Darlington New Nuclear Project

Application for a Licence to Construct a Reactor Facility

October 2022

ONTARIO POWER GENERATION

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Executive Summary

Ontario Power Generation Inc. (OPG) currently holds a Nuclear Power Reactor Site Preparation Licence (PRSL) [R-1] for the Darlington New Nuclear Project (DNNP). OPG is applying for a Canadian Nuclear Safety Commission (CNSC) Licence to Construct (LTC) the first of up to four nuclear power reactors on the Darlington New Nuclear site.

Cleaner Energy Solution

The world is facing a climate change crisis. Most scientific and international expert organizations concur that unless humanity stops burning fossil fuels for energy, greenhouse gas (GHG) emissions will make large parts of the world effectively uninhabitable within this century. Meanwhile, the demand for energy will increase worldwide, whether to address economic imbalances, reduce poverty, improve standards of living, support increasing world populations, or simply to satisfy consumer demand for goods and services. As a result, cleaner energy solutions that do not release significant GHGs are required now.

International experts have determined that the demand for clean energy sources will increase by a factor of 2 to 3 by 2050 [R-34].

Hydropower, wind, solar, geothermal, tidal power and nuclear are the primary low-GHG emitting ways of generating electricity and heat. Some of these can be used to produce clean fuels, such as hydrogen, to “store” energy and transport it where it is needed, while batteries and pumped hydro storage may be used to store energy for future use.

There is no single solution that will meet all needs in all circumstances. System planners must take many factors into account to meet the demands of governments, people and industrial users who require a stable and reliable supply of electricity and thermal energy.

While many analyses have been performed to identify the right technical solutions and in what proportion, the evidence is clear as stated by experts like the International Energy Agency [R-35] and the Intergovernmental Panel on Climate Change [R-36]. Nuclear power is essential in attaining GHG emission reduction targets.

Having analyzed the electricity and energy needs of Ontario, OPG, in consultation with the Provincial government and the Independent Electricity System Operator, has concluded that a mix of technologies will be needed in Ontario. This includes maintaining existing nuclear power energy and capacity where possible (for example through Darlington Refurbishment) as well as adding new nuclear capacity.

New Nuclear Technology

For nuclear power to play a significant role in helping the world decarbonize, nuclear power must be deployed quickly and cost effectively.

OPG has identified that new nuclear must be simpler than existing reactors. Simplicity enables safety and efficiency – reactors that are easier and more efficient to construct, operate and maintain.

Through 2019-2021, OPG undertook an extensive and rigorous selection process to identify the best nuclear technology option for deployment at OPG's existing DNNP site. This decision process resulted in selection of GE-Hitachi as the technology partner with their BWRX-300 small modular reactor (SMR).

The BWRX-300 is a leading example of a reactor design that is much simpler than past reactor power plant designs. As well, the programs for operation of new nuclear are expected to be simpler while remaining effective. These enhancements do not detract from safety – on the contrary, they enable increased safety.

Nuclear Safety

Safety is a critical consideration in a technology selection and in that regard, the BWRX-300 meets and exceeds regulatory requirements as well as OPG's expectations. Incorporation of the lessons learned through nine previous generations of boiling water reactor (BWR) technology, deployed in many countries around the world, has enabled incorporation of significantly increased passive safety features above and beyond any regulatory requirement. GEH has brought world leading safety to the BWRX-300 design.

As a result of that increased safety as well as simplicity of design, the BWRX-300 can have a much smaller emergency planning zone (EPZ).

In this licence application, OPG is drawing on a team of highly experienced and capable world-class companies. OPG has an established record of decades of safe and reliable operations recognized in CNSC annual regulatory reviews and by World Association of Nuclear Operators (WANO) among others. OPG chose GE-Hitachi as its SMR technology partner in large part because of its proven capability and experience in reactor design and support of safe nuclear operations around the world.

OPG, in partnership with its team of industry experts, will ensure the construction of this SMR meets or exceeds regulatory and safety requirements while achieving the efficiency improvements expected for a fleet of SMRs.

Application

This licence application provides the information required to demonstrate that OPG meets or exceeds the applicable requirements of the *Nuclear Safety and Control Act* (NSCA) [R-37] and the associated Regulations.

This application demonstrates that the DN site remains suitable for a new Nuclear Generating Station (NGS). The application and supporting documents provide the results of the design and preliminary safety analyses for the BWRX-300 reactor that demonstrates its overall safety.

The application also provides an overview of the supporting programs with which OPG will oversee and control all the activities undertaken on the DN site.

Implementation of these programs by qualified staff will ensure that all work is performed with quality to the appropriate standard and with minimal impact to the public, workers, and the environment.

Collectively, these elements ensure that safety is the overriding priority in any activities covered by this licence application, which include:

- the completion of any remaining site preparation activities;
- the construction and the fuel-out commissioning of a single BWRX-300 reactor on the DNNP site; and,
- the construction of support structures, such as cooling water systems, to support up to four units on the DNNP site.

Site Preparation

OPG has advanced on the conduct of site preparation activities under the existing licence [R-1]. These activities will continue under the existing licence and may extend into the early stages of a new construction licence.

DNNP Commitments

OPG has made further progress in completing commitments made during the Joint Review Panel (JRP) process, as accepted by the Government of Canada.

OPG has completed DNNP commitments required for site preparation early works activities and is moving forward on commitments tied to the construction phase. OPG will continue to report the status and closure of commitments listed in the DNNP Commitments Report [R-2], which is summarized in Section 5.6 of this Application.

Public Engagement and Communications

OPG values the relationships it has with communities, the public and stakeholders.

OPG fosters open and ongoing communications and engagement programs with the public and stakeholders in communities where our facilities are located, as well as with the broader public.

OPG keeps the public and stakeholders informed about DNNP as part of the existing engagement and communications activities for the Darlington Nuclear Generating Station (DNGS) and provides regular updates on the DNNP through various methods and forums, thereby ensuring transparent disclosure of its activities and their potential impacts.

OPG's relationship with the local community remains strong due to ongoing open engagement and sustainable partnerships with community stakeholders, including Indigenous Nations and communities, government, media, business leaders, educational institutions, interest groups, and community organizations.

Indigenous Engagement

The lands on which the DNNP is situated are the traditional and treaty territory of the Williams Treaties First Nations, which includes Curve Lake First Nation, Hiawatha First Nation, Alderville First Nation, Chippewas of Beausoleil First Nation, Chippewas of Georgina Island First Nation, Chippewas of Rama First Nation, and the Mississaugas of Scugog Island First Nation. It is also within the shared traditional territory of many nations including the Chippewa and Mississauga Anishinaabeg, the Haudenosaunee, and the Huron-Wendat peoples.

OPG acknowledges the Aboriginal and Treaty rights of Indigenous Nations and communities as recognized in the Constitution Act, 1982 and regularly undertakes engagement with Indigenous Nations and communities with asserted or established Aboriginal and treaty rights and/or interests proximate to the project site.

OPG recognises that engagement begins with relationship-building and establishment of trust and is committed to respect, openness and transparency in building these relationships. Regular project updates are provided to Indigenous Nations and Community representatives and OPG has hosted a number of tours of the DNNP site and our current operations and fuel storage facilities. At this stage of the project, the formal engagement has included information sharing and requests for feedback from the communities, with an emphasis on working together to recognise and minimize environmental impacts, discussion of economic benefit opportunities, as well as the potential to incorporate Indigenous Knowledge into the DNNP.

Overall Conclusion

In summary, this licence application contains sufficient information to demonstrate that OPG meets all the requirements of the NSCA and the associated Regulations and demonstrates that OPG:

- is qualified to carry on the activities to be licensed; and
- will, in carrying on that activity, make adequate provision for the protection of the environment, the health and safety of persons and the maintenance of national security and measures required to implement international obligations to which Canada has agreed.

Specifically, as evidenced throughout this application and supporting documents, OPG asserts that consistent with the scope of this application:

- Nuclear safety will be assured such that the public, personnel and the environment are protected
- The DNNP site is suitable for the construction of a new BWRX-300 nuclear reactor;
- A management system is in place to effectively conduct the proposed licensed activities through the licence period;
- Staff are qualified and competent to carry on the proposed licensed activities; and
- Transparency and appropriate Indigenous and public consultations will continue.

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1.0 Overview

1.1 Introduction

Ontario Power Generation Inc. (OPG) is responsible for approximately half of the electricity generation in the Province of Ontario. OPG provides low-cost power in a safe, clean, reliable, and sustainable manner for the benefit of the people of Ontario and our shareholder, the Province of Ontario.

The Darlington Nuclear (DN) site is home to the four-unit Darlington Nuclear Generating Station (DNGS) and the Darlington Waste Management Facility (DWMF). DNGS was commissioned by OPG's predecessor company, Ontario Hydro, with the first unit commissioned in 1990.

The Darlington New Nuclear Project (DNNP) is located on the DN site in the Municipality of Clarington, in the Regional Municipality of Durham. The lands on which the DNNP is situated are the traditional and treaty territory of the Williams Treaties First Nations, which includes Curve Lake First Nation, Hiawatha First Nation, Alderville First Nation, Chippewas of Beausoleil First Nation, Chippewas of Georgina Island First Nation, Chippewas of Rama First Nation, and the Mississaugas of Scugog Island First Nation. It is also within the shared traditional territory of many nations including the Chippewa and Mississauga Anishinaabeg, the Haudenosaunee, and the Huron-Wendat peoples.

OPG acknowledges the Aboriginal and Treaty rights of Indigenous Nations and communities as recognized in the Constitution Act, 1982 and regularly undertakes engagement with Indigenous Nations and communities with asserted or established Aboriginal and treaty rights and/or interests proximate to the project site. Figure 1.1-1 provides an aerial view of the DNGS site, the DNNP site to the east of it.

Since 2012, OPG has had a Power Reactor Site Preparation Licence (PRSL) for the DNNP [R-8]. In 2020, OPG applied to renew the PRSL and in 2021 the current PRSL 18.00/2031 [R-1] was issued.

The PRSL allows OPG to conduct site preparation activities for the future construction and operation of a new Nuclear Generating Station (NGS) with a maximum net electrical output of 4800 megawatt electric (MWe).
Since 2012, OPG has continued to fulfill the requirements of the PRSL. Through annual reporting and regular updates, OPG has provided the status and progress of the DNNP activities and the commitments that OPG made during the Joint Review Panel (JRP) environmental assessment approval process, as accepted by Government of Canada and documented in the *Darlington New Nuclear Project Commitments Report* [R-2]

After a thorough review of several reactor options, in December 2021 OPG announced that the selection process was complete and that OPG had chosen the General Electric Hitachi (GEH) BWRX-300 reactor for deployment at the DNNP site. The BWRX-300 is a 300 MWe (approximate gross output) water-cooled, natural circulation Small Modular Reactor (SMR) utilizing simple, natural phenomena-driven safety systems.

On December 2, 2020 [R-3], OPG notified the Canadian Nuclear Safety Commission (CNSC) that it would be proceeding with an application to construct a new power reactor on the DNNP site. This application document and its supporting documents constitute OPG's application for a Licence to Construct (LTC) at the DNNP site. The licence, if approved, will replace the current PRSL. Figure 1.1-2 lays out the tentative DNNP timeline up to and including the issuance of an operating licence. OPG recognizes the timeline will depend on the successful completion of the project deliverable and the CNSC's approval of the proposed licence applications. Further details are provided in the project schedule [R-24]

To support this application and to meet commitments made during the Environmental Assessment (EA) conducted for DNNP, OPG has undertaken a comprehensive review [R-4] of the Environmental Impact Statement (EIS) for the DNNP site, taking into consideration up to four BWRX-300 units. The review concluded that DNNP is not likely to cause significant residual adverse environmental effects, provided the mitigation measures are implemented, and the conclusions of the original EA remain valid. A site evaluation study [R-5] confirmed that the DNNP site remains suitable for a new NGS.

A preliminary safety analysis was conducted for the construction and operation of a BWRX-300 reactor on the DNNP site. The results of the analysis are documented in the Preliminary Safety Analysis Report (PSAR) [R-6] and are referenced throughout this application.

The sections that follow provide an overview of the project, the proposed licensed activities, a summary of the safety analysis and design, a description of the programs to support the activities, and other matters of regulatory interest including public and Indigenous Nations and communities' engagement.

Throughout this application, reference is made to several key supporting documents. These supporting documents along with this document, make up the entirety of the licence application that meets regulatory requirements.

Figure 1.1-1: Aerial View of DNGS from the West,

DNNP Site is Immediately East of the Existing Station.

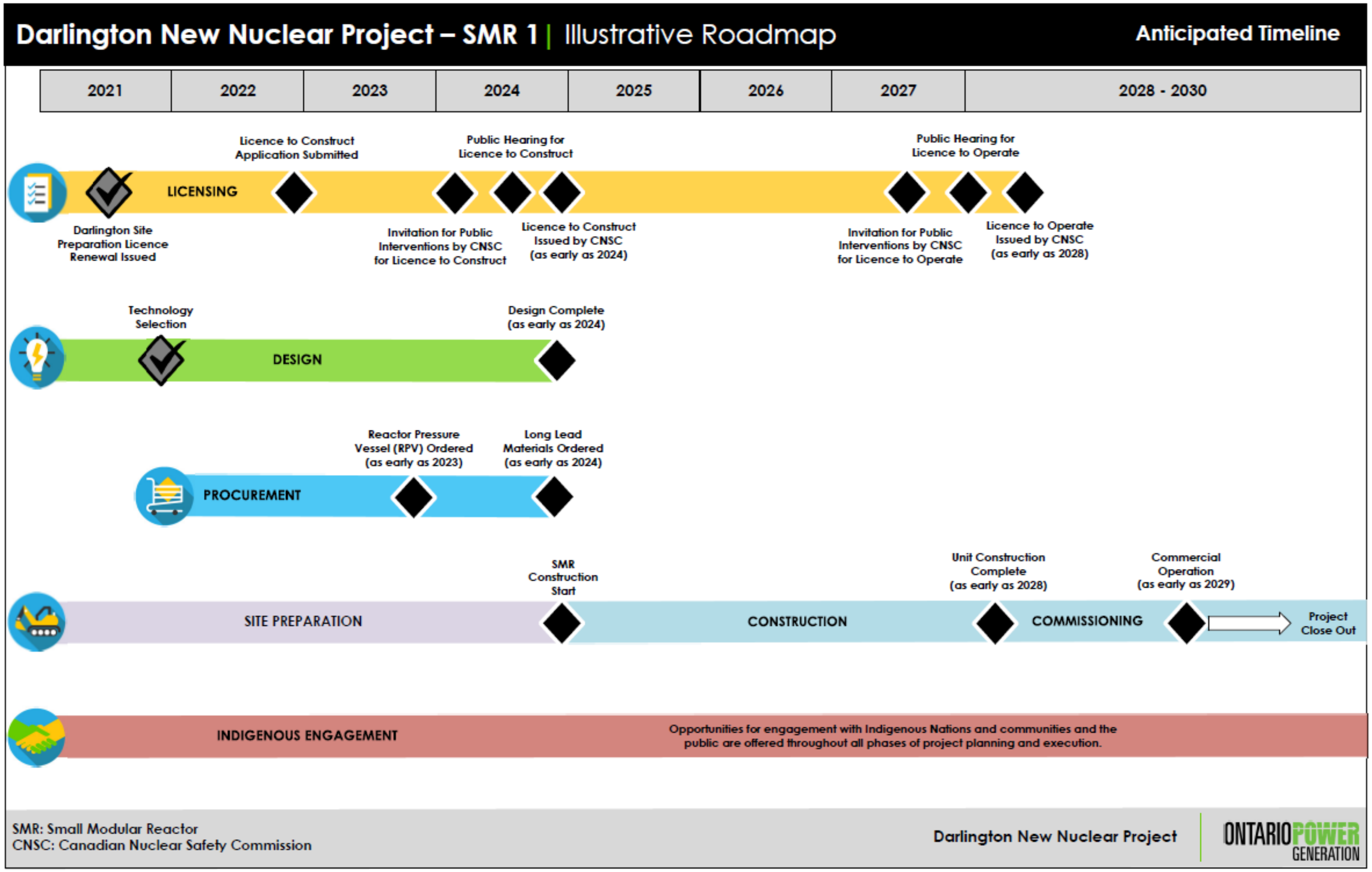


Figure 1.1-2: DNNP Project Roadmap

1.2 General Description of Applicant

1.2.1 Name and Business Address

Licence Applicant and Complete Legal Name: Ontario Power Generation Inc.
Business Address: 700 University Avenue 19th Floor, Toronto, Ontario M5G 1X6

1.2.2 Mailing Address

Same as information in Section 1.2.1.

1.2.3 Persons who have Authority to Interact for the Applicant with CNSC

The persons who have the authority to interact for OPG with the CNSC for the purposes of this Application are as follows:

Mr. Mark Knutson
Title: Senior Vice President, Enterprise Engineering & Chief Nuclear Engineer
Contact Information: mark.knutson@opg,com (905) 839-6746 Ext. 5418
Address: 889 Brock Road, Pickering, Ontario L1W 3J2

Mr. Dragan Popovic
Title: Senior Vice President, SMR Execution
Contact Information: d.popovic@opg.com (416) 277-2173
Address: 230 Westney Road South, Ajax, Ontario, L1S 7P9

Mr. Andy Owen
Title: Vice President, Security & Emergency Services
Contact Information: andy.owen@opg.com (289) 387-1403
Address: 889 Brock Road, Pickering, Ontario L1W 3J2

Ms. Sara Irvine
Title: Director, Nuclear Regulatory Affairs
Contact Information: sara.irvine@opg.com (289) 314-3367
Address: 889 Brock Road, Pickering, Ontario L1W 3J2

Ms. Sevana Bedrossian
Title: Manager, Regulatory Affairs – DNNP Licensing

Contact Information: sevana.bedrossian@opg.com (416) 716-3879
Address: 230 Westney Road South, Ajax, Ontario, L1S 7P9

1.2.4 Proof of Legal Status

A copy of OPG's Articles of Incorporation [R-7] has been previously submitted to the CNSC as part of the original PRSL application [R-9]. The corporate information is as follows:

Corporation's Legal Name: Ontario Power Generation Inc.
Corporation Number: 001720591
Date of Incorporation: January 1st, 2007
Registered Office Address: 700 University Avenue 19th Floor, Toronto, Ontario M5G 1X6

1.2.5 Evidence that Applicant has Ownership of the Site

A copy of the Transfer/Deed of Land filed in the Land Titles for the Province of Ontario showing OPG-Darlington Inc. as the owner of the DN site was attached in OPG's original PRSL application [R-9]. The survey that identifies the property described in the Transfer/Deed of Land was also provided along with the original application.

Effective June 20, 2007, by Instrument Number DR614956, the registered owner of the lands changed from OPG-Darlington Inc. to Ontario Power Generation Inc. [R-11].

On March 31, 2015, Ontario Power Generation Inc. sold and transferred certain lands to Her Majesty the Queen in Right of the Province of Ontario, represented by the Minister of Transportation and similarly to the Municipality of Clarington on September 30, 2015. These lands were transferred to the province and to the municipality to support highway infrastructure projects including the Highway 407/401 Durham East Link, and interchange improvements at Holt Road and Highway 401. These transfers have no impact on OPG's safe management of the land covered under the PRSL. No licensed activities were intended to be performed on the transferred lands and there is no change under the current application.

As a result of the sale and transfer of these lands, PIN 26606-0133 (LT) was replaced by PINs 26606-0363 (LT), 26606-0366 (LT) and 26606-0367 (LT).

The legal descriptions for the new PINs that replaced PIN 26606-0133 (LT) are available upon request.

1.2.6 Identification of Persons Responsible for Management and Control of Licence

Mr. Mark Knutson, Senior Vice President, Enterprise Engineering & Chief Nuclear Engineer, is responsible for management and control of the licence.

1.2.7 Billing Contact Person

Information regarding billing contact person is as follows:

Mr. Mark Knutson
Title: Senior Vice President, Enterprise Engineering & Chief Nuclear Engineer
Contact Information: mark.knutson@opg,com (905) 839-6746 Ext. 5418
Address: 889 Brock Road, Pickering, Ontario L1W 3J2

1.2.8 Legal Signing Authority

Information regarding the legal signing authority is the same as provided in Section 1.2.7.

2.0 General Description of the Project

This section of the licence application describes the activities to be licensed and provides a short descriptive overview of the project.

2.1 Facility and activities to be licensed

2.1.1 Licence period

OPG is requesting a ten (10) year licence to authorize the activities as described in section 2.1.2.

2.1.2 Statement of the main purpose

<u>Activities to be Licensed</u>

OPG is requesting a licence to construct a Class 1A nuclear facility at OPG's DN site located in the Municipality of Clarington, in the Regional Municipality of Durham. The proposed activities include:

- the completion of any remaining activities under the existing site preparation licence;
- the construction of one powerblock, which includes the structures, systems and components (SSCs) associated with the reactor building, control building, turbine building, and auxiliaries;
- the construction of the support structures for up to four BWRX-300 units; and
- the inspection and testing of equipment, and the conduct of fuel-out commissioning (i.e. the commissioning of systems prior to loading fuel in the reactor).

The project is known as OPG's Darlington New Nuclear Project (DNNP) and its primary purpose is to supply low-greenhouse gas emitting electrical power to the Ontario grid. The construction period is planned to start immediately following the granting of this licence.

Under the proposed licence, OPG will continue to possess and use prescribed information that is required for, associated with, or will arise from the proposed activities. This permission is currently held under the PRSL.

OPG currently holds separate licences to import/export of controlled information for the DNNP. If additional importing/exporting of controlled information or equipment is deemed necessary in the future, separate licenses or permits will be sought.

This Application is not requesting permission to possess, store, or use nuclear fuel. Fuel receipt, possession, and fuel-in commissioning will be addressed as part of future licensing activities.

<u>Description of the Nuclear Facility</u>

OPG is planning to construct a nuclear facility which utilizes the GEH BWRX-300 reactor technology. The DNNP BWRX-300 will be built adjacent to the current DNGS, comprised of four CANDU nuclear reactors, on the north shore of Lake Ontario in Clarington, Ontario. Further details about the site are discussed in Section 2.1.3.

The BWRX-300 is a 300 MWe (approximate gross output) water-cooled, natural circulation SMR. It is a tenth-generation evolution of GEH's Boiling Water Reactor (BWR) design and builds upon lessons learned and improvements from previous generations. The BWRX-300 is an evolution of the 1,520 MWe Economic Simplified Boiling Water Reactor (ESBWR) which has received design certification by the United States Nuclear Regulatory Commission (USNRC) [R-12].

Table 2.1-1 below summarizes some of the basic design parameters for the BWRX-300. Figure 2.1-1 provides a conceptual view of the BWRX-300 facility at the DNNP site. Figure 2.1-2 shows the general layout of the powerblock and Figure 2.1-3 provides a conceptual cross section of the powerblock.

Table 2.1-1: Basic BWRX-300 Design Parameters

Parameter Description	Value
Current/Intended Purpose	Commercial – Electric
Main Intended Application (once commercial)	Baseload with load-following capabilities
Output Power (gross)	~300 MWe, 870 MWth
Reactor Type	BWR
Core Coolant	H_2O
Neutron Moderator	H_2O
Steam Supply System	Direct-cycle
Primary Circulation	Natural
Thermodynamic Cycle	Rankine
Secondary Side Fluid	n/a (No secondary side since it is direct-cycle)
Fuel Lattice Shape	Square
Fuel Bundles	GNF2 (240-bundle core configuration)
Rods per Fuel Bundle	92
Fuel Material	UO_2 with 3.81/4.95% (avg./max. enrichment)
Refuelling Cycle	12-24 months

Figure 2.1-1: DNNP BWRX-300 Conceptual View

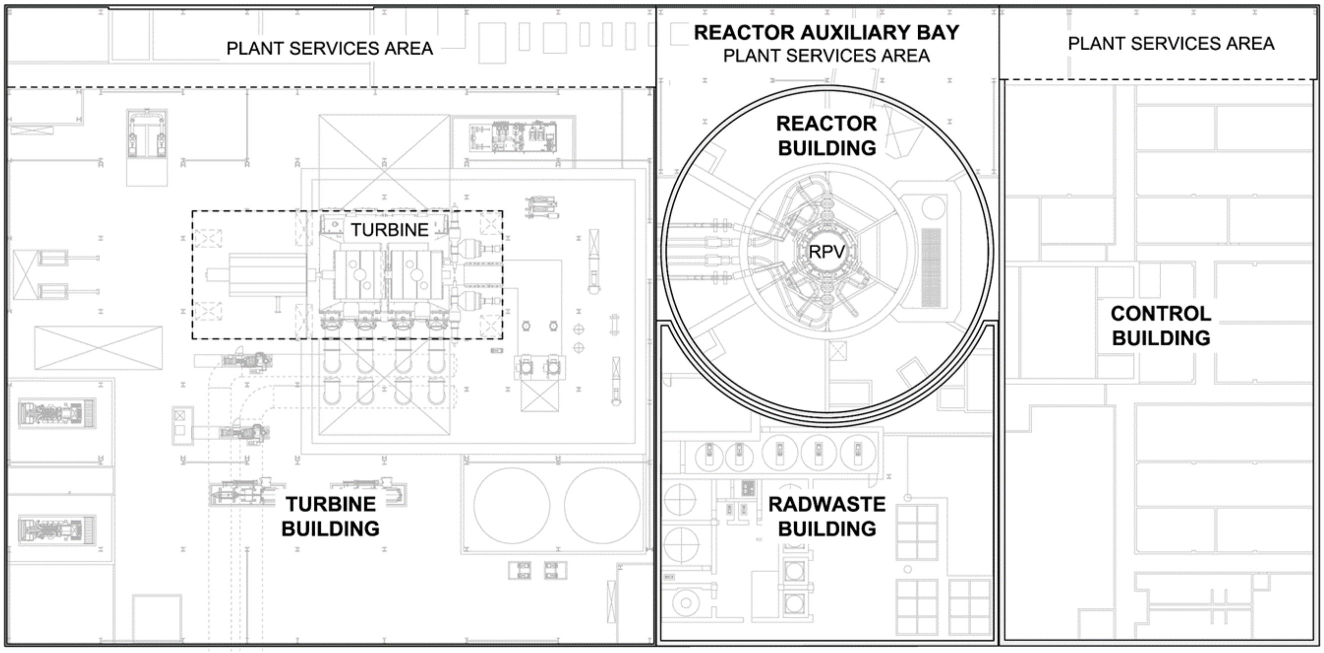


Figure 2.1-2: DNNP BWRX-300 Powerblock Layout

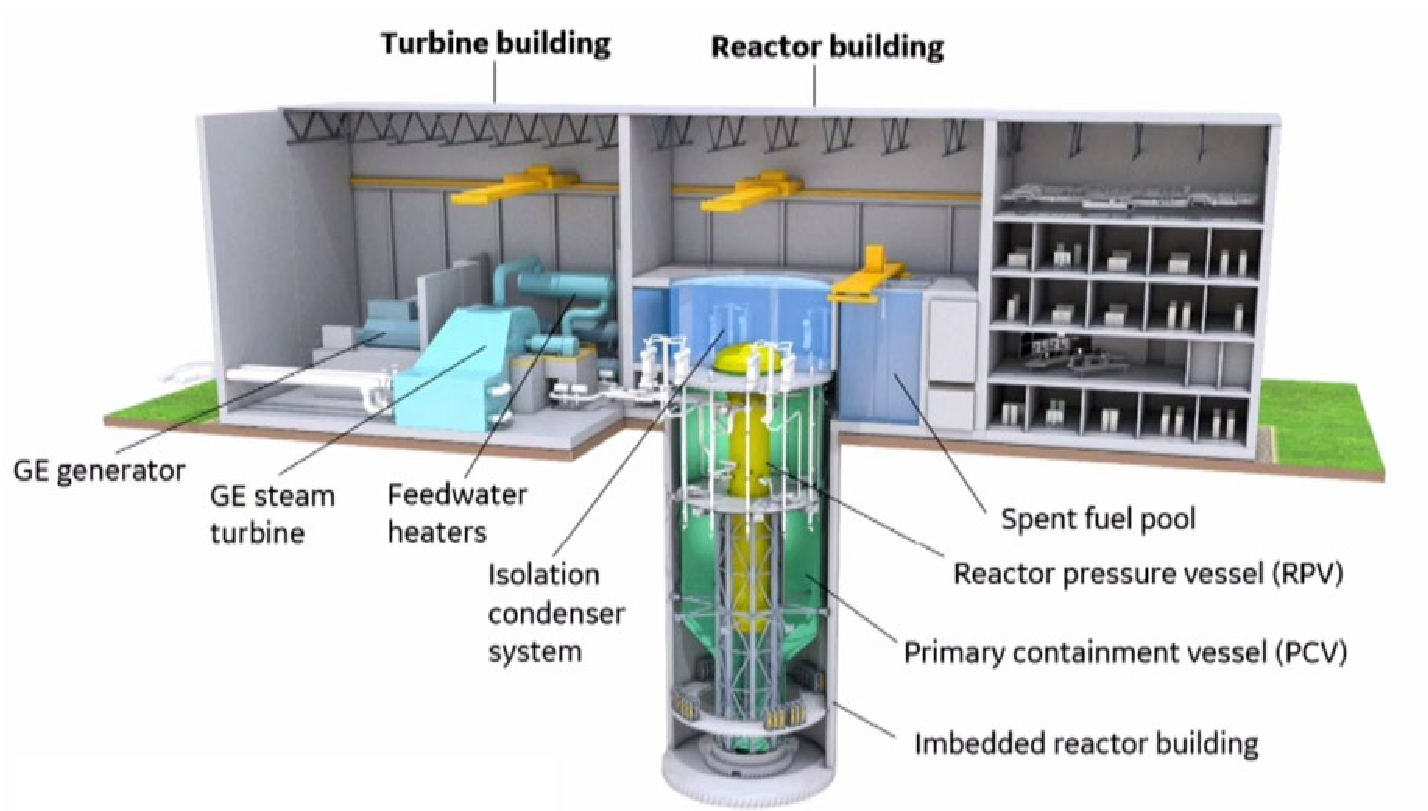


Figure 2.1-3: DNNP BWRX-300 Powerblock Conceptual Cross Section

The facilities to be constructed will consist of the following:

DNNP BWRX-300 Powerblock

Reactor Building (RB) - A shear wall building made of reinforced concrete, steel or steel-plate composite floors and walls. Fuel handling equipment and pools containing water needed for the BWRX-300 passive safety class cooling systems are in the above-grade portion of the RB. A portion of the RB extends below grade where the Steel-plate Composite Containment Vessel (SCCV) and the Reactor Pressure Vessel (RPV) mostly reside. The below-grade portion also houses reactor support and safety class systems and most essential power supplies and equipment. The Secondary Control Room (SCR) located in the RB provides safe shutdown capability as a backup to the Main Control Room (MCR).

Control Building (CB) - Houses the MCR and electrical, control and instrumentation equipment. It is also the entrance and exit for the BWRX-300 power block during normal operations.

Turbine Building (TB) – Houses the steam turbine generator, standby diesel generators, main condenser, condensate and feedwater systems, condensate purification system, turbine-generator support systems, bridge crane, and parts of the off-gas system.

Radwaste Building (RWB) – Houses equipment associated with the handling, processing, and packaging of solid and liquid radioactive waste generated by the nuclear facility. The RWB is used to house equipment and processes that package waste into approved containers.

Facilities/Buildings Outside Powerblock

Plant Service Area (PLSA) – A warehouse that runs alongside the length of the nuclear facility which provides an area for maintenance and storage.

Security building – Controls access to the Protected Area and includes a vehicle inspection area.

Lake Water Intake/Discharge Structures - The BWRX-300 at DNNP will use a once-through lake water cooling system. The water intake will be supplied from Lake Ontario to circulate through the condenser and then discharge to the lake through a discharge duct and outfall structure. Additionally, the shoreline of the lake will include shoreline protection to control erosion. Note that these structures are sized adequately to support up to four BWRX-300 units.

Cooling Water System Pump House – Contains the cooling water system pumps and auxiliary equipment needed to cool the nuclear facility.

Switchyard – A new switchyard will be built on the north side of the nuclear facility for electrical distribution to the Ontario grid. Note that these structures are sized adequately to support up to four BWRX-300 units.

The DNNP site may share some infrastructure with the existing DNGS site such as the Water Treatment Facility, meteorological tower, sewage lift station, roads, etc.

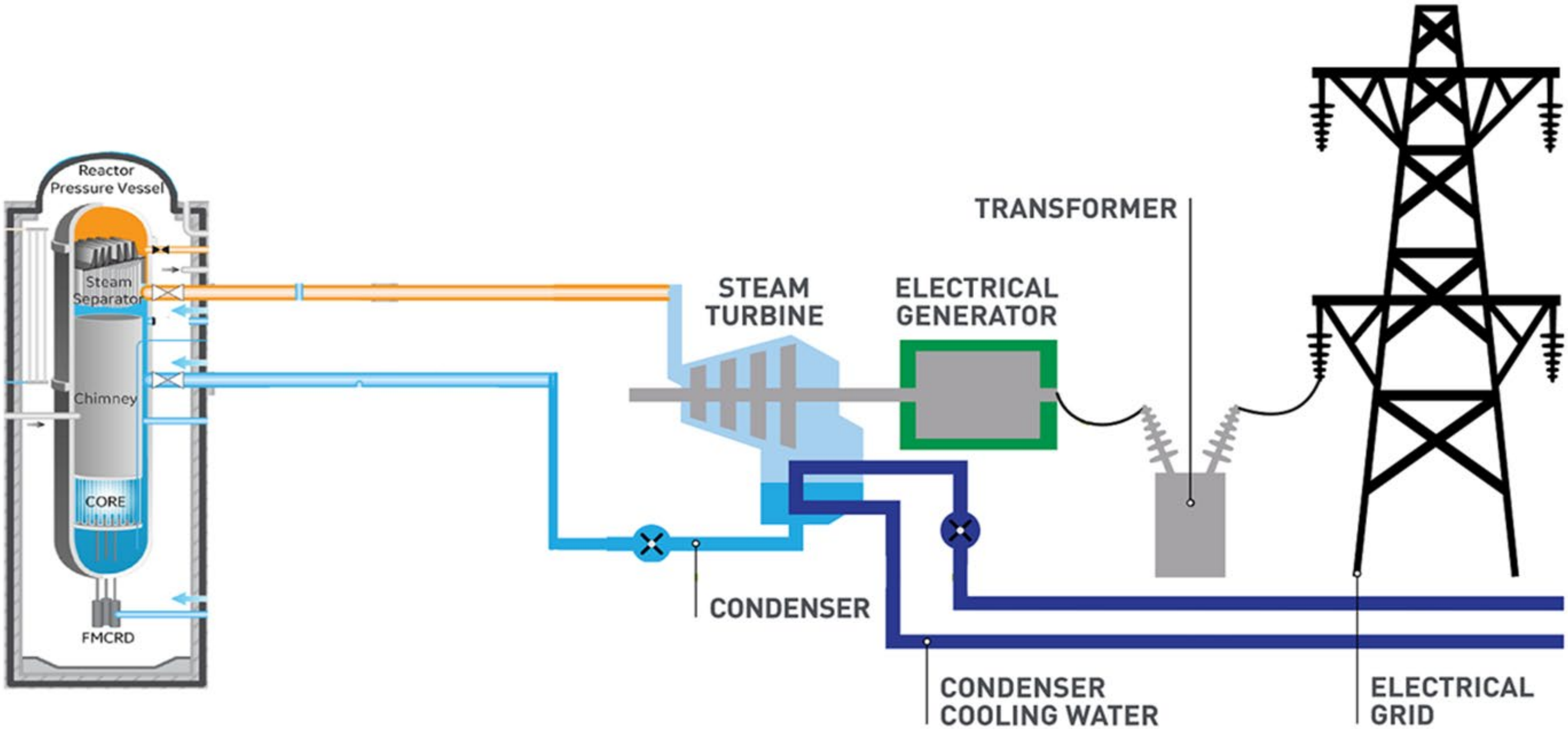


Figure 2.1-4: BWRX-300 Major Systems

Description of Structures, Systems and Components

The following are the major systems and structures of the BWRX-300. Additional detailed descriptions are provided in Section 4.5 of this Application and supporting documents:

Nuclear Boiler System (NBS): the NBS generates and delivers steam to the turbines for safe power generation. The system is comprised of three primary subsystems: the RPV, Main Steam, and RPV Level Instrumentation.

Fine Motor Control Rod Drive System (FMCRD): the FMCRD provides electric-motor-driven positioning of the control rods into and out of the reactor core for reactivity and shutdown control. In response to manual or automatic signals from the Reactor Protection System, the control rods can also be rapidly inserted into the core hydraulically using stored energy in accumulators.

Isolation Condenser System (ICS): the ICS removes decay heat after any reactor isolation and shutdown event during power operations. The ICS limits increases in steam pressure and maintains the RPV pressure at an acceptable level. Reactor heat is transferred from each Isolation Condenser (IC) heat exchanger tube to the surrounding IC pool water by condensation and natural circulation.

Reactor Protection System (RPS): the RPS provides for control of reactivity in various postulated events. RPS is a system of instrument channels, trip logic, trip actuators, manual controls, and shutdown logic circuitry that initiates the rapid insertion of control rods by hydraulic force to shutdown the reactor when specific conditions are detected.

Primary Containment Vessel (PCV): the PCV encloses the RPV and some of its related systems and components. The PCV is dry, leak tight, and located mostly below grade. It provides confinement of radioactive fission products, steam, and water released in the unlikely event of a loss of coolant accident (LOCA).

Passive Containment Cooling System (PCCS): the PCCS is a passive containment heat removal system that maintains the containment within its pressure limits for Design Basis

Accidents (DBAs) such as a LOCA. It consists of several heat exchangers that transfer heat from containment to the reactor cavity pool. PCCS operation requires no sensing, control, logic, or power actuated devices for operation.

Boron Injection System (BIS): the BIS is a manual, independent and diverse system for adding negative reactivity to the reactor. The BIS provides an additional means of reactivity control for the extremely low probability event where control rod insertion (hydraulic or motor) is not successful in meeting the operating limits and conditions (OLC) for effective shutdown. The system provides the operator with a diverse means of increasing the shutdown depth.

Reactor Water Cleanup (RWCU): the RWCU system provides a cleanup flow path from the RPV to the filter/demineralizers during most reactor operating modes. The cleanup or filtration function and ion removal function is performed by the condensate system.

Shutdown Cooling (SDC): the SDC system supports RPV startup and shutdown/cooldown operations. It does this by providing cooling during the shutdown operations and RPV water level management during startup operations.

Isolation Condenser Pool Cooling and Cleanup System (ICPCCS): the ICPCCS ensures the ICS pools are continuously cleaned and maintained at proper temperatures.

Fuel Pool Cooling and Cleanup (FPC): the FPC system has the following purposes:
Maintain the Spent Fuel Pool (SFP) temperature below specified values,
Maintain water quality in the SFP, reactor cavity pool, and cask pit,
Control the SFP, reactor cavity pool, and cask pit water levels.

Containment Inerting System: establishes and maintains an inert atmosphere within containment and maintains a slightly positive pressure in containment to prevent air (oxygen) in-leakage into the inerted spaces from the RB. This system is intended to preclude the combustion of hydrogen and prevent damage to essential equipment and structures.

Fuel and Fuel Cycle: the core design uses a 240-bundle core configuration with Global Nuclear Fuel's GNF2 with its low hydraulic resistance for improved natural circulation.

Instrumentation and Control (I&C): the I&C system (also referred to as the Distributed Control and Information System (DCIS)) is a completely integrated control and monitoring system for the nuclear facility. The I&C system provides control, monitoring, alarming, and recording functions.

Electrical Systems: the electrical system is a completely integrated power supply and transmission system for the nuclear facility. It is divided into subsystems based on safety classification.

Steam and Power Conversion System: the BWRX-300 uses common Balance of Plant (BOP) equipment for its power conversion systems. The steam turbine generator will consist of a multi-extraction single casing high pressure section, nuclear moisture separators, and 2-stage reheaters with multiple double flow low-pressure turbine sections.

Reactor Closed Cooling Water System (RCCWS) and Turbine Closed Cooling Water System (TCCWS): the RCCWS and TCCWS continuously circulate cooling water through various auxiliary equipment heat exchangers and reject heat to the Plant Service Water System.

Plant Service Water System: continuously circulates water from Lake Ontario through the RCCWS and TCCWS HXs and back to the lake.

Makeup Water System (MWS): the MWS provides the storage and transfer of demineralized water for the needs of the plant.

Condensate Storage and Transfer System (CSTS): the CSTS takes water from the Condensate Storage Tank (CST) and provides it to interface systems as required. The CST provides storage capacity for condensate rejected from the Condensate and Feedwater System, for condensate quality Liquid Waste Management System (LWMS)effluent during normal operation, and for Condensate and Feedwater System and condenser hotwell inventory during system maintenance outages.

Chilled Water Equipment (CWE): the CWE provides chilled water to the cooling coils of air handling units and other coolers in various nuclear facility buildings. The chilled water absorbs the rejected heat from these coolers and is pumped through the refrigerant chillers where the heat is transferred to RCCWS and TCCWS.

Radioactive Waste Management Systems: The LWMS is designed to control, collect, process, handle, store, and dispose of liquid radioactive waste generated during normal nuclear facility operation, including anticipated operational occurrences (AOOs). The Solid Waste Management System (SWMS) is designed to collect, monitor, process, handle, package, and temporarily store wet and dry solid radioactive waste prior to shipment. The Off-gas System provides steam condensation and maintains main condenser vacuum.

For further details, refer to Section 4.5 Physical Design. Figure 2.1-4 shows the BWRX-300 major systems.

<u>Practices and Safety Concepts</u>

The BWRX-300 uses proven fuel, materials and a combination of proven and innovative manufacturing techniques while incorporating passive safety response systems and simple design concepts. It implements a safety strategy structured on the five Defence Lines (DLs) of International Atomic Energy Agency's (IAEA) Defence-in-Depth (DiD) methodology. The BWRX-300 DiD concept uses Fundamental Safety Functions (FSF) to define the interface between the DLs and the physical barriers. The FSFs for the BWRX-300 are:

- Control of reactivity;
- Fuel cooling;
- Long-term heat removal; and
- Containment of radioactive materials.

Further details are provided in Section 4.4 Safety Analysis.

<u>General Approach to Design</u>

The high-level design goals for OPG's DNNP BWRX-300 are to develop an electrical generation facility that:

- is low risk to the public;
- has a low environmental impact;
- is highly reliable; and,
- provides low-cost electricity.

To achieve these goals, the design incorporates:

- industry design and operating experience;
- lessons learned from previous designs and events;
- proven and reliable technology;
- advanced design and construction methods;
- proven engineering processes, practices, and tools; and,
- compliance with national and international codes and standards.

The BWRX-300 design achieves a low public risk by implementing:

- Improved safety margins by design, based on understanding of Postulated Initiating Events (PIE) and event sequences. Improved margins reduce the consequences of some PIEs and increase coping times for the operator, and therefore can reduce the sensitivity of these events to immediate actions (e.g., increase in heat sink capacity to increase operator coping time).
- Inherent safety characteristics are used to limit consequences of certain PIEs. This is accomplished by using risk informed design approaches, Operating Experience (OPEX), as well as natural physical characteristics (e.g., size and location of RPV nozzles to reduce potential for inventory losses, natural circulation to remove core decay heat, etc.).
- Operational simplicity reduces the number, complexity, and frequency of operational tasks as well as the maintenance requirements to maintain safe and reliable operation. Operational simplicity reduces the number of complex systems required to operate the station and improves reliability (e.g., passive systems such as PCCS that do not require any significant operator actions to be taken or change in system state to perform its design function).
- Leveraging significant experience from the operational BWR fleet. Hundreds of thousands of hours of safe and reliable operation have been experienced by the current BWR fleet worldwide. Lessons learned from operating and design experience is used to inform the BWRX-300 design. This experience aids in improving design and safety margins (e.g., RPV isolation valves to eliminate a non-isolatable LOCA).
- A proven and modern generation of fuel design is used because of its advanced performance characteristics and reliability. Using a proven fuel type

with known performance characteristics improves predictability and certainty of behavior under all operating conditions.

- Complimentary design features for low probability events. Features have been incorporated into the design to mitigate the impact of certain extremely low probability event sequences (design extension conditions (DECs)) to ensure their consequences remain acceptable (e.g., BIS).
- Practical elimination of event sequences. Using Industry OPEX and risk informed engineering approaches, some traditional event sequences have been practically eliminated (e.g., non-isolatable large break LOCA, control rod drops, and control rod ejection).

The BWRX-300 design minimizes its environmental impact by:

- Maintaining a relatively small overall terrestrial and marine footprint. The design also makes use of an existing site and support infrastructure.
- Maintaining a low waste signature for both conventional and radioactive waste. This is accomplished using advanced design and construction methods that reduce waste during construction as well as the amount of irradiated material for decommissioning. In addition, the design requires fewer operational and maintenance activities that generate waste.
- Requiring a smaller Emergency planning zone (EPZ). A smaller EPZ demonstrates that even during a very low probability event, the overall impact to the surrounding environment is reduced.

High reliability is achieved in the design by using:

- Well established system/component designs with proven OPEX where applicable. With established components and supply chain, design engineering reliability has been well established. Using known system and component designs increases design and operating performance certainty.
- Low complexity systems and components require less reliance on engineered redundancy for improved reliability.
- The use of passive systems that do not require external inputs or a change in state to perform their design functions (e.g., ICS heat removal).
- Reduction in single point vulnerabilities (SPV). SPV are those single components whose failure can cause the station to reduce output or shutdown. Ensuring

numbers of SPV are reduced improves overall station predictability and reliability.

The design supports a low levelized cost of electricity generation resulting by using:

- Advanced construction methods which enable a reduced deployment time; and,
- Fewer systems, components, and personnel required to maintain safe and reliable operation.

Further details on the design approach and methodology are provided in Section 4.5 Physical Design.

General Plans for Construction

The BWRX-300 draws from proven construction methods used in industries such as the water and tunneling industry. The design allows for modularized construction for large components that can be consistently lifted, set, aligned, and fixed in place.

A construction sequence model will be developed to support detailed construction sequencing for determination of travel routes, laydown areas, fabrication and storage facilities, and general construction site layout. This model will determine the required site improvement changes.

Improved Construction Approaches

The construction of the BWRX-300 plant uses a standardized construction plan, which only requires minor site-specific modifications. The key excavation feature will be the reactor shaft development. Straight line shaft excavation using proven methods used by tunneling and hydraulic industries will minimize material removal and backfill requirements, and any environmental impacts.

Steel-composite construction technology (Steel Bricks™) will be used in the DNNP reactor. Steel Bricks™ will be used for the walls and floors of the RB. Steel Bricks™ provide modularity and off-critical path construction capability. Steel Bricks™ utilize formed steel shapes, generally rectangular in nature, that provide both strength and permanent concrete forming. No rebar or rebar assembly is required. Fabricating the individual Steel Bricks™ and forming the blocks into standard truck size shipping panels is the first stage of modularization to perform construction off critical path. This technique of factory construction provides quality control (QC) benefits as well as cost and schedule benefits. Final field assembly into partial ring segments will be performed onsite in a separate fabrication which will also be off critical path construction.

This modularized construction will be conducted in three stages:

- Shop fabrication and pre-assembly of the largest standard shipping load to site;
- Site fabrication to further assemble material to largest possible size for setting with equipment available at site; and
- Final in-place assembly.

This approach will result in less in-situ assembly and inspection and increase the amount of assembly in both onsite fabrication areas and in-shop fabrication with the QC, cost and schedule benefits noted above.

Approaches to Component Manufacture

The nuclear and power industries have significant experience with many of the components utilized in the BWRX-300, along with an existing supply chain which minimizes risks for the project. The risks minimized in this fashion include decreased uncertainty in the manufacturing, material behaviour, testing, Quality Assurance (QA), and demonstration of regulatory and code requirements.

OPG's DNNP will employ factory assembly of the turbine and generator which has proven to be effective in the combined cycle industry. This approach eliminates the need for open-top or open-ended assembly of tolerance critical equipment in a construction environment, reduces the risk of foreign particle entry, and minimizes exposure of the internal components to elements. Also, assembling this equipment in a shop environment by craftspeople that perform these tasks on manufacturer specific equipment daily will improve quality. All such quality improvements are valuable to safety of the plant.

2.1.3 Description of the Site

As shown on Figure 2.1-5, the DN site is located on the north shore of Lake Ontario, about 65km east of the City of Toronto, in the Municipality of Clarington, Region of Durham in Ontario, Canada.

In general, the portion of the DN site proposed for the DNNP is the easterly one third (approximately) of the overall DN site. The DNNP site comprises of approximately 180 hectares of land which is bisected by the CN railway The DNNP site is bounded to the north by Energy Drive, to the south by Lake Ontario, to the west by Holt Road and to the east by the St, Mary's Cement plant. Figure 2.1-6 illustrates the DNNP site boundary.

Figure 2.1-7 is a topographic map illustrating the present contours of the DN site. The existing DNGS site structures are also represented. The DNNP site varies in elevation with the southwest corner already at the lowest elevation. This is a benefit, minimizing environmental impacts as less excavation and site grading will be required. Refer to Sections 3.0 and 5.2 of this Application document for more information.

Optimization of the land and design requirements for the DNNP BWRX-300 have resulted in situating the facility in the southwest corner of the DNNP site near Lake Ontario. Figure 2.1-5 shows the proposed layout and proposed areas for the DNNP site and include the following:

- BWRX-300 powerblock and protected area;
- Condenser Cooling Water (CCW) intake/discharge channel, and forebay structure;
- Switchyard and transmission corridor;
- Site roads and infrastructure, including the heavy haul route;
- Administrative building and parking lot;
- Site service connections;
- Soil spoils stockpile;
- Available space for construction and support (trailer areas, pre-assembly area, batch plant, fabrication shop, truck holding and turn-around, laydown areas);
- CN railway and decommissioned rail line spur; and
- Heavy haul route.

The DN property is bounded by fencing and signage designating OPG property. The DNNP protected area includes the powerblock area and is fenced in as shown in Figure 2.1-8.

The inner and vital areas of the site will be enclosed by fencing and structures, per the requirements of the Nuclear Security Regulations and REGDOC-2.12.1 *High Security Facilities, Volume I: Nuclear Response Force, Version 2* [R-38] and REGDOC-2.12.1, *High Security Facilities, Volume II: Criteria for Nuclear Security Systems and Devices* [R-39]. OPG has determined the facility exclusion zone to be 350m from the boundary of the

RB. The specificities of the exclusion zone can be found in Chapter 15 of the PSAR. This exclusion zone will lie entirely within the DN site, extending eastward onto vacant DNNP land and westward onto dry fuel storage buildings on the east end of DNGS. Conversely, the exclusion zone from the adjacent DNGS covers a portion of the DNNP land and extends onto the DNNP construction site.

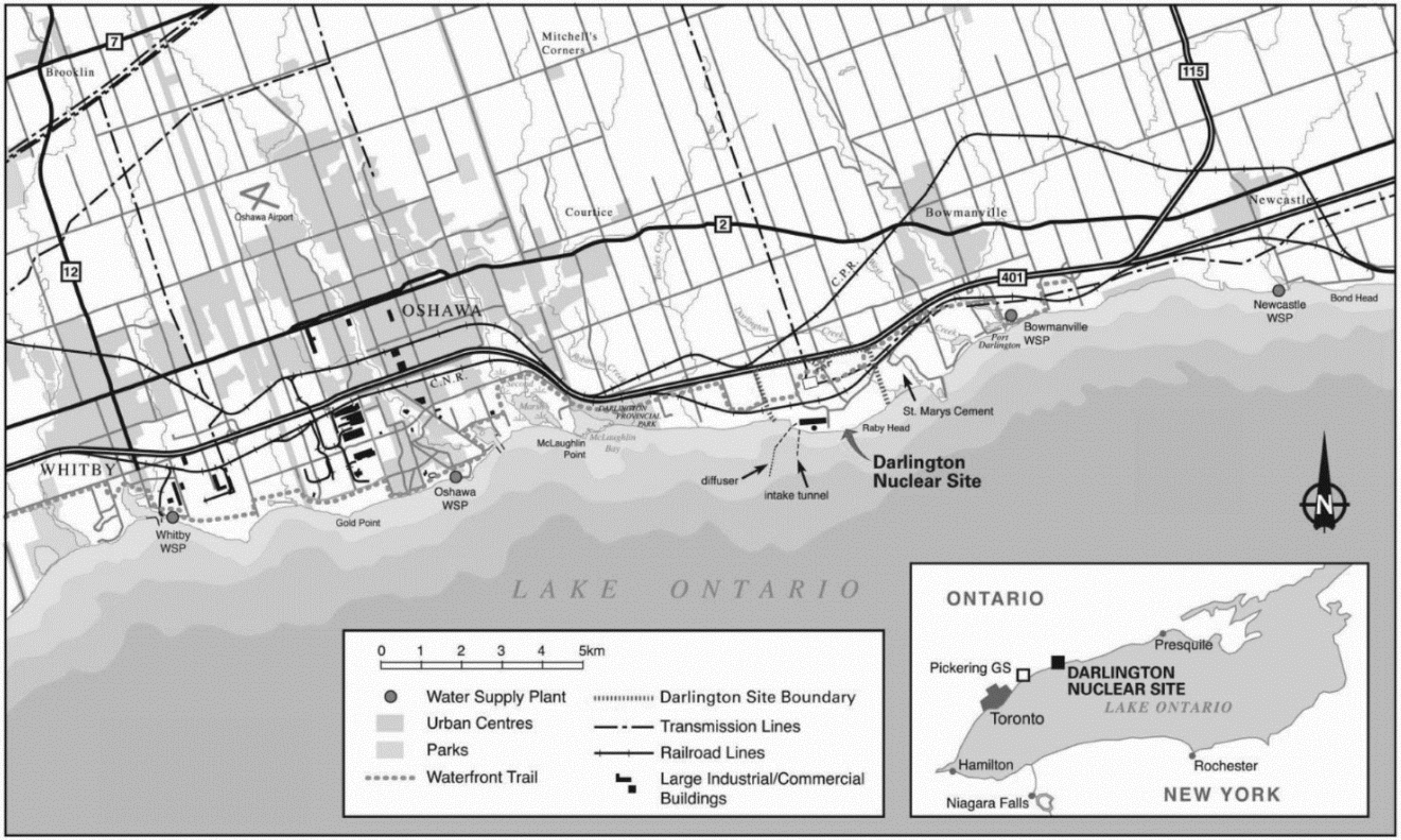


Figure 2.1-5 Location of Darlington Nuclear Site

Figure 2.1-6: Photograph of DN Site Illustrating DNNP Site Boundary

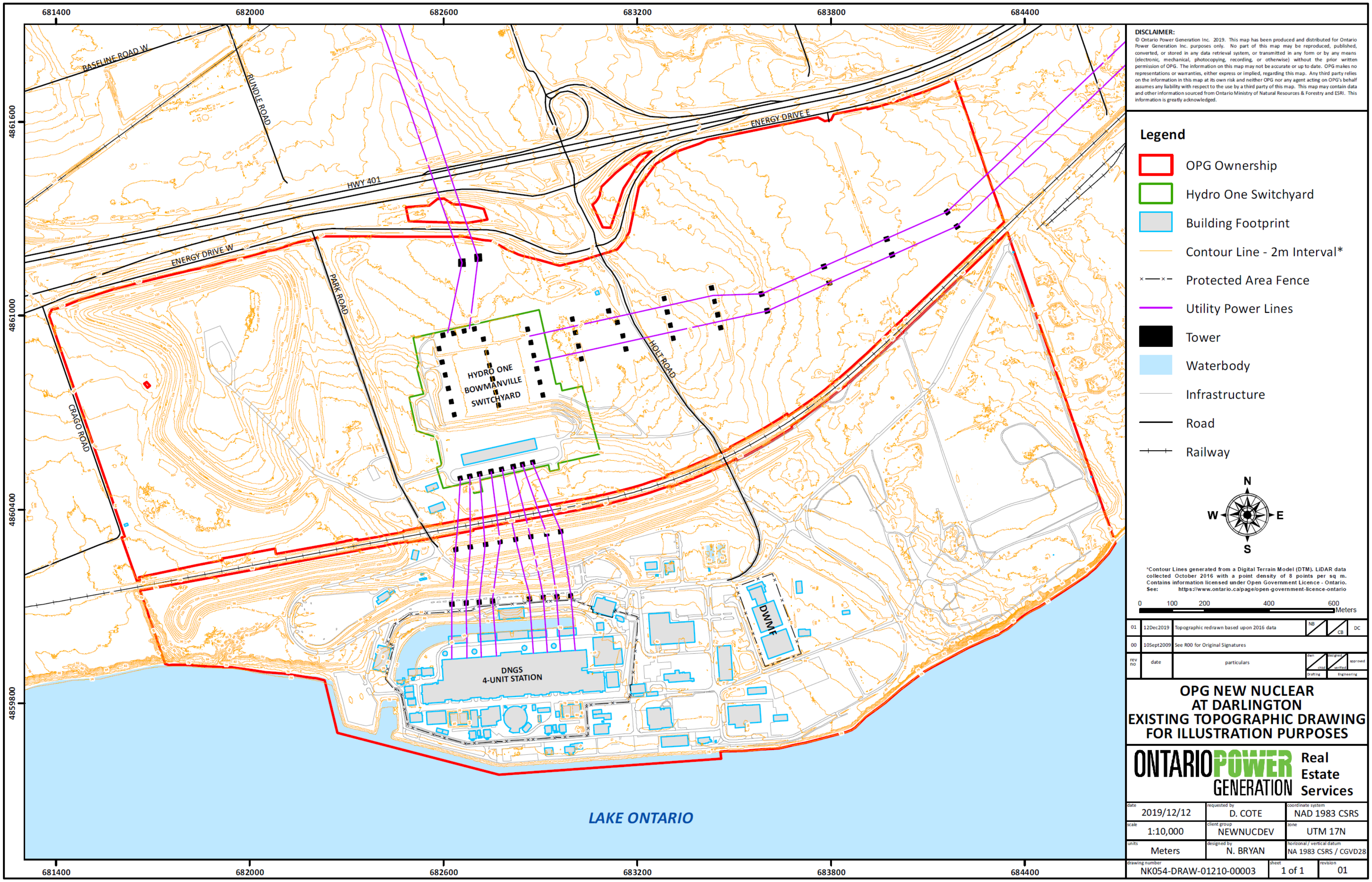


Figure 2.1-7: Darlington Existing Site Topographic Drawing

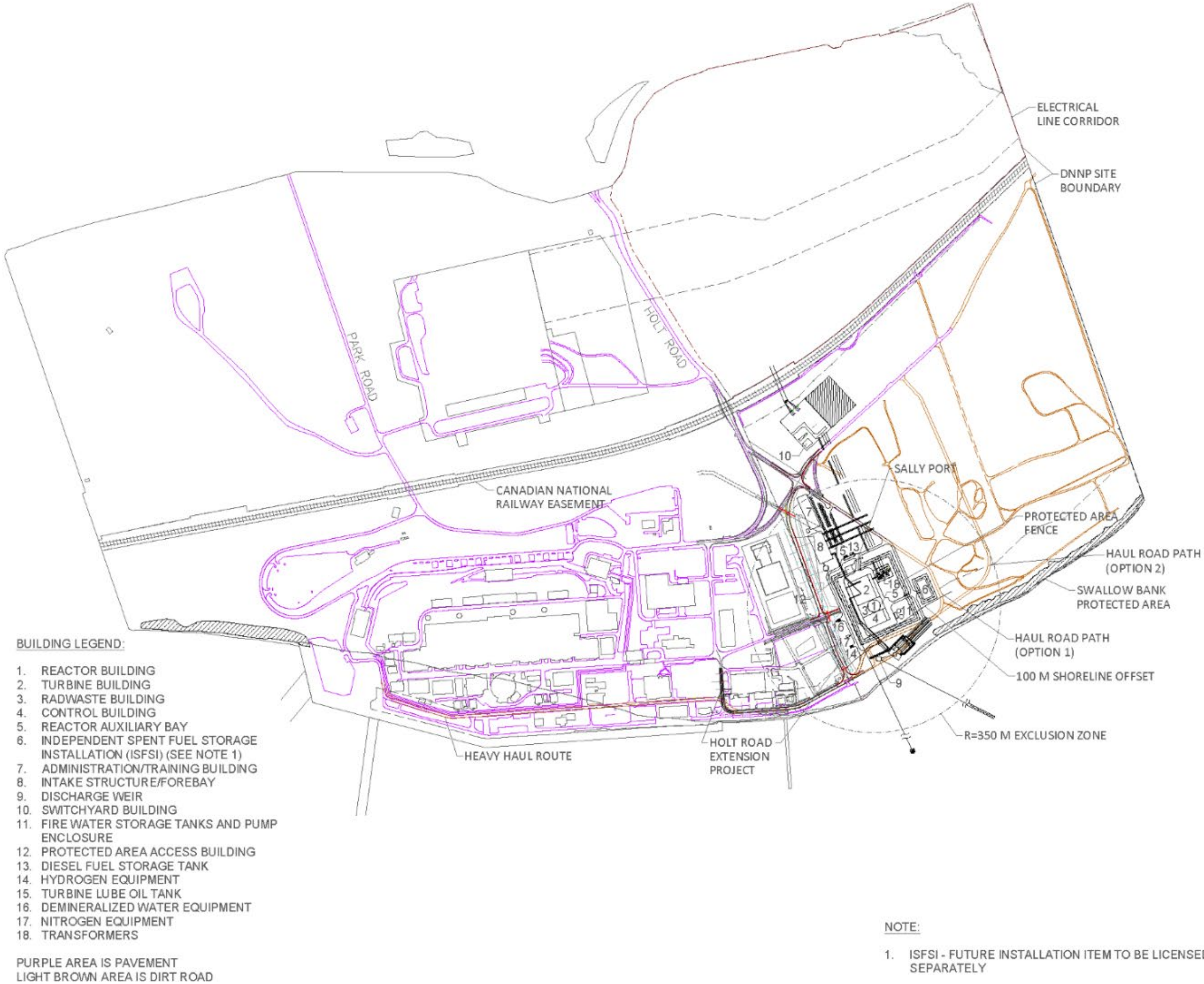


Figure 2.1-8: Proposed Site Layout for Construction

2.1.4 Description of the facility's existing licensing status

OPG currently holds a Nuclear Power Reactor Site Preparation Licence (PRSL), PRSL 18.00/2031 [R-1] for the DNNP. The PRSL allows OPG to conduct the site preparation activities for the future construction and operation of a new NGS with a maximum net electrical output of 4800 MWe.

2.1.5 Nuclear and hazardous substances

The handling of nuclear substances is not part of this Application as the proposed activities to be licensed under the LTC will not involve handling of radioactive materials and will not generate any radioactive wastes. Any activities which would require construction-related tools containing radioactive nuclear substances, as defined in the Nuclear Substances and Radiation Devices Regulations will be performed under the authority of CNSC nuclear substance and radiation device licences.

Any hazardous substances that may be present and/or hazardous wastes generated will be limited to those employed during standard construction processes. These would include chemicals, fuel, lubricants and compressed gases used during operation and maintenance of equipment, as well as solvents and cleaners used to clean the equipment. Additional substances on site may consist of paint, aerosol cans, oil and electrical components used in the construction and relocation of services and utilities, construction of support facilities, and explosives used during excavation activities. PSAR Chapter 20 provides details on hazardous substances.

2.2 Other relevant information

2.2.1 Permits, Certificates and other Licences

OPG currently holds a PRSL [R-1]. The licenced site preparation activities include:

- Construction of access control measures;
- Clearing and grubbing vegetation;
- Excavation and grading of the site;
- Installation of services and utilities;
- Construction of administrative and physical support facilities inside future protected area;
- Construction of environmental monitoring and mitigation systems; and

- Construction of flood protection and erosion control measures.

OPG possesses various import/export licences that allows for the importing and exporting of controlled nuclear information relating to the BWRX-300. OPG maintains an up-to-date list of import and export licences for the DNNP that is available for CNSC review upon request.

OPG is currently acquiring specific federal, provincial and municipal permits, approvals and authorizations to support specific site preparation activities such as an Environmental Compliance Approval (ECA), an Endangered Species Act (ESA) permit, and a Site Planning permit. These will be in place prior to the commencements of those activities. Certain permits, approvals, certificates or authorizations will continue through the construction phase with certain amendments or modifications. OPG will inform the CNSC of any amendments, modifications or new applications for permits, approvals or authorizations as the project advances.

2.2.2 Similar Facilities

Canadian nuclear power plants are currently in operation in Ontario and New Brunswick. OPG operates the Pickering Nuclear Generating Station (PNGS), and the DNGS that is adjacent to the DNNP site.

The currently operating reactors in Canada consist of CANDU pressurized heavy-water reactors that are used to produce electricity.

The DNNP will similarly consist of a nuclear reactor that will be used to produce electricity. The supporting infrastructure are similar to those located at the adjacent DNGS site.

As previously noted, the BWRX-300 is a tenth-generation evolution of GEH's BWR design and an evolution of the 1,520 MWe ESBWR.

2.2.3 Supporting Information

The Fukushima accident was a Beyond Design Basis Accident for which the lessons learned have been well-documented through the completion of an independent review by INPO in April of 2012. To a large degree, the lessons learned have informed the design of the BWRX-300, primarily in how the BWRX-300 copes with Station Blackout events. This includes incorporation of passive design features, in addition to the availability of emergency power supply, capability of using non-permanent sources of electric power and heat sink makeup water, and reliable long-term decay heat removal in accident conditions. This has made the BWRX-300 undoubtedly safer.

As previously noted, the BWRX-300 offers a well-balanced combination of innovation and reliance on proven technology, which incorporates the lessons learned from the Fukushima accident. Most of these novel features of the BWRX-300 when compared to the currently operating BWR and Advanced Boiling Water Reactor (ABWR) fleet are also present in the ESBWR design [R-13].

The underlying safety strategy is based on the application of an enhanced DiD strategy supported by increasing emphasis on inherent safety characteristics, passive safety features and reduced reliance on operator intervention. Accident prevention and mitigation measures are also enhanced to practically eliminate DECs or make them extremely unlikely to occur. The demonstration of plant safety functions during a beyond design basis external event is part of the diverse and flexible coping strategies that form the basis for compliance with regulatory requirements related to the Fukushima event.

DNNP's BWRX-300 uses the same proven BWR fuel design that is successfully operating today. See Section 4.5.8 of this Application for additional information. Because the fuel is the same, the methods for performing nuclear design of the bundles and the core arrangements for those bundles have been demonstrated by many years of successful operation. Similarly, use of the same established fuel provides assurance that the thermal-mechanical design methods for the fuel cladding and fuel assemblies are proven and that these fuel bundles can be successfully manufactured, shipped, and reliably operated.

Given new design features, specific consideration is given to the validation of safety case through adequate experimental investigations. PSAR [R-6] Chapters 3-7, 9 and 14 contain relevant Research and Development (R&D) information to support safety evaluations for different systems. The process of incorporation of R&D in design is described in PSAR Chapter 17.

3.0 Site Evaluation

The DNNP Site Evaluation [R-5] builds on the previous evaluation conducted for the DNNP site preparