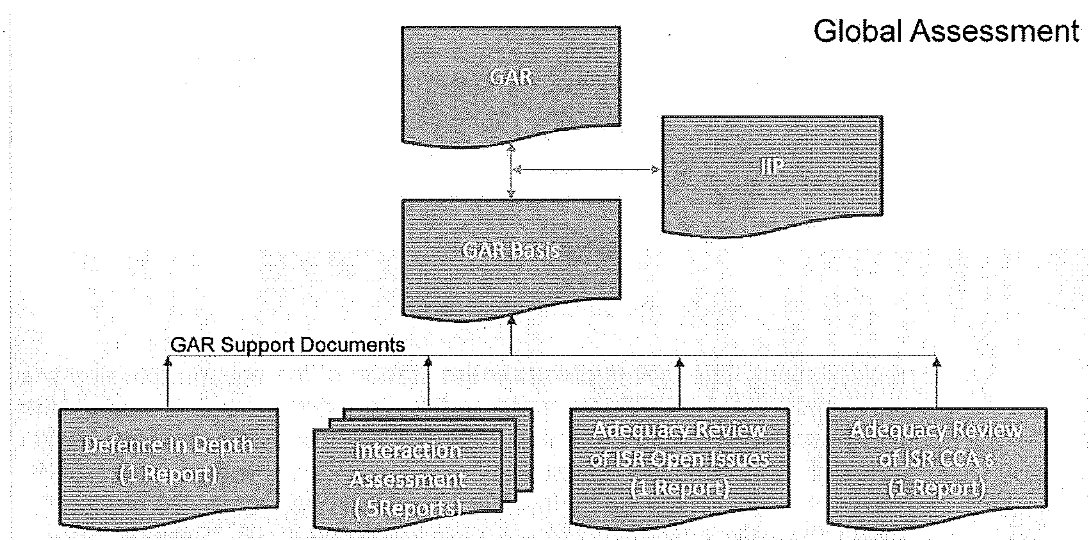
The GA uses the results of the EA and ISR and examines them in an integrated manner. It assesses the strengths, opportunities for improvement, and actions to address the opportunities for improvement, in order to provide an overall risk judgment on the acceptability of continued operation. The GA further assesses the adequacy of the actions arising from the EA and ISR that are identified for Life Extension and presents the required actions in the Integrated Implementation Plan (IIP).

This document is OPG's Global Assessment Report (GAR) for the refurbishment and continued operation of Darlington NGS. It includes:

- A description of Darlington's overall plant performance including a discussion on continuous improvement (Section 2.0),
- The results of the Environmental Assessment for Darlington refurbishment and continued operation, and the results of the Integrated Safety Review performed at Darlington (Section 3.0),
- The Basis for Continued Operation (Section 4.0), and
- An overview of the Integrated Implementation Plan (Section 5.0).

A detailed assessment, the GAR Basis and Support Documents (Figure 1), has been prepared to expand on key topics discussed throughout this report [R-8]. Appendix D of this report provides a cross-reference table showing how the requirements of RD-360 Section 6 [R-1] are met.

Figure 1: Global Assessment Model



2.0 OVERALL PLANT PERFORMANCE

Darlington has performed well throughout its operational life and recently has been recognized as a top performing plant by industry peers. The level of performance is based on Darlington's inherent design features, its programs and processes required to operate, maintain, and manage the plant, and the organization and staff that execute these programs and processes.

This section discusses Darlington's strengths, including aspects of the plant design, programs and processes, as well as the staff and organization. Examples of ongoing safety enhancements are discussed which demonstrate OPG's drive for continuous improvement in all areas. These include lessons learned from the March 2011 earthquake and tsunami at the Fukushima Daiichi Nuclear Power Plant, strategic initiatives to improve radiation protection, and other Safety Improvements.

2.1 Design

Darlington is the most modern of the operating CANDU designs in Canada. The robust design of Darlington NGS is a significant contributor to its overall performance.

Darlington consists of 4 x 935MWe (gross power) CANDU pressurized heavy water reactors that have a combination of engineered inherent and passive safety characteristics, supported by administrative safety features. The engineered design features include:

- Group 1 and group 2 systems which are independent and physically separated and which provide similar safety functions such that failure of one system in one group can be mitigated by the operation of another system within the second group,
- Multiple cooling systems that can keep the reactor cool under all circumstances, including electrical power disruptions, using both active and passive heat removal systems, and
- Multiple barriers and protective systems that prevent releases of radioactive material into the environment.

2.2 Programs and Processes

Darlington is operated and maintained in accordance with nuclear industry codes and standards consistent with regulatory and safety requirements. Included in Darlington's operating, maintenance, engineering and other support programs and processes are many that are recognized as strengths by the industry. Important improvements to nuclear safety programs are also being developed and implemented. These include the Safe Operating Envelope (SOE) development and implementation, maintenance program implementation, and Probabilistic Risk Assessment (PRA) development.

Safe Operating Envelope Development and Implementation

The SOE is the boundary within which the station operates in order to ensure the safety of the worker, the public and the environment. The SOE at Darlington complies with Canadian Standard Association (CSA) Standard N290.15, "Requirements for the safe operating envelope of nuclear power plants" [R-9] and ensures that the limits and conditions in the safety analysis are contained in all operating documentation. The SOE provides a comprehensive foundation for operational decision-making and action to ensure the plant remains within its defined safety envelope.

Maintenance Program Implementation

The Darlington maintenance program ensures that plant equipment is repaired or replaced to provide a high degree of overall plant safety and reliability. This is evident in the historically low forced generation losses for the station. Of particular note, the preventive maintenance (PM) program ensures that safety-related equipment is maintained to prevent potential failure. The PM program was recognized in 2012 by nuclear industry peers as meeting industry best practices.

Probabilistic Risk Assessment (PRA) Development

A PRA is a comprehensive model of the probability and consequences of postulated event sequences that could lead to the release of radioactivity to the environment. It incorporates knowledge of plant design, operation, maintenance, and testing. The PRA is used to provide a systematic approach to estimating total risk and identification of potential improvements to safety.

Darlington has had a PRA since the station was designed (Darlington Probabilistic Safety Evaluation). The first Darlington A Risk Assessment (DARA) optimized for operational use was issued in April 2001. The scope of the PRA for Darlington addressed the full range of accident consequences, from the relatively benign to the most severe, including events involving multiple reactors.

Darlington applied an industry leading methodology when the DARA was updated in 2011 and is S-294 [R-10] compliant. The risk models now include events such as fires, floods, and earthquakes. Darlington also prepared a seismic PRA which is first of its kind in Canada.

The PRA demonstrates that the risk to the health of the public, living or working in the vicinity of Darlington, is very low and compare well with requirements for new plant designs.

2.3 Staff and Organization

Nuclear safety is a core value at OPG. This is reflected in the Nuclear Safety Policy [R-10] that is endorsed by OPG's board of directors. The policy places nuclear safety as the overriding priority above that of cost, schedule or production. It requires that all employees conduct themselves in a manner consistent with the behaviour of a healthy

Nuclear Safety Culture. Such conduct requires that staff always consider how their everyday activities can impact on the fundamental safety functions of the station.

In addition to the Nuclear Safety Policy [R-10], OPG has established extensive programs and procedures and employs qualified staff to safely and effectively manage the nuclear plants. The programs and training were developed based on regulatory requirements, CSA standards, International Atomic Energy Agency (IAEA) Guides, World Association of Nuclear Operators (WANO) recommendations and best nuclear industry practices from around the world. As part of continuous improvement, the programs and training are kept up to date, based on self-assessments, benchmarking, and ongoing use of industry OPEX.

Performance excellence in the safe operation, maintenance and support activities by qualified staff at Darlington was recognized by nuclear industry peers in 2012.

OPG is committed to a strong safety culture that supports worker safety, public safety, environmental safety and reactor safety. Recently, OPG became the first employer in Ontario to be awarded the Infrastructure Health and Safety Association ZeroQuest® Platinum award. This recognition was achieved following five years of sustained safety performance. A strong safety culture was also evident when OPG achieved a significant worker safety milestone of more than 12.8 million hours worked at Darlington without a lost time accident from May 2008 to March 2012.

OPG also has a strong commitment to the environment and to the community. Darlington was selected from over 100 sites across North America to receive the Corporate Habitat of the Year Award from the Wildlife Habitat Council (WHC). OPG was also the recipient of the WHC William W. Howard C.E.O. Award which recognizes a corporation that has a history of striving for excellence in biodiversity, conservation, education, and community outreach.

Recognition by external agencies and peers indicates that Darlington not only has strong organizational structure and procedures, but also a strong Nuclear Safety Culture with staff that perform their work in accordance with high safety standards.

2.4 Continuous Improvement

The continuous improvement process through which OPG strives to continually improve the safety and performance of its nuclear power plants, is longstanding, ongoing, and covers all aspects of operation. Current performance is compared to management expectations, industry standards of excellence, internal and external operating experience, and regulatory requirements to identify areas with opportunities for improvement, prepare action plans and incorporate enhancements.

Established programs and processes are used to identify and address areas for improvement. OPG participates with industry partners in developing new or revised codes and standards, in research and development activities, in the application of emerging technologies, and in the exchange of OPEX. This is done through membership in organizations such as the WANO, the CANDU Owners Group, the CSA and the Electric Power Research Institute.

The following sections describe some of the more significant areas of ongoing improvement.

2.4.1 Fukushima Operating Experience

Following the March 2011 earthquake in Japan, the safety systems at the Fukushima Daiichi Nuclear Power Plant operated as designed and the reactors were automatically shut down. However, the tsunami that followed disabled power to the critical support systems that kept the fuel cooled.

OPG acted promptly to understand what had happened at Fukushima Daiichi and confirmed that the OPG nuclear fleet remains safe for continued operation. OPG and the other Canadian reactor operators, worked to develop actions to enhance safety based on the lessons learned from the Fukushima event. These actions were approved by the CNSC and are being completed on schedule [R-12].

Darlington was designed and built to withstand earthquakes, flooding, and high winds. Darlington has procedures and equipment in place to maintain fuel cooling at all times, during and following such events. Darlington is enhancing its procedures and equipment to ensure that even in the most unlikely natural events, the nuclear fuel will continue to be cooled.

Darlington has redundant safety systems to handle both internal and external events, including very severe ones. In addition, some of the recent changes made at Darlington include enhancements to existing flood protection, the addition of portable diesel driven pumps and generators, and training of staff to use the portable pumps and generators. At the time of the Fukushima earthquake, OPG had already been working on procedures to enhance Darlington's emergency response in the event of extreme disasters including natural disasters. These are referred to as Severe Accident Management Guidelines (SAMG). Staff training was already in progress. Further enhancements to the SAMG are underway.

A mutual aid agreement is in place with all Canadian nuclear utilities to provide support in the event of an emergency. A Regional Emergency Response Support Centre is in progress to provide additional support equipment.

OPG continues to have a strong presence in international forums and with all Canadian operators of nuclear generating stations to ensure that the broad lessons learned from Fukushima are applied at OPG. In addition, OPG chairs the CANDU Industry Integration Team, formed with industry partners to resolve common issues and incorporate learning opportunities affecting the domestic and international CANDU fleet.

2.4.2 Radiation Protection Initiatives

Consistent with the policy of continuous improvement, OPG has been exploring strategic improvement initiatives which would reduce radiation dose to workers and to the public, to ensure that the dose would be As Low As Reasonably Achievable

(ALARA) during refurbishment and continued operation. These strategic initiatives include radiation source elimination, shielding, and worker protection.

2.4.3 Other Safety Improvements

OPG undertook an initiative to explore potential Safety Improvement Opportunities (SIOs) for the Darlington NGS. The SIO work is not a requirement of RD-360, but was done to identify plant system modifications which will reduce the potential for public exposure to radiation or to address important regulatory issues. The identification of potential improvements came from many sources including:

- The Darlington A Risk Assessment (DARA),
- Generic industry regulatory action items, and station specific action items,
- Reviews of OPEX, and international standards updates,
- Reviews of Technical Operability Evaluations, which are assessments of the safety of the plant after impairments to safety functions have been identified,
- Assessment of insights from advancements in new reactor designs,
- Safety margin improvements, safety analysis, operational insights, and
- Darlington station initiatives.

Those SIOs considered to provide the greatest benefit were modelled in the DARA to quantify their risk reduction benefits and then subjected to a Benefit Cost Analysis (BCA). BCA is a methodology used for evaluation of proposed alternatives to ensure that the costs of implementing a proposed course of action are commensurate with the benefits gained. They were also assessed for regulatory significance, Defence in Depth, economic benefit and operational flexibility. A multidisciplinary team of senior technically experienced staff reviewed the results. The following SIOs were approved for implementation:

- A Containment Filtered Venting System (CFVS). The purpose of the CFVS is to provide controlled and filtered emergency venting of containment to prevent over-pressurization and assure containment integrity in the unlikely event of a multi-unit Severe Accident. A Severe Accident is a Beyond Design Basis Accident [1](BDBA) that involves significant core degradation. In addition to the CFVS modification, a Shield Tank Overpressure Protection modification will be implemented. The purpose of this modification is to enhance the relief capacity of the shield tank surrounding each unit's calandria vessel to prevent shield tank failure and to limit the containment over pressurization in the unlikely event of a multi unit Severe Accident.
- Powerhouse Steam Venting System (PSVS) enhancements. These enhancements are related to duplication of the programmable controller logic of the current PSVS to improve the reliability of the PSVS that is an important system to protect plant systems following a steam line break.

[1] A BDBA is an event with a frequency of occurrence less than 1 in 100,000 reactor years.

- A third Emergency Power Generator (EPG3). The third EPG is planned to be able to withstand a seismic event which is more demanding than the Design Basis Earthquake for which the existing two EPGs are designed, and to increase emergency power reliability when one EPG is not available. A Design Basis Earthquake is a representation of the combined effects, at the site, of a set of possible earthquakes having a very small probability of being exceeded during the life of the plant.
- An alternate and independent supply of water as a component of an Emergency Heat Sink (EHS). A permanently installed alternate make-up supply to the Heat Transport System will allow for direct addition of water following BDBAs, such as station black-out leading to a total and sustained loss of heat sinks. Complementing this modification is the installation of diesel driven fire water pumps that will supply emergency water under BDBA conditions.

In addition, OPG will install two auxiliary Shutdown Cooling (SDC) pumps in each unit, separately located and of a diverse design from the existing Shutdown Cooling pumps. The new pumps will provide additional Defence in Depth and operational flexibility during reactor maintenance outages.

These SIOs were selected for inclusion in the IIP because they result in the highest overall risk reduction and Defence in Depth.

3.0 RESULTS OF THE ENVIRONMENTAL ASSESSMENT AND INTEGRATED SAFETY REVIEW

3.1 Environmental Assessment

The Darlington Environmental Assessment (EA) concluded that the refurbishment and continued operation, "taking into account the mitigating measures identified in the EA Screening Report" [R-13], is not likely to cause significant adverse environmental effects. This conclusion is contained in the CNSC's Record of Proceedings, "Environmental Assessment on the Refurbishment and Continued Operation of the Darlington Nuclear Generating Station" [R-5].

The EA was a forward-looking assessment of the potential impacts of the Darlington refurbishment and continued operation activities. In order to evaluate the environmental effects, the EA considered:

- The interactions between the environment, the individual workers, and activities that comprise the Life Extension work,
- Changes in the existing environment that may result from these interactions,
- Feasible mitigation measures to eliminate, reduce or control adverse effects, and
- The residual environmental effects that would remain after application of the mitigation measures.

Implementation of mitigation measures is part of the EA Follow-Up Program which ensures that EA conclusions continue to be valid.

The EA determined that OPG has comprehensive environmental programs and procedures in place to monitor and mitigate potential adverse conditions and to promote beneficial outcomes. Historical monitoring of environmental conditions in advance of station construction, and of effects during the more than 20 years of operations, established that environmental effects associated with Darlington have not been significant. The EA determined that most of the environmental interactions resulting from the refurbishment and continued operation were not expected to result in measurable, adverse effects and that those interactions with adverse effects would be mitigated such that the residual effects would be minor in nature.

The EA also showed a number of beneficial effects in both the local and regional communities. These include increased and sustained employment opportunities that in turn will support housing growth and property values. Darlington is one of a number of OPG facilities in the Region of Durham. OPG continues to work to maintain the collaborative, supportive relationship that has existed between OPG and members of the local community and the area municipalities.

OPG is committed to the implementation of the mitigating actions and carrying out the EA Follow-Up Program as described in the Darlington EA Screening Report [R-13] in order to ensure that significant adverse environmental effects do not occur as a result of the Darlington refurbishment and continued operation.

EA Mitigation Measures and Follow-Up Program

The EA assessed activities in the refurbishment and continued operation of Darlington NGS to identify potential environment interactions. Where an interaction would result in a negative effect to the environment, mitigation measures will be applied. OPG has committed to complete these mitigation measures to ensure there would not likely be any significant adverse environmental effects.

The purpose of the EA Follow-Up Program is to confirm that the predictions of environmental effects are accurate and that the mitigation measures are effective; and if not, that the mitigation measures are modified such that they become effective. A detailed plan to address the EA Follow-Up Program elements is published in the "Darlington Nuclear Refurbishment and Continued Operation Environmental Assessment Follow-Up Program" report [R-14], which was developed in consultation with the CNSC, stakeholders and the public.

Building on positive experience with the current ongoing Darlington monitoring programs, a key feature of the EA Follow-Up Program will be to respond appropriately to changing conditions in the environment, as determined through systematic monitoring for change and effects, based on selected performance measurement thresholds. This adaptive management approach will incorporate measurement metrics and performance thresholds suitable for the sensitivity and nature of the parameters being measured (e.g., impingement of fish in the lake intake structures).

The EA Follow-Up Program actions are tracked in the IIP and include measures to monitor environmental impacts and to mitigate Severe Accidents, specifically as described in Section 2.4.3, the installation of:

- A Containment Filtered Venting System,
- Powerhouse Steam Venting System enhancements,
- A third Emergency Power Generator, and
- An alternate and independent supply of water as an Emergency Heat Sink.

3.2 Integrated Safety Review

Periodically over the life of a nuclear generating station, plant safety is compared to the most modern licensing requirements, codes, and standards. The purpose is to ensure that the plant design, the management systems that control the design, and station operation and maintenance practices have taken into account the improvements incorporated in modern codes and standards since the licensing basis was originally established.

To satisfy the requirements in RD-360, OPG developed a basis document for the development of the ISR [R-15]. The objectives of the basis document were achieved through reviews of:

- Safety Factor Review Tasks (assessment of 17 Safety Factors)[2]. A Safety Factor Review Task represents one or more IAEA review elements or a task required to ensure that one or more CNSC safety areas and programs in RD-360 are addressed.
- Modern codes and standards (review of 103 modern codes and standards).
- Current and historic licensing issues.

The Darlington ISR [R-6] documents a systematic and comprehensive review of the design, plant condition and operation of the station and concluded that:

- There are no issues that affect safe long-term operation of Darlington.
- There are no issues that affect current safe operation.
- There are no issues that have a negative effect on the Core Damage Frequency (i.e. the sum of frequencies of all event sequences that can lead to significant core degradation).
- There is a high level of compliance with modern codes and standards.
- A number of equipment improvements related to aging management have been identified and will be addressed.

Subsequent to completion of the ISR in 2011, supplementary assessments were carried out and the results were input to the Global Assessment. These additional inputs included:

- Results of assessments of any new revisions to the Darlington ISR Basis Codes after the ISR code effective date of July 31, 2008 through to December 2012. The ISR Basis Codes are the list of codes and standards that were used to assess Darlington design and performance during the ISR.
- Results of the review of external safety significant industry OPEX, beyond that done earlier.
- Updates to Component Condition Assessments (CCA) as defined in Appendix C. The CCAs were reviewed to confirm the scope and identify any new scope that will impact Darlington refurbishment and continued operation. CCA recommendations resulting from the update of the CCAs have been identified in the IIP for further analysis. No major scope was identified from this review.
- Resolutions to CNSC comments on the 2011 ISR.
- Results of the review of Design Guide Exceptions [R-16] and the Technical Standards & Safety Authority Exemptions [R-17] during construction. These exceptions are approved deviations from specific requirements that meet the overall intent of the Design Guides or pressure boundary codes and standards.

The results from the additional assessments were subjected to the same safety significance ranking and prioritization process as that used for the original ISR, and

[2] The Security Safety Factor Report is Security Protected correspondence due to the sensitive nature of its contents and is not addressed in this report.

were documented in an ISR Addendum [R-18]. These additional reviews identified one area of deviation from the current licensing basis. A detailed review of the fire protection standard CSA N293 [R-19] identified issues relating to the original code of record. This was reported to the CNSC [R-20] and an action plan has been developed for timely resolution of the issue. Closure of the actions is being tracked in the IIP.

The ISR Gaps resulting from the reviews are clauses for which a safety requirement in a code or standard is not met, or for which the intent of the clause is not met, depending on the type of code or standard. ISR Gaps were consolidated into ISR Issues and were resolved in accordance with established OPG processes [R-21]. ISR Issues were then prioritized with respect to their importance to nuclear safety and assessed to assign a recommended resolution method in accordance with OPG's "Nuclear Refurbishment Issue Prioritization Process – Darlington" [R-22]. The outcome of this process recommended actions to address ISR Issues or the identification of Acceptable Deviations. An Acceptable Deviation is an ISR Issue with a low or very low overall safety significance that has been dispositioned as "no further action required".

The consolidated ISR results are summarized in the 14 Safety Control Areas (SCA) defined by CNSC. These SCAs are used to evaluate how well licensees meet regulatory requirements and CNSC expectations for the performance of programs. The ISR conclusions are summarized for each of the Safety Control Areas in Appendix E.

To ensure equipment reliability for the period following Life Extension, the ISR assessed the state of the physical plant and plant performance, and confirmed that the Integrated Aging Management Program (IAMP) [R-23] conforms to the current industry best practices.

In summary:

- The issues identified by the ISR review are primarily ISR Gaps against modern codes and standards and are of low safety significance.
- The reviews confirmed the existing management systems will ensure plant safety during the continued operation of the station.
- Darlington has a robust design, it is well maintained and operated, aging management and equipment condition assessment programs are in place and the aggregate risk is acceptably low for both the current operation and following refurbishment.
- Completing proposed Safety Improvement projects will continue to enhance the current strong safety performance during long term operation.

4.0 BASIS FOR CONTINUED OPERATION

As discussed in the sections which follow, the Global Assessment concludes that Darlington Nuclear Generating Station will continue to operate safely for the planned Life Extension. The results of the Global Assessment confirm that implementation of identified Safety Improvements resulting from the EA, the ISR, and OPEX, will further enhance safety of the public, workers, and the environment.

In addition to OPG's management programs and process, and its commitment to continuous improvement, the basis for continued operation takes into account:

- Adequacy of current plant design.
- Adequacy of planned Safety Improvements.
- Defence in Depth.
- Interaction of Acceptable Deviations.
- Integrated Implementation Plan (IIP) Schedule.

4.1 Adequacy of Current Plant Design

The overall safety of Darlington is assessed against the probabilistic and deterministic safety goals established by the CNSC for new nuclear power plants in Regulatory Document CNSC RD-337, "Design of New Nuclear Power Plants" [R-24]. These safety goals include risk to the public in terms of Core Damage Frequency and Large Off-Site Release Frequency. RD-337 identifies the limits of such events occurring as 1 in 100,000 and 1 in 1,000,000 per reactor year respectively [R-24]. Based on the current plant design, the Darlington A Risk Assessment (DARA) concluded that the assessed Core Damage frequencies for internal event, fire, flood and seismic events meet the new build risk limit stated in RD-337.

The seismic Large Off-Site Release Frequency (as defined in Appendix C) is estimated at 3.77 in 1,000,000 per reactor year including all seismic initiating events up to and including a low probability, high consequence earthquake. However, this estimate does not consider the SIOs committed as part of refurbishment. These SIOs are described in Section 2.4.3, i.e., the installation of:

- A Containment Filtered Venting System and Shield Tank Overpressure Protection,
- Powerhouse Steam Venting System enhancements,
- A third Emergency Power Generator, and
- An alternate and independent supply of water as an Emergency Heat Sink.

The impact of these additional safety functions on the seismic Large Release Frequency is expected to be significant, particularly with the installation of the Containment Filtered Venting System above. On this basis, the seismic risk post-refurbishment will be close to the limit for new reactor designs.

The gaps in the IIP against CNSC RD-337 related to deterministic requirements will be resolved with the implementation CNSC Regulatory Document RD-310, "Safety Analysis for Nuclear Power Plants" [R-25] and the complete validation and refinement of SAMG implementation (i.e. the remaining Phases 3 and 4 of the four phase SAMG implementation process). These have been included in the IIP.

In accordance with CNSC RD-310, events are classified based on the results of probabilistic studies and engineering judgment, into the following three classes of events:

1. Anticipated Operational Occurrences (AOOs).
2. Design Basis Accidents (DBAs).
3. Beyond Design Basis Accidents (BDBAs).

The Darlington Safety Analysis results have demonstrated that Darlington meets the CNSC Consultative Document C-6, "Requirements for the Safety Analysis of CANDU Nuclear Power Plants" [R-26], dose limits under which it is licensed. In addition, OPG has completed a review of the Darlington Safety Report events against the above classes of events and has determined that while improvements are planned to comply with CNSC RD-310, no gaps exist relating to currently identified AOOs and DBAs.

Based on the current plant design and its operation, the DARA concluded that Darlington's assessed Core Damage Frequencies for internal event, fire, flood, and seismic events meet the new build plant risk limit stated in CNSC RD-337. Following implementation of the IIP design improvements, the Darlington assessed Large Off-Site Release Frequency will approach the new build risk limit.

4.2 Adequacy of Planned Safety Improvements

The planned Safety Improvements, contained in the IIP [R-27], will resolve all open ISR Gaps (i.e. those requiring resolution), and will enhance the overall safety and reliability of the plant.

The adequacy of the Safety Improvements was evaluated both in terms of scope and schedule to confirm that the actions fully addressed the issues in an appropriate and timely manner.

4.2.1 Adequacy of ISR Open Issue Resolutions

ISR Open Issues require resolution (i.e. they have not been already closed or deemed to be an Acceptable Deviation). The adequacy assessment of the ISR Open Issues has determined that:

- The implementation of the action plan, as indicated in the IIP, will result in compliance with the requirements of the associated modern codes and standards,
- The ISR identified a number of issues requiring resolution during the review against modern codes and standards, and Safety Factor Review Tasks. The

action plan and the implementation timeframe for each Safety Improvement are included in the IIP, and

- There is no safety impact for planned improvements that are scheduled to be completed post-restart of the units.

The IIP identifies the action plan and implementation timeframe for each of the Safety Improvements [R-27].

The detailed assessment of the adequacy of ISR Open Issue resolutions is provided in the GAR support document "ISR Open Issues and Acceptable Deviations – Adequacy Review" [R-28].

4.2.2 Adequacy of Component Condition Assessment Recommendations

The CCAs reviewed for the ISR focused on safety-related components with the objective of identifying opportunities to improve the reliability of those components. The recommendations from the CCAs underwent a thorough adequacy assessment to confirm that:

- The actual condition of the safety-related equipment is known or will be confirmed.
- Equipment is defined as being in "Good" condition if it meets all its functional design requirements, with only a slight reduction in operating margins, some slight aging degradation is evident, or if the aging management practices are adequate but have not been optimized to ensure that the component remains in like new condition. The safety-significant (Reactor Safety 1 and 2) equipment identified as being in less than "Good" condition is being improved to a condition considered "Good" or better by the end of the Life Extension Window.
- Aging management practices are adequate to maintain the equipment during the extended life.

In cases where the adequacy assessment identified incremental actions, additional analysis will be conducted to determine if they meet the Life Extension requirements [R-27].

The review of the 1099 CCAs covered in the ISR confirmed that:

- 995 out of the 1099 CCAs are managed through the Integrated Aging Management Program.
- 104 out of the 1099 CCAs are identified in the IIP to restore the condition of the equipment to "Good" or, to confirm the physical condition is "Good". Associated aging management practices will be improved to maintain the equipment through continued operation.

The CCA adequacy review determined that the majority of the safety-related equipment is in "Good" or "Very Good" condition (i.e. that it meets all design requirement margins and has little evidence of aging). The assessment also found that aging management practices are adequate to ensure that this equipment will be well maintained throughout the extended life of the plant. The assessment determined that there was adequate follow up, including enhancements to equipment aging management practices, for equipment that meets design requirements but has somewhat eroded operating margins or evidence of aging. There is a recognized risk for components that will not be restored to "Good" prior the restart of the refurbishment unit. The Integrated Aging Management Program will ensure that the conditions of the components are restored to "Good" in a timely fashion based on their safety-significance, thereby mitigating this risk.

The detailed assessment of the adequacy of Component Condition Assessment recommendations is provided in the GAR support document, "Adequacy Review of ISR Condition Assessments (CCAs)" [R-29].

4.3 Defence in Depth

Defence in Depth [R-30] is a comprehensive approach to safety. The general objective of Defence in Depth is to ensure that a single equipment or human failure at one level of the five levels of defence, and even a combination of failures at one level of defence, would not jeopardize the integrity of subsequent levels.

The detailed assessment of Defence in Depth confirms that Darlington meets the Defence in Depth requirements identified in CNSC RD-337 [R-24].

The approach taken in this assessment was based on the Defence in Depth requirements identified in CNSC RD-337, with specific assessment guidance provided by the IAEA Safety Report Series No. 46, "Assessment of Defence in Depth for Nuclear Power Plants" [R-30]. The approach analyzed the five independent levels of defence. All levels of Defence in Depth rely on multiple barriers of protection to prevent or limit equipment failures or human errors and mitigate the consequences should these failures or errors occur. The intent of the review was to confirm that for each of the five levels of defence, barriers are not unnecessarily challenged, and if they are, they do not all fail. The results of the assessment for the five levels of Defence in Depth are summarized below:

Level 1 (Prevention of deviations from normal operation, and to prevent failures of SSCs)

> The first level of defence requires a high quality in the design and construction of the plant with barriers to prevent the occurrence of abnormal operating conditions. This is particularly important for the physical barriers surrounding the radioactive material in the fuel.
>
> The Defence in Depth assessment of Level 1 has reaffirmed safe, conservative operation of the plant by qualified staff and a continued focus on preventive maintenance during refurbishment and into the extended life of the plant. This ensures reliable functionality of plant equipment under normal operation and therefore prevents Anticipated Operational Occurrences (AOOs) and failures. The ISR confirmed that the engineering and maintenance programs are in place as required to ensure the integrity of the physical barriers.

Level 2 (Prevention of AOOs from escalating to accidents and to return the plant to a state of normal operation)

The second level of defence is the provision of barriers to prevent AOOs from progressing to accidents. The plant design possesses a number of strong features regarding Level 2 Defence in Depth. Digital computerized monitoring is used extensively in the design. Automatic reactor control features detect and respond to abnormal conditions before these conditions progress to the point that the next level of barriers are required to act.

A large number of safety related system tests are completed routinely to detect problems regarding plant equipment. A well-established framework of operating procedures is in place to respond to equipment malfunctions in a timely manner thereby ensuring that the plant stays within its well-defined SOE.

The assessment of the Level 2 defences concludes that barriers in place are mature and robust. With implementation of additional measures to deal with additional identified inspections and preventative maintenance, this set of barriers is expected to remain effective for the extended life of the plant. The anticipated improvements to the safety analysis, consistent with CNSC RD-310, will result in the identification and assessment of AOOs. This will provide additional insights into this Defence in Depth level.

Level 3 (Minimize the consequences of accidents)

The third level of defence is the barriers to minimize the consequences of accidents should they occur by providing inherent safety features, fail-safe design, additional equipment, and mitigating procedures. The DARA assessment shows that the overall plant design has a very low Core Damage Frequency indicating robustness in the design, and reliable equipment that is capable of responding effectively to accident scenarios.

Level 4 (Radioactive releases caused by severe plant conditions are controlled)

The fourth level of defence includes those barriers to control severe plant conditions (i.e. operator action to maintain water in the calandria such that fuel cooling is assured). Significant progress in the SAMG program implementation has resulted in Darlington strengthening its capability to respond to low probability Severe Accidents. Completion of the DARA to include external hazards has improved OPG's understanding and confidence in the adequacy of these barriers in Level 4 Defence in Depth. Implementation of lessons learned from the Fukushima event, installation of additional hydrogen mitigation equipment, and a Containment Filtered Venting System (CFVS) for BDBAs will add further capability to this Defence in Depth level.

Level 5 (Mitigation of radiological consequences)

The fifth level of defence is associated with the management and mitigation of radiological off-site consequences should an accident occur.

In the event of a nuclear plant accident, OPG is prepared with the necessary staff, equipment and procedures to support the Province in managing and mitigating off-site radiological consequences as required by the Provincial Nuclear Emergency Response Plan. As a result of the events at Fukushima, OPG has conducted a review of its emergency response program with a view to addressing lessons learned from this event and identifying further areas for enhancement. This review considered challenges that might arise from a total and sustained loss of electrical power within the station. Based on the assessment, OPG has committed to a number of actions that will enhance and further strengthen the already strong barriers for this Defence in Depth level.

In conclusion, a review of the five levels of defence performed in support of the Global Assessment has confirmed that Darlington meets the Defence in Depth requirements of CNSC RD-337 since a robust set of barriers is in place for the five levels of defence. No additional gaps were identified or improvements required beyond those being addressed in the IIP and OPG's continuous improvement program. The levels of defence – particularly Level 4 and 5 – will be further strengthened as the planned Safety Improvements are implemented. The adequacy of Darlington's Defence in Depth has been confirmed by the recent DARA update that concluded that the risk to the public is very low.

The detailed assessment of the Defence in Depth is provided in the GAR support document, "Darlington NGS Defence in Depth Assessment" [R-31].

4.4 Interaction of Acceptable Deviations

An assessment was performed to determine if ISR Issues that were not individually significant could become more significant when grouped together. The interaction assessment focused on the ISR Issues that were categorized as Acceptable Deviations (AD), which includes ISR Issues with a low or very low overall safety significance that have been dispositioned as "No further action required".

The interaction assessment used established formal techniques of symptom classification to group the ADs. Barrier analysis was then used to determine the combined negative effects within each group. This methodology was executed by subject matter experts who specialize in the design and operation of CANDU power plants.

The interaction assessment identified five safety-related groups of ADs that could have a potential combined impact, these were:

- Seismic Equipment and Programs.
- Pressure Boundary Components.
- Accident Management.
- Special Safety Systems (Shutdown Systems, Emergency Coolant Injection, Containment).
- Human Performance.

The interaction assessment determined that the interaction between the ADs within each of the five safety-related groups was minimal. The interaction assessment also concluded that the combined impact from the ADs as a whole was negligible.

The detailed assessment of the interaction of Acceptable Deviations is provided in the GAR support documents on the interaction assessment:

- "Evaluation of Seismic, Pressure Boundary and Accident Management ISR Acceptable Deviation Issues for Aggregate Effects between Issues" [R-32].
- "Evaluation of Human Performance ISR Acceptable Deviation Issues for Aggregate Effects between Issues" [R-33].
- "Evaluation of Special Safety Systems ISR Acceptable Deviation Issues for Aggregate Effects between Issues" [R-34].
- "Evaluation of Darlington NGS Fire Protection ISR Acceptable Deviation Issues for Aggregate Effects of Combinations of the Gaps" [R-35].
- "Acceptable Deviations Aggregate Review" [R-36].

4.5 Integrated Implementation Plant (IIP) Schedule

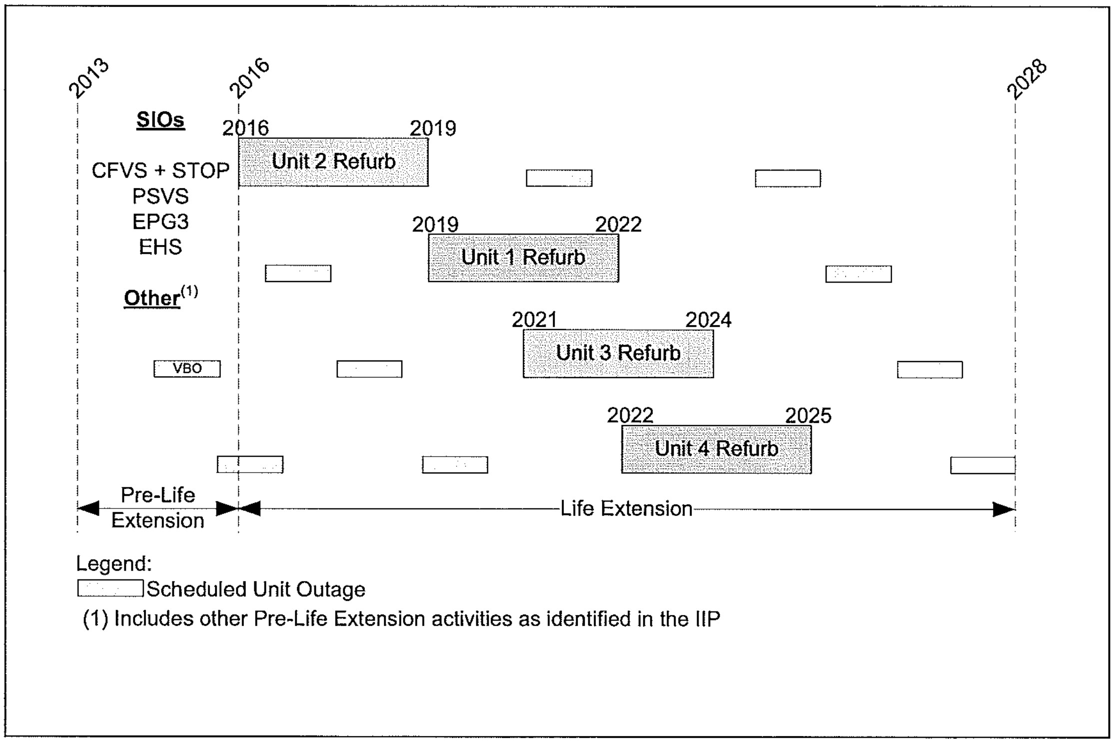
All IIP activities will be undertaken before the end of the Darlington Life Extension Window. This window is defined as the period from the beginning of the first refurbishment outage (Unit 2 Refurbishment in 2016) through the first complete maintenance outage following the last refurbishment outage. Figure 2 illustrates the Darlington Life Extension Model which includes the Life Extension Window. The dates in Figure 2 are based on current planning assumptions.

Field work which cannot be done during normal unit operation will be completed during refurbishment outages or other unit outages in the life extension window. IIP field work that can be done without the unit being shut down will be completed within the non-outage portion of the Life Extension Window.

Operational improvements identified in the IIP will be developed and implemented throughout Darlington's Life Extension Window. In some cases, such as SAMG, program improvements are already in-progress in advance of this window.

Work that does not require a unit to be in the refurbishment outage conditions (i.e. defueled or HTS/Moderator dewatered) will be executed during the Life Extension timeframe. OPG's plan is that much of the life extension work will be done prior to breaker open, particularly in the cases of the unit 3 and 4 refurbishment. Prioritization of the work is based on the impact on safety such as the EA Safety Improvement Opportunity initiatives (which will be addressed very early on) and Ageing Management criteria.

Figure 2: Darlington Life Extension Model (Indicative)



5.0 INTEGRATED IMPLEMENTATION PLAN (IIP)

The IIP was developed in accordance with RD-360 and contains the committed Safety Improvements resulting from the EA and ISR. The IIP also contains component replacements and refurbishments to support the safe long-term operation of the station. The timeframe for implementation has been established using approved OPG processes and procedures and is consistent with the safety-first culture of an industry recognized top performing station.

The major activities included in the IIP are:

- Replacement of Fuel Channels, Feeders, Calandria Tubes, and End Fittings.
- Replacement of the Primary Heat Transport Liquid Relief Valves.
- Implementation of safety-related CCA recommendations.
- Implementation of changes to Safety Analysis to comply with CNSC RD-310.
- Completion of implementation of Severe Accident Management Guidelines (SAMGs).
- Design and installation of a Containment Filtered Venting System (CFVS).
- Provision of Shield Tank overpressure protection.
- Enhancements to the Powerhouse Steam Venting System (PSVS).
- Installation of a third Emergency Power Generator (EPG3).
- Provision of an alternate and independent supply of water as an Emergency Heat Sink (EHS).
- Installation of diesel driven fire water pumps.
- Installation of two auxiliary Shutdown Cooling pumps.
- Completion of the Environmental Assessment Follow-Up Program.

The IIP is documented in a separate report, "Darlington NGS Integrated Implementation Plan (IIP)" [R-27].

6.0 CONCLUSIONS

The Global Assessment Report presents the significant results of the Environmental Assessment (EA) and the Integrated Safety Review (ISR) for Darlington Life Extension. It also reflects a thorough understanding of the condition of the plant safety-related equipment and its required maintenance, and the positive basis for continued operation.

The Global Assessment (GA) determined that Darlington Nuclear Generating Station has performed well throughout its operational life and has been recognized by its peers as a top performer. This performance is the result of a robust design, solid engineering, operations and maintenance programs and processes that incorporate continuous improvement, and staff and organization committed to nuclear safety as a core value and overriding priority.

The Darlington EA concluded that, subject to the implementation of its associated mitigating actions, the refurbishment and continued operation of Darlington NGS is not likely to cause significant adverse environmental effects and will provide a number of benefits to local and regional communities.

The ISR concluded that there were no safety issues that would limit the safe continued operation of Darlington. The issues identified are primarily gaps against the modern codes and standards related to fire protection requirements, and equipment reliability requirements to address aging of SSCs for extended life. Resolution of these gaps will be addressed under the IIP.

The GA determined that the current design, with planned Safety Improvements, will meet to the extent practicable, the regulatory design requirements of a new nuclear power plant as defined in CNSC RD-337. In assessing the adequacy of the actions included in the IIP, it was determined that the ISR Gaps would be adequately addressed resulting in enhanced safety and reliability of the plant and, in addition, the ISR Gaps that were assessed as part of the ISR as being Acceptable Deviations would collectively have very little impact on safety.

The GA included a review of the CCA recommendations and determined that the majority of the safety-related equipment in the plant is in "Good" or better condition. The aging management practices are adequate to maintain this equipment throughout the extended life of the plant and there is adequate follow up for that equipment that is not currently in "Good" or "Very Good" condition.

Finally, the GA included a detailed assessment of Darlington's Defence in Depth against the Defence in Depth requirements for new nuclear power plants (CNSC RD-337). It was determined that these requirements are not only met at Darlington but that the Defence in Depth barriers will be further strengthened as a result of the implementation of the IIP.

All aspects of RD-360 were evaluated and an adequate justification for continued operation is documented. The results demonstrate that Darlington is a safe and reliable nuclear power plant today, that there are opportunities for further improvements and that implementation of these improvements, as documented within the IIP, will result in Darlington being an even safer and more reliable supplier of clean electrical power to the Province of Ontario for another 30 years.

Appendix A: References

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[R-17] OPG Report, NK38-REP-03680-10184 R000, “MCCR / AECB Concessions and Agreements Assessments”, May 22, 2013.

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[R-20] OPG Document, Station Condition Record D-2013-07853, “ISR Fire Prot Review Discovered Code of Record Issues”, August 2, 2013.

[R-21] OPG Document, N-INS-00770-10004 R001, “Nuclear Refurbishment Gap Resolution Process – Darlington”, April 11, 2011.

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[R-23] OPG Document, N-PROG-MP-0008 R005, “Integrated Aging Management”, May 4, 2011.

[R-24] CNSC Regulatory Document, RD-337, “Design of New Nuclear Power Plants”, July 2012.

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[R-29] OPG Report, NK38-REP-03680-10200, "Adequacy Review of ISR Component Condition Assessments (CCAs)", 2013.

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Appendix B: Nomenclature

ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
CANDU	Canadian Deuterium Uranium
CCR	Code Compliance Review
CNSC	Canadian Nuclear Safety Commission
CSA	Canadian Standard Association
CFVS	Containment Filtered Venting System
DBA	Design Basis Accident
EHS	Emergency Heat Sink
EPG	Emergency Power Generator
EPG3	Third Emergency Power Generator
FHA	Fire Hazard Assessment
FSSA	Fire Safe Shutdown Assessment
IAEA	International Atomic Energy Agency
IAMP	Integrated Aging Management Program
NGS	Nuclear Generating Station
OPEX	Operating Experience
OPG	Ontario Power Generation
PIP	Periodic Inspection Plan
PM	Preventive Maintenance
PSVS	Powerhouse Steam Venting System
SAMG	Severe Accident Management Guidelines
SCA	Safety Control Area
SIO	Safety Improvement Opportunity
SPRA	Seismic Probabilistic Risk Assessment
SSC	Structures, Systems, and Components
TRF	Tritium Removal Facility
TSD	Technical Support Document
WANO	World Association of Nuclear Operators
WHC	Wild Habitat Council
MWe	Mega Watt Electrical

Appendix C: Glossary

Acceptable Deviation is an ISR Issue with a low or very low overall safety significance that has been dispositioned as 'No further action required'.

Benefit Cost Analysis (BCA) is a methodology used for evaluation of proposed alternatives for resolution of ISR Gaps to ensure that the costs of implementing a proposed course of action are commensurate with the benefits gained.

Beyond Design Basis Accidents (BDBAs) are events or combinations of events with a frequency of less than 1 per 100,000 reactor years.

Core Damage Frequency is the sum of frequencies of all event sequences that can lead to significant core degradation. The limit is less than 1 in 100,000 per reactor year.

Component Condition Assessment (CCA) provides:

1. An assessment of the current condition of Structures, Systems, and Components,
2. An assessment of component life, given the status of the current programs for inspection and maintenance, and
3. Recommendation of actions required for the SSCs to reach the target extended plant life.

Closed ISR Issue is an ISR Issue for which one of the following applies:

- The ISR Issue has been addressed by changes in processes, programs, and/or system modifications after the code review report was issued to OPG, or

- An OPG review of processes and/or documents not reviewed as part of the initial code review has demonstrated that the requirements of all the ISR Gaps that comprise the ISR issue have been met (are in compliance).

Darlington A Risk Assessment (DARA) is a Probabilistic Risk Assessment (PRA) of Darlington NGS. The DARA is used to identify the major sources of risk and assess the magnitude of radiological risk to the public from accidents due to operation of nuclear reactors while at power as well as during outage.

Defence in Depth consists of a hierarchical deployment of multiple levels of barriers between radioactive materials and the environment using equipment and procedures that ensures no single human error or equipment failure at one level of defence, nor even a combination of failures at more than one level of defence, propagates to jeopardize Defence in Depth at the subsequent level or leads to harm to the public or the environment.

Design Basis Earthquake (DBE) is an artificial representation of the combined effects, at the site, of a set of possible earthquakes having a very small probability of being exceeded during the life of the plant.

Environmental Assessment (EA) is an assessment carried out under the Canadian Environmental Assessment Act to identify whether a specific project is likely to cause significant environmental effects.

EA Follow Up Program is intended to confirm that the predictions of environmental effects are accurate and that the mitigation measures are effective, and if not, that they are modified such that they become effective.

Final ISR Report is the document that summarizes the results and major findings of all the Safety Factors, the ISR Aggregate Review and the disposition of all gaps that were identified.

Global Assessment (GA) provides an overall risk judgment on the acceptability of continued plant operation based on the significant ISR results and the EA results, and corresponding mitigation actions and improvements. The assessment of risk includes consideration of plant strengths which are areas of the refurbished plant design, condition and operation that generally meet or exceed new requirements imposed on the nuclear industry over the last 10 years, including those for new plants. The Global Assessment takes into account the Safety Improvements to address the issues identified in the EA and the ISR and the Safety Improvements resulting from identified opportunities to reduce the overall plant risk. The Global Assessment also incorporates the results of the Defence in Depth assessment.

Global Assessment Report (GAR) summarizes the results of the Global Assessment by providing a high level summary of the EA and the ISR and an overall judgment on nuclear safety.

Integrated Implementation Plan (IIP) is the integrated results of the EA and ISR, identifying all necessary Safety Improvements, committed plant modifications, safety upgrades, compensatory measures and improvements to operation and management programs that will apply to the project and to long term operation.

Integrated Safety Review (ISR) is a comprehensive assessment of an existing nuclear generating station in order to determine:

1. The extent to which the plant conforms to modern standards and practices,

2. The extent to which the licensing basis will remain valid over the proposed extended operating life,

3. The adequacy of the arrangements that are in place to maintain plant safety for long-term operation, and

4. The improvements to be implemented to resolve safety issues that have been identified.

ISR Basis Codes are the list of codes and standards that were used to assess Darlington design and performance during the ISR. They were derived using the Nuclear Refurbishment Integrated Safety Review procedure [R-15].

ISR Gap is a clause for which a safety requirement in a code or standard is not met or for which the intent of the clause is not met depending on the type of code or standard [R-15].

ISR Open Issue is an ISR Issue requiring resolution.

Large Off-Site Release Frequency is the sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{14} Becquerel of Cesium-137. The limit is less than 1 in 1,000,000 per reactor year. A greater release may require long term relocation of the local population [R-24].

Life Cycle Management Plan (LCMP) is a plan based on a process for timely detection and mitigation of aging effects in structures, systems, and components important to plant safety, reliability and economics.

Life Extension is a set of activities for extending the safe operating life of a nuclear power plant beyond its design life. It involves the replacement or refurbishment of major components (e.g. pressure tubes) or substantial modifications to the plant, or both.

Nuclear Safety Culture refers to the core values and behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment.

Probabilistic Risk Assessment is a comprehensive model of the plant incorporating knowledge about plant design, operation, maintenance, testing and response to abnormal events. It identifies the various sequences that lead to radioactivity release, and calculates their frequencies of occurrence and consequences.

Reactor Safety 1 is an Operational Safety Requirement (OSR) system that is also a System Important to Safety (SIS) whose failure results in a Total Loss of Redundancy (TLR) or System Unavailability impairment condition.

Reactor Safety 2 is an OSR system that is also a SIS whose failure results in a Partial Loss of Redundancy (PLR) impairment condition, or is an OSR system that is also a non-SIS system whose failure results in a Total Loss of Redundancy or System Unavailability impairment condition.

Review Task is a task that represents one or more IAEA review elements or a task required to ensure that CNSC safety areas and programs in RD-360 are addressed. Various review elements were translated into review tasks for purposes of performing the ISR.

Safe Operating Envelope (SOE) is the set of limits and conditions within which the nuclear generating station must be operating to ensure compliance with the safety analysis upon which reactor operation is licensed and which can be monitored by or on behalf of the operator and can be controlled by the operator.

***Safety and Control Areas* (SCAs)** are defined by the CNSC and used to evaluate how well licensees meet regulatory requirements and CNSC expectations for the performance of programs. There are 14 safety and control areas.

Safety Factor is a topic required by the CNSC for inclusion in an ISR as listed in IAEA Safety Standards Series, Safety Guide No. NS-G-2.10, "Periodic Safety Review of Nuclear Power

Plants" [R-7] or as required to ensure that CNSC safety areas and programs in the CNSC Regulatory Document RD-360 are addressed.

Safety Improvements are changes to processes or plant to address the issues identified in the EA, the ISR, and the Safety Improvements resulting from identified opportunities to reduce the overall plant risk.

Severe Accident is a Beyond Design Basis Accident (BDBA) that involves significant core degradation.

Severe Accident Management Guidelines (SAMG) refers to the set of documents that collectively describe the response to be taken by plant staff to respond to an accident that progresses beyond the basis for emergency procedures.

Appendix D: RD-360 GAR/IIP Requirements Compliance Cross-reference table

RD-360 Section	RD-360 Requirement	OPG Compliance
6.3.1 Global Assessment	The *Global Assessment Report* presents significant ISR results, including plant strengths.	***GAR*** *Section 3.2* *Appendix E*
	The *Global Assessment Report* presents the *Integrated Implementation Plan* for corrective actions and Safety Improvements.	***GAR*** *Section 5.0*
	The *Global Assessment Report* presents an overall risk judgment on the acceptability of continued plant operation.	***GAR*** *Section 4.0* *Section 6.0* ***GAR Support Documents*** *[R-8]*
	Interactions between Safety Factors, individual shortcomings, corrective actions, and Safety Improvements, including compensatory measures, should be considered in assessing the overall plant safety and the acceptability of continued operation.	***GAR*** *Section 4.4* ***GAR Support Documents*** *[R-32]* *[R-33]* *[R-34]* *[R-35]* *[R-36]* *[R-28]* *[R-29]*
	The Global Assessment should also show the extent to which the safety requirements of the Defence in Depth concept are fulfilled.	***GAR*** *Section 4.3* ***GAR Support Document*** *[R-31]*

RD-360 Section	RD-360 Requirement	OPG Compliance
6.3.2 Integrated Implementation Plan When developing the *Integrated Implementation Plant*, the licensee:	Identifies a list of shortcomings for each of the Safety Factors identified in the ISR.	***IIP*** *[R-27]*
	Identifies a list of strengths with respect to fulfilling the safety requirements of the Defence in Depth concept.	***GAR*** *Section 4.3*
	Evaluates the safety significance and ranking of each of the shortcomings and prioritizes corrective measures. Significant safety issues should be addressed immediately.	***GAR*** *Section 3.2* *Section 4.2*
	Justification for proposed exemptions from the modern standards and practices should be provided, taking the safety significance, physical practicality, and other information into account, as appropriate.	***GAR*** *Section 4.4*
	Develops corrective actions and Safety Improvements for each of the shortcomings, as far as practicable.	***IIP*** *[R-27]*
	In view of each of the ISR Safety Factors, evaluate the acceptability of plant operation over the next review period in an integrated assessment.	***GAR*** *Section 4.0*

RD-360 Section	RD-360 Requirement	OPG Compliance
6.4 Confirmation of the Adequacy of the Global Assessment Report The Commission reviews the licensee's *Global Assessment Report* for acceptability by assessing:	The completeness of the assessment.	***GAR*** *All Sections*
	The significance and ranking of identified safety issues for all assessed Safety Factors.	***GAR Support Documents*** *[R-28]*
	The adequacy for proposed corrective actions and Safety Improvements, or justification of proposed exemptions.	***GAR*** *Section 4.1* *Section 4.2* *Section 4.4* ***GAR Support Documents*** *[R-28]* *[R-29]* *[R-32]* *[R-33]* *[R-35]* *[R-36]*
	The adequacy of proposed implementation schedule.	***GAR*** *Section 4.5*
	The adequacy of the proposed measures for assuring the quality of the Life Extension activities.	***GAR Support Documents*** *[R-4]* *[R-37]*
	Conformance with the EA results.	***GAR*** *Section 3.1* ***IIP*** *[R-27]*

Appendix E: ISR Results by CNSC Safety and Control Areas

The CNSC evaluates 14 Safety and Control Areas (SCA) at each nuclear power plant to confirm that the licensees meet expectations for the provision of measures to protect health, safety and the environment and with respect to Canada's international obligations. The CNSC has assessed Darlington Nuclear Generating Station as fully satisfactory over the last five years [R-38].

This appendix reviews the purpose of each SCA and summarizes findings of the Integrated Safety Review (ISR) for the area. The attached Table 1 provides a road map of the Safety Factor Reports and codes and standards relevant to the SCAs. It also contains the list of ISR Issues associated with each SCA. An ISR Issue is a consolidation of ISR Gaps with similar scope. The list of ISR Issues is based on the status of all ISR Gaps and Issues as of October 30, 2013.

E.1.0 MANAGEMENT SYSTEM

> The management system SCA covers the framework that establishes the processes and programs required to ensure an organization achieves its safety objectives, continuously monitors its performance against these objectives, and fosters a healthy safety culture.

The Management System SCA was considered under the review of the Organization and Administration, Procedures, Human Factors and Quality Management Safety Factors [R-37]. In addition to the 29 Review Tasks associated with these Safety Factors, OPG Programs and 17 modern codes and standards related to the Management subject area were assessed.

The ISR review confirmed that OPG governance fully meets the requirements of CSA Standard N286-05, "Management System Requirements for Nuclear Power Plants" [R-39].

The assessment of the ISR Review Tasks confirmed that OPG has a comprehensive set of governance that establishes the overall management system which provides assurance that systems and equipment in use at OPG are designed, procured, fabricated, installed, operated and maintained in accordance with the applicable regulations, standards and that activities conducted by OPG are of the desired quality.

The Management System uses rigorous processes to ensure all work activities are planned and controlled to maintain the plant configuration and condition within the design and licensing bases. Changes that could impact on nuclear safety are assessed and implemented in a controlled manner.

No ISR Gaps related to the Review Tasks were identified. A review of 17 modern codes and standards also identified no ISR Gaps related to this SCA.

The CNSC has consistently assessed the management system SCA as satisfactory over the last five years [R-38]. Given the processes in place within OPG, it is expected that the management system SCA will continue to meet regulatory requirements and expectations during continued operation.

E.2.0 HUMAN PERFORMANCE MANAGEMENT

The human performance management SCA covers activities that enable effective human performance through the development and implementation of processes that ensure that licensees have sufficient staff in all relevant job areas with the necessary knowledge, skills, procedures and tools in place to safely carry out their duties.

Human Performance was considered under the review of the Safety Factors for Organization and Administration, Procedures, and Human Factors [R-37].

The assessment of Review Tasks related to this SCA confirmed that OPG has a comprehensive set of governance, training programs and facilities that ensure the safe and reliable operation and maintenance of the plant. There are procedures to ensure a minimum number of qualified staff, appropriate to the operating state of the plant and emergencies are in place at all times. There are fitness for duty guidelines relating to hours of work, health and substance abuse. Also, the procedures provide an appropriate level of detailed guidance.

The assessment of the Review Tasks did not identify ISR Gaps.

A review of 7 modern codes and standards identified ISR Gaps related to this SCA that were consolidated into 5 ISR Issues. The ISR Issues were determined to be not safety significant and classified, in accordance with approved procedures, as Acceptable Deviations. No further actions are required.

The CNSC has consistently assessed the Human Performance SCA as satisfactory over the last five years [R-38]. Given the processes in place within OPG, it is expected that the Human Performance SCA will continue to meet regulatory requirements and expectations during continued operation.

E.3.0 OPERATING PERFORMANCE

The operating performance SCA includes an overall review of the conduct of the licensed activities and the activities that enable effective performance.

The operating performance SCA was considered under the review of the Safety Performance [R-50], Use of Experience from Other Plants and of Research Findings [R-40], Organization and Administration, and Procedures Safety Factors [R-37].

The assessment of the Organization and Administration, Procedure, and Safety Performance Safety Factors confirmed the safety performance expectations are achieved through a variety of monitoring techniques that ensure the organization is alert to detect and respond to indicators that signal declining performance. It also confirmed that, in order to improve performance, a set of key performance indicators is established, measured, and trended. Identified adverse trends are analyzed forming the basis for development of corrective actions designed to improve plant and equipment performance, operating requirements and practices, and overall human performance.

The "Use of Experience from Other Plants and of Research Findings" Safety Factor report concluded that OPG governance meets the requirements of the relevant Review Tasks. The OPEX program in use at Darlington meets the overall intent of identifying, classifying and delivering both internal and external OPEX to the relevant departments. OPG has adequate processes in place to utilize the feedback of safety experience from other nuclear power plants and the findings of research.

A review of 12 modern codes and standards identified one ISR Issue related to this SCA which was determined to be not safety significant. It was classified as an Acceptable Deviation and no further action is required.

The CNSC has consistently assessed the Operating Performance SCA as fully satisfactory over the last five years [R-38]. Given the processes in place within OPG, it is expected that the Operating Performance SCA will continue to meet regulatory requirements and expectations during continued operation.

E.4.0 SAFETY ANALYSIS

> The safety analysis SCA includes maintenance of the safety analysis that supports the overall safety case for the facility. Safety analysis is a systematic evaluation of the potential hazards associated with the conduct of a proposed activity or facility and considers the effectiveness of preventive measures and strategies in reducing the effects of such hazards.

The safety analysis SCA was considered under the Safety Factors for deterministic safety analysis, probabilistic safety analysis and hazard analysis [R-41].

The objectives of the safety analyses Safety Factor were met. Overall, the safety analysis related programs and procedures at OPG are comprehensive, resulting in a systematic and disciplined approach to identifying, prioritizing and addressing safety analysis issues. The reviews confirmed that analyses documented in the Safety Report or stand-alone assessments demonstrate that the safety goals defined in the draft CNSC Consultative Document C-6 under which Darlington was licensed are met.

A review of 26 modern codes and standards identified ISR Gaps related to this SCA arising from new scope and methodology for safety analysis. The ISR Gaps are addressed as follows:

- OPG has completed and submitted to the CNSC an updated Probabilistic Risk Assessment (PRA) and thereby remedied all of the ISR Gaps related to the scope and completion of the PRA.
- OPG has completed a Fire Hazard Assessment (FHA) and a Fire Safe Shutdown Assessment (FSSA) and submitted it to the CNSC. Once the CNSC has accepted these documents, all ISR Gaps related to these assessments will be included in the IIP.
- Ongoing projects at the industry level via the CANDU Owners Group as well as OPG's own initiatives aim to become fully compliance with the requirements defined in the CNSC Regulatory Document RD-310, "Safety Analysis for Nuclear Power Plants" [R-25].

In addition, there was one minor ISR Gap requiring a documentation update and another ISR Gap will be closed once the defueling assessments have been completed.

The CNSC has consistently assessed the safety analysis SCA as satisfactory over the last five years [R-38]. Given the processes in place within OPG, it is expected that the safety analysis SCA will continue to meet regulatory requirements and expectations during continued operation.

E.5.0 PHYSICAL DESIGN

The physical design SCA relates to activities that affect the ability of structures, systems and components to meet and maintain their design basis, given new information arising over time and taking changes in the external environment into account.

The physical design SCA was covered under the Plant Design [R-42], Equipment Qualification [R-52], and elements of the Aging and Actual Condition of SSCs [R-45] Safety Factors. The majority of codes and Review Tasks for the "Aging and Actual Condition of SSCs" Safety Factors are relevant to the fitness for service SCA so these aspects are discussed in Section E.6.0.

A total of 49 modern codes and standards related to this SCA were reviewed.

The review of the Plant Design Safety Factor verified that the Darlington plant design function was well established, documented and implemented during the original design. The plant design function is being adequately maintained and has the capability to maintain continued design integrity over the post-refurbishment period given that all identified ISR Gaps are resolved. The majority of ISR Issues identified during reviews against modern codes and standards under plant design are related to fire protection.

Fire Protection

Five modern codes and standards related to fire protection (design) were reviewed. Gaps were mainly related to the design clauses that are applicable to new designs. In addition, some issues were identified relating to the original code of record. This was reported to the CNSC [R-20] and an action plan was developed for timely resolution of the issue. Closure of the actions is being tracked in the IIP.

ISR Gaps and related ISR Issues will be closed through minor modifications to the station, programmatic and operational changes and additional analysis to better define the Life Extension scope. The additional analysis may result in further modifications depending on its outcome.

One of the more important enhancements which will be executed is the addition of diesel fire pumps to the fire water system, thereby improving the Emergency Service Water System that will supply emergency water under BDBA conditions.

Implementation of the fire protection program to meet modern codes and standards ensures the plant is further protected from fire and fire damage is minimized.

Human Factors

The ISR confirmed that the designers explicitly considered the human-machine interface and used design reviews and mock-ups to verify and validate the design of Darlington. Since the original design, human factors standards have been issued and OPG's design practices are consistent with these standards.

The ISR Gaps identified primarily deal with annunciation system requirements. Completion of identified improvements to the Darlington annunciation system will bring the station into a high level of compliance with the modern codes and standards.

Some ISR Gaps were raised regarding implementation of human factors engineering and ergonomic standards in the layout of main control room controls, instrumentation, displays and meeting requirements of modern codes and standards. The ISR Gaps identified are of low safety significance and are either already resolved or in the IIP scope.

Pressure Boundary

A review of 6 pressure boundary related codes identified two ISR Gaps. Completion of additional analysis on the fuelling machine is required to verify that reinforcement of threaded connections meet ASME Boiler and Pressure Vessel code and a minor revision to a Periodic Inspection Program is required to expand the scope of support inspections to include supplementary supporting steel structures.

Equipment Qualification

The Equipment Qualification Safety Factor concluded that the programs and processes which maintain equipment qualification for the current operating life of the

Darlington have been adequately established and are assessed as effective. These programs and associated processes are evaluated for effectiveness on an ongoing basis to ensure that compliance will be maintained throughout the post-refurbishment period.

Environmental Qualification

The small number of ISR Gaps identified in the review of CSA Standard N290.13 [R-43] has been closed through completion of the Environmental Qualification Project. This was acknowledged by the CNSC [R-44].

Seismic Qualification

Gaps to modern requirements were primarily associated with changes in the seismic analysis tools and techniques, definition of the Design Basis Earthquake and use of historical seismicity to determine the ground motion for seismic qualification. Most ISR Gaps are addressed with the issuance of the Darlington Seismic Probabilistic Risk Assessment (SPRA) and with the issuance of an updated Seismic Design Guide. Issuance of a consolidated seismically-qualified equipment list will close the remaining ISR Gap.

The Darlington SPRA identified that no specific mitigating measures or modifications are required. The SPRA has confirmed that Darlington is robust and well suited to withstand the seismic hazard. Notwithstanding this conclusion, there are planned modifications to install a third Emergency Power Generator (EPG3) and Containment Filtered Venting System (CFVS) which are expected to further reduce station risk due to a seismic event and other low probability events.

Other ISR Gaps in this SCA will be closed with the installation of additional hydrogen mitigation equipment, analysis of anchorage configurations and concrete to verify code requirements, and minor changes to an OPG governing document.

The CNSC has consistently assessed the physical design SCA as satisfactory over the last five years [R-38]. Given the processes in place within OPG, it is expected that the physical design SCA will continue to meet regulatory requirements and expectations during continued operation.

E.6.0 FITNESS FOR SERVICE

The fitness for service SCA covers activities that affect the physical condition of structures, systems and components to ensure that they remain effective over time. This includes programs that ensure all equipment is available to perform its intended design function when called upon to do so.

The fitness for service SCA was covered under the Plant Design [R-42] and Environmental Qualification [R-52], Safety Analysis [R-41], Aging and Actual Condition of SSCs [R-45] Safety Factors. The conclusions of the review of the first two Safety

Factors are provided in Sections E.4.0 and E.5.0. In all, 17 modern codes and standards related to this SCA were reviewed.

OPG manages the aging and obsolescence of SSCs through the Integrated Aging Management Program (IAMP). The IAMP is consistent with best industry practices and ensures the safe long term operation of the station.

The IAMP establishes an integrated set of activities that ensure:

- The long term health of SSCs is assured,
- The high operational reliability of equipment, and
- The safety and operating margins are monitored and maintained.

To support the station equipment reliability program, a comprehensive assessment of the plant condition is documented in a series of Component Condition Assessments. The CCAs for ISR systems provided an input to the scope of the refurbishment outage. They identified a number of recommendations for incremental work to improve component condition or aging management practices for continued operation of the station. The recommendations are included in the IIP.

A review of the related modern codes and standards found gaps, most of which are related to inspection and testing. These ISR Gaps will be resolved via the update to various documents and pre-defined maintenance, including the updating and CNSC acceptance of the Periodic Inspection Programs (PIPs) for Concrete Containment Structures to ensure compliance with CSA N287.7-08, "In-service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants" [R-46]. In addition, a small set of ISR Gaps will be addressed through analysis that may result in recommendations for additional work which will be added to the IIP.

In summary, Darlington is confirmed to have good aging management programs in place to manage aging of SSCs for Life Extension thereby ensuring equipment reliability and availability for long term safe station operation.

Major Components

As part of the overall process for managing aging and obsolescence of SSCs, OPG has set up a focused program to address the following four major components:

1. Feeders,
2. Fuel channels,
3. Reactor components and structures, and
4. Steam generators.

As part of this dedicated program, all four major components have a Life Cycle Management Plan (LCMP) which provides a detailed vulnerability assessment considering degradation mechanisms, mitigation strategies, impacts and confidence levels for operation to end of life. These LCMPs are based on inspection results and industry OPEX. The review to determine activities to restore or sustain component condition through refurbishment has concluded that the feeders, fuel channels, calandria tubes and end fittings are life-limiting and will be replaced during refurbishment.

To assure their long term safe operation, internal inspections of the calandria, lattice tubes and guide tube locators will also be performed.

It has been determined that there is no technical reason or economic justification for replacing the steam generators during the refurbishment outage. However, the planned periodic review will identify any necessary future actions required to ensure that steam generators continue to meet design requirements and that they are operated and maintained in a reliable and cost effective manner for the life of the station.

Tritium Removal Facility (TRF)

The TRF has its own life-cycle plan to address aging and obsolescence of the TRF SSCs. The assessment is being completed in a phased approach using CCAs. Phase 1 which covers operation to end of life in 2025 has been completed and the results are used to define the scope of a program that ensures reliable operation of the TRF to the end of life in 2025. Phase 2 of the CCAs was to identify any work needed to extend the TRF to the end of 2055. This work has been completed and is being used in Phase 3 where an integrated system assessment will be performed. Phase 3 will refine the Life Extension scope and cost, review alternate options and recommend any additional work. A management decision concerning future work required for the TRF will be made by the end of 2017.

The CNSC has assessed the fitness for service SCA as satisfactory or fully satisfactory for the last five years and fully satisfactory for the three years prior since 2010 [R-38]. Given the processes in place within OPG, it is expected that the fitness for service SCA will continue to meet regulatory requirements and expectations during continued operation.

E.7.0 RADIATION PROTECTION

The radiation protection SCA covers the implementation of a radiation protection program in accordance with the Radiation Protection Regulations. This program must ensure that contamination and radiation doses received are monitored and controlled.

The ISR addressed specific aspects of the radiation protection SCA in the review of multiple Safety Factors and in the review of 13 modern codes and standards. Roadmaps to the relevant discussions in the Safety Factor Reports and to relevant sections of the codes and standards were provided in Section 3.9 of the Final ISR Report [R-6].

The ISR demonstrated that OPG has adequate physical and administrative provisions in place to ensure radiation protection requirements are met.

A review of modern codes and standards identified several ISR Gaps related to the radiation protection SCA which were consolidated into ISR Issues. Two of the three ISR Issues requiring actions will be closed with an update to radiation protection training and training documents, and enhanced instructions on storage of radioactive material. The third action requires an assessment of options to improve inter-zonal ventilation flow. The IIP will be updated to incorporate the conclusions of the inter-zonal ventilation flow assessment.

The CNSC has assessed the radiation protection SCA as satisfactory or better over the last five years [R-38]. Given the processes in place within OPG, it is expected that the Radiation Protection SCA will continue to meet regulatory requirements and expectations during continued operation.

E.8.0 CONVENTIONAL HEALTH AND SAFETY

The conventional health and safety SCA covers the implementation of a program to manage workplace safety hazards and to protect personnel and equipment.

The conventional health and safety SCA is not within the scope of the ISR. It has been addressed as part of the ongoing licensing process. The CNSC has consistently assessed the conventional health and safety SCA as fully satisfactory over the last five years [R-38]. Given the processes in place within OPG, it is expected that the conventional health and safety SCA will continue to meet regulatory requirements and expectations during continued operation.

E.9.0 ENVIRONMENTAL PROTECTION

The environmental protection SCA covers programs that identify, control and monitor all releases of radioactive and hazardous substances and effects on the environment from facilities or as the result of licensed activities.

The ISR review of the Environment Safety Factor [R-47] confirmed that the Darlington operating organization has an adequate program for the surveillance of the radiological and non-radiological impacts of the station on the environment.

A review of 11 modern codes and standards related to this SCA identified several ISR Gaps against one code (IAEA NS-G-3.2, "Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants" [R-48]). One set of ISR Gaps was closed when the Geological and Hydrogeological Environmental Technical Support Document (TSD) was issued for the Darlington EA. This TSD addressed the geology and hydrogeology of the site, including the enhanced groundwater model. The remaining ISR Gaps were not considered to be significant. No further actions are required based on the review of modern codes and standards.

The CNSC has consistently assessed the environmental protection SCA as satisfactory over the last five years [R-38]. Given the processes in place within OPG, it is expected that the environmental protection SCA will continue to meet regulatory requirements and expectations during continued operation.

E.10.0 EMERGENCY MANAGEMENT AND FIRE PROTECTION

The emergency management and fire protection SCA covers emergency plans and emergency preparedness programs that exist for emergencies and for non-routine conditions including any results of exercise participation. This also includes conventional emergency and fire response. This SCA includes the fire response rating while fire protection operations, design and analysis are discussed and rated in the appropriate SCA of operating performance, safety analysis or physical design.

The emergency management and fire protection SCA was addressed in the ISR review of the Emergency Planning [R-49] and the Hazard Analysis [R-41] Safety Factors. A discussion of the results of the Hazard Analysis Safety Factor was provided in Section E.4.0 for the Safety Analysis SCA.

The review of the Emergency Planning Safety Factor (which includes fire protection) confirmed that the Darlington operating organization has adequate plans, staff, facilities and equipment to respond to emergencies and that the operating organization's arrangements are adequately co-ordinated with local, provincial, and national systems and are regularly exercised.

Emergency Management

The review confirmed that OPG's governance continues to be in compliance with Provincial requirements.

The review of 5 modern codes and standards identified a total of 6 ISR Issues related to the emergency management SCA of which 3 were not safety significant and classified as Acceptable Deviations. No further actions were required. The remaining 3 ISR Issues are related to documentation improvements to better align with the Provincial Nuclear Emergency Response Plan. These issues have been resolved.

Fire Protection

A review of 5 modern codes and standards identified a total of 3 ISR issues related to the Fire Protection SCA of which one was not safety significant and classified as an Acceptable Deviation. No further action was required. The remaining 2 ISR Issues will be addressed through updates to the Fire Safety Plan and to the training program for extinguishment procedures of flammable and combustible liquids for all employees involved in the fuel oil transfer system or in the fuel oil tank refilling process.

Fire protection was also discussed in the Safety Analysis, Physical Design, Fitness for Service and Radiation Protection SCA assessments for elements other than emergency management (see Sections E.4.0 – E.7.0).

The CNSC has assessed the emergency management and fire protection SCA as satisfactory or fully satisfactory over the last five years [R-38]. Given the processes in place within OPG, it is expected that the emergency management and fire protection SCA will continue to meet regulatory requirements and expectations during continued operation.

E.11.0 WASTE MANAGEMENT

The waste management SCA covers internal waste-related programs that form part of the facility's operations up to the point where the waste is removed from the facility to a separate waste management facility. This also covers the planning for decommissioning.

Nuclear waste management aspects were covered in the ISR Safety Performance [R-50], Plant Design [R-42], Emergency Planning [R-49] and Environment [R-47] Safety Factors. These assessments confirmed that the Darlington operating organization has nuclear waste management programs and procedures which control the handling, storage and disposal of radioactive waste. Operating and maintenance practices focus on limiting the production of nuclear waste and facilitating its handling, storage and disposal.

A review of three modern codes and standards identified no ISR Gaps related to the nuclear waste management portion of this SCA.

The conventional waste management and decommissioning portion of this SCA are not within the scope of the ISR. It has been addressed as part of the ongoing licensing process.

The CNSC has consistently assessed the waste management SCA as satisfactory over the last three years [R-38]. This SCA was not rated prior to that time. Given the processes in place within OPG, it is expected that the waste management SCA will continue to meet regulatory requirements and expectations during continued operation.

E.12.0 SECURITY

The security SCA covers the programs required to implement and support the security requirements stipulated in the regulations, in their licence, in orders, or in expectations for their facility or activity.

The Security SCA was considered under the review of the Security Safety Factor Report. Due to the confidentiality of the subject, the results of the review are not provided in this report.

The CNSC has consistently assessed the security SCA as satisfactory over the last five years [R-38]. Given the processes in place within OPG, it is expected that the security SCA will continue to meet regulatory requirements and expectations during continued operation.

E.13.0 SAFEGUARDS AND NON-PROLIFERATION

The safeguards and non-proliferation SCA covers the programs and activities required for the successful implementation of the obligations arising from the Canada/IAEA safeguards agreements as well as all other measures arising from the Treaty on the Non-Proliferation of Nuclear Weapons.

The Safeguards SCA was covered under the assessment of the ISR Safeguards Safety Factor [R-51]. The assessment concluded that the Darlington NGS operating organization complies with the regulatory requirements applicable to safeguards as prescribed by the CNSC.

A review of seven modern codes and standards related to this SCA identified no ISR Gaps.

The CNSC has assessed the safeguards SCA as “satisfactory” or “fully satisfactory” over the last five years [R-38]. Given the processes in place within OPG, it is expected that the safeguards SCA will continue to meet regulatory requirements and expectations during continued operation.

E.14.0 PACKAGING AND TRANSPORT

The packaging and transport SCA covers the safe packaging and transport of nuclear substances and radiation devices to and from the licensed facility.

The packaging and transport SCA is not within the scope of the ISR. It has been addressed as part of the ongoing licensing process.

The CNSC has assessed the packaging and transport SCA as satisfactory over the last three years [R-38]. This SCA was not rated prior to that. Given the processes in place within OPG, it is expected that the packaging and transport SCA will continue to meet regulatory requirements and expectations during continued operation.

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Table 1: CNSC Safety and Control Area Summary and Cross-references

The Codes and Standards listed below are defined and referenced in reference [R-15].

Safety & Control Area	Information in the Darlington ISR	Relevant Codes Reviewed in ISR	Relevant ISR Issues Requiring Action[3]

Management System	Safety Factor Reports:
	• Management
	  o Organization and Administration Safety Factor
	  o Human Factors Safety Factor
	  o Quality Management Safety Factor
		• CNSC S-99
		• CNSC S-210
		• CNSC G-129
		• CNSC G-228
		• CNSC G-323
		• SOR/2000-202
		• SOR/2000-203
		• CSA N286
		• CSA N286.7
		• IAEA GS-R-3
		• IAEA NS-G-2.11
		• IAEA NS-G-2.3
		• IAEA NS-G-2.4
		• IAEA NS-G-2.6
		• IAEA NS-G-2.8
		• IAEA NS-R-2
		• IAEA SSR-2/2
			No ISR Issues requiring action have been identified for the Management System Safety and Control Area

Human Performance Management	Safety Factor Reports:
	• Management
	  o Organization and Administration Safety Factor
	  o Procedures Safety Factor
	  o Human Factors Safety Factor
		• CNSC G-323
		• CNSC RD-204
		• CSA N286
		• IAEA NS-G-2.4
		• IAEA NS-G-2.8
		• IAEA NS-R-2
		• IAEA SSR-2/2
			No ISR Issues requiring action have been identified for the Human Performance Management Safety and Control Area

[3] Note that Codes and Standards may be applicable to multiple Safety Factors/SCAs. Each ISR Issue was allocated to the most applicable SCA.

Safety & Control Area	Information in the Darlington ISR	Relevant Codes Reviewed in ISR	Relevant ISR Issues Requiring Action[3]

Operating Performance	Safety Factor Reports:
	• Safety Performance
	• Use of Experience from Other Plants and Research Findings
	• Management
	  o Organization and Administration Safety Factor
	  o Procedures Safety Factor
		• CNSC S-99
		• CNSC S-210
		• CNSC G-228
		• SOR/2000-202
		• SOR/2000-203
		• CSA N286
		• IAEA GS-R-3
		• IAEA NS-G-2.2
		• IAEA NS-G-2.4
		• IAEA NS-G-2.11
		• IAEA NS-R-2
		• IAEA SSR-2/2
			No ISR Issues requiring action have been identified for the Operating Performance Safety and Control Area

Safety Analysis	Safety Factor Reports:
	• Safety Analysis
	  o Deterministic Safety Analysis Safety Factor
	  o Probabilistic Safety Assessment Safety Factor
	  o Hazard Analysis Safety Factor
		• CNSC R-7
		• CNSC R-8
		• CNSC R-9
		• CNSC R-10
		• CNSC R-77
		• CNSC S-98
		• CNSC RD/GD-98
		• CNSC S-294
		• CNSC G-144
		• CNSC G-149
		• CNSC G-306
		• CNSC RD-310
		• CSA N286
		• CSA N286.7
		• CSA N290.13
		• CSA N288.2
		• CSA N289.1
		• CSA N290.1
			• D027 - Severe Accident and Beyond Design Basis Accident (BDBA) Analysis/ SAMG
			• D028 - Systematic Analysis of Anticipated Operational Occurrences (AOOs)
			• D030 - Identification and Classification of Events per CNSC RD-310
			• D068 - Severe Accident and Beyond Design Basis Accident (BDBA) Design/ SAMG
			• D068 - Severe Accident and Beyond Design Basis Accident (BDBA) Design/ SAMG
			• D143 - Severe Accident and Beyond Design Basis Accident (BDBA) Program/ SAMG
			• D301 - Potential Impacts From Channel Defueling
			• D399 - Acceptance Criteria for Anticipated Operational Occurrences
			• D400 - Deterministic Safety Analysis Uncertainties
			• D424 - Anticipated Operational Occurrences (AOOs)
			• D425 - No Best Estimate Analysis of Operational Events

Safety & Control Area	Information in the Darlington ISR	Relevant Codes Reviewed in ISR	Relevant ISR Issues Requiring Action[3]
		• CSA N290.4 • CSA N290.5 • CSA N293 • IAEA NS-G-1.2 • IAEA SSG-2 • IAEA NS-G-3.3 • IAEA SSG-9 • IAEA NS-R-1	• D426 - Source Term Analysis Not Complete for BDBE (beyond design basis events) • D430 - CNSC Acceptance of FHA and FSSA • D431 - CNSC Acceptance of FHA • D441 - CNSC Acceptance of CCR, FHA and FSSA
Physical Design	Safety Factor Reports: • Plant Design • Aging & Actual Condition of Systems, Structures, and Components • Equipment Qualification	• ASME B&PV Code, Section III • ASME B&PV Code, Section VIII • ASME B31.1 • ASME N509 • ASME N510 • ANSI NIRMA CM 1.0 • CNSC R-7 • CNSC R-8 • CNSC R-9 • CNSC R-10 • CNSC R-77 • CNSC RD-337 • CNSC G-276 • CNSC G-278 • CSA B51 • CSA N285.0 • CSA N285.2 • CSA N285.3 • CSA N285.6 • CSA N286	***Various*** • D013 - Long Term Control of Hydrogen in Containment • D303 - Extension of the Containment Envelope Requirements • D304 - Change to Extension of the Containment Envelope • D321 - Threaded Connections • D328 - Post Accident Monitoring – Configuration Management • D332 - Reactor Control System Requirements for AOOs • D412 - Predicted Failure mode of Anchorage Systems • D413 - Concrete Cover for Reinforcement • D344 - Self-Vented Pressure Regulating Valves • D355 – Library Functions • D356 - Compliance with ASME Boiler Pressure Vessel Code (BPVC), Section III, NF • D500 - Adequacy of the N289.3-M81 Code Review Report

Safety & Control Area	Information in the Darlington ISR	Relevant Codes Reviewed in ISR	Relevant ISR Issues Requiring Action[3]
		• CSA N286.7 • CSA N287.1 • CSA N287.2 • CSA N287.3 • CSA N287.5 • CSA N289.1 • CSA N289.2 • CSA N289.3 • CSA N289.4 • CSA N289.5 • CSA N290.1 • CSA N290.13 • CSA N290.4 • CSA N290.5 • CSA N290.6 • CSA N291 • CSA N293 • National Building Code of Canada • National Fire Code of Canada • NFPA-20 • NFPA-24 • IAEA NS-G-3.3 • IAEA SSG-9 • IAEA NS-R-1 • IAEA NS-R-2 • IAEA SSR-2/2 • IAEA NS-R-3 • NUREG-0700 • NUREG-0711	***Seismic Qualification*** • D345 - Consolidated Seismically Qualified Equipment List • D352 - Time History Compatibility with the Design Ground Response Spectrum ***Human Factors*** • D260 - Human Factors - Annunciation Improvements ***Fire Protection*** • D044 - Fire Alarm Systems • D045 - Fire Suppression • D046 - Fire Protection Seismic Requirements • D047 - Material Storage • D048 - Fire Protection Requirements for Storage Tanks • D059 - Lightning Protection • D080 - Fire Separation • D115 - Fire Protection Requirements for Laboratories • D116 - Fire Protection Requirements for Building Materials • D141 - Fire Protection Requirements for Indoor Fuel Oil Systems • D215 - Fire Department Connections to Fire Mains • D225 - Fire Protection Water Supply • D226 - Control Room Complex Fire Protection • D227 - Fire Hydrant Requirements • D251 - Fire Protection Requirements for Air-Cleaning Units • D297 - Fire Protection Air Filter Media Requirements • D428 - Detection of Significant Fire Hazards

Safety & Control Area	Information in the Darlington ISR	Relevant Codes Reviewed in ISR	Relevant ISR Issues Requiring Action[3]
			• D429 - Fire Separations Corrective Actions • D432 - Canadian Electrical Code Review for Changes Impacting Fire Protection • D436 - Emergency Lighting in Airlocks and Transfer Chambers • D439 - Fire Protection of Charcoal Filters • D442 - Fire Endurance of Fire Alarm Cable • D444 - Fire Stopping • D445 - Combustible Insulation • D446 - Combustible Material In Ducts • D448 - Fire Dampers • D452 - Spatial Separation and Exposed Building Face • D459 - Emergency Lighting in the 020 TRF • D460 - Fuel Supply Shut Off Valves • D461 - Means of Egress • D464 - Signage Requirements • D466 - Elevator Hoistway and Machine Room Requirements • D467 - Vertical Service Shafts • D468 - Pipe Insulation Requirements • D472 - Oil Storage Tank and Piping Requirements • D473 - Documentation • D475 - Valves Controlling Water Supplies • D476 - Underground Pipe • D477 - Size of Bypass • D479 - Fire Pump Disconnecting Means • D482 - Monitoring of Fire Pump Alternate Power Source • D484 - Magnetic Locks
Fitness for	Safety Factor Reports:	• CNSC S-98	• D011 - Changes to In-service Examination and

Safety & Control Area	Information in the Darlington ISR	Relevant Codes Reviewed in ISR	Relevant ISR Issues Requiring Action[3]
Service	• Plant Design • Aging & Actual Condition of Systems, Structures, and Components • Equipment Qualification • Safety Analysis	• CNSC RD/GD-98 • CNSC S-210 • CSA N285.4 • CSA N285.5 • CSA N285.8 • CSA N286 • CSA N287.5 • CSA N287.7 • CSA N290.13 • CSA N293 • IAEA NS-G-2.6 • IAEA NS-G-2.12 • IAEA NS-R-2 • IAEA SSR-2/2 • IAEA Safety Report Series 15 • National Fire Code of Canada	Testing Requirements for Concrete Containment Structures • D119 - Storage Tank Leak Detection • D167 - Ventilation Systems Disconnect Switch Testing • D182 - Thermal Insulating Materials • D184 - Fire Protection Program Audit • D247 - In-service Testing of Air Treatment Systems • D300 - Inspection Requirements for Safety-Related Structures • D346 - Environmental Qualification of Equipment for Beyond Design Basis Accident (BDBA) Analysis • D370 - Qualification of Inspection Procedures and Demonstration of their Effectiveness • D397 - Time Limited Aging Analysis for Civil Structures and Components • D398 - Transient/Fatigue Monitoring Program • D416 - N285.4 PIP Governance References N285.4-05 not N285.4-09 UPD2 • D420 - New Erosion and Corrosion Inspection Requirements in N285.4-09 UPD2 not reflected in Current PIP Governance. • D421 - Extended Life Inspection Schedules in N285.4-09 UPD2 are not Reflected in PIP Governance • D422 - Assessment of Prior Operating Non-Conforming State is required when Dispositioning Inspection Results • D423 - Governance does not Ensure that Qualifications of Examination Personnel are Included Within Inspection Reports

Safety & Control Area	Information in the Darlington ISR	Relevant Codes Reviewed in ISR	Relevant ISR Issues Requiring Action[3]
			• D469 - Inspection, Testing and Maintenance Requirements • D501 - Aging and Actual Conditions of SSCs SFR - CNSC Type II Inspection of CCAs
Radiation Protection	Safety Factor Reports • Safety Performance Safety Factor • Plant Design Safety Factor • Safeguards Safety Factor • Safety Performance Safety Factor • Aging and Actual Condition of Systems, Structures, and Components Safety Factor • Environment Safety Factor • Equipment Qualification Safety Factor • Safety Analysis Safety Factor Report o Procedures Safety Factor o Human Factors Safety Factor o Deterministic Safety Analysis Safety Factor	• CNSC G-129 • CNSC G-228 • CNSC R-116 • CSA N286-05 • CSA N293-07 • SOR/2000-202 • SOR/2000-203 • IAEA NS-G-1.13 • IAEA NS-G-2.4 • IAEA NS-G-2.7 • IAEA NS-G-3.2 • IAEA NS-R-1 • IAEA NS-R-2 • IAEA SSR-2/2	• D081 - Radioactive Material Storage • D266 - Lack of ALARA & Radiation Protection Training for Plant Design Staff • D498 - Airflow from Zone 3 to Zone 2 Does not Meet Design Requirements
Conventional Health & Safety	Conventional Health and Safety is not in the scope of the IIP or GAR. It is addressed as part of the ongoing licensing process.	N/A	N/A
Environmental Protection	Safety Factor Reports • Environment Safety Factor Report • Safety Performance Safety	• CNSC S-99 • CNSC G-296 • CNSC S-296	No ISR Issues requiring action have been identified for the Environmental Protection Safety and Control Area.

Safety & Control Area	Information in the Darlington ISR	Relevant Codes Reviewed in ISR	Relevant ISR Issues Requiring Action[3]
	Factor Report (Environment related Performance Indicators)	• CSA N288.1-M87 • CSA N288.4-M90 • ISO 14001:2004 • MISA Ontario Reg 215/95 • IAEA NS-G-1.13 • IAEA NS-G-2.7 • IAEA NS-G-3.2 • IAEA NS-R-1	
Emergency Management and Fire Protection	• Emergency Planning Safety Factor Report • Hazard Analysis Safety Factor	• CSA N293-07 • National Building Code of Canada • National Fire Code of Canada • NFPA-20 • NFPA-24 • CNSC G-225 • Province of Ontario Nuclear Emergency Plan • IAEA GS-G-2.1 • IAEA GS-R-2 • IAEA NS-G-3.2	***Fire Protection*** • D170 - Fire Safety Plan Requirements • D181 - Fire Safety Training Requirements ***Emergency Management*** • D337 - No Governance Reference for Zone Definitions • D338 - No Governance to Maintain 5 MDUs • D339 - Radiological Event Monitoring Support for Non-OPG Events
Waste Management	Nuclear Waste is discussed in: • Safety Performance Safety Factor Report • Plant Design Safety Factor Report • Emergency Planning Safety Factor Report	• CNSC G-296 • CNSC S-296 • ISO 14001:2004	No ISR Issues requiring action have been identified for the Waste Management Safety and Control Area

Safety & Control Area	Information in the Darlington ISR	Relevant Codes Reviewed in ISR	Relevant ISR Issues Requiring Action[3]
	• Environment Safety Factor Report Conventional Waste and Decommissioning are not in the scope of the ISR		
Security	The Security Safety and Control Area is addressed separately within the ISR due to the confidential nature of the information.	• CNSC S-298 • CNSC G-208 • CNSC G-274 • CNSC RD-363 • SOR/2000-209 • Design Basis Threat • IAEA INFCIRC/225/ Rev 4 • 2009 Darlington Site Security Report	Any actions are considered separately from this assessment due to the confidential nature of the information.
Safeguards	• Safeguards Safety Factor Report	• Nuclear Safety and Control Act • CNSC S-99 • CNSC RD-336 • AECB-1049 Rev 2 • SOR/2000-202 • IAEA INFCIRC/164 • IAEA INFCIRC/164/ADD.1	No ISR Issues requiring action have been identified for the Safeguards Safety and Control Area
Packaging & Transport	Packaging and Transport is not in the scope of the IIP or GAR. It is addressed as part of the ongoing licensing process.	N/A	N/A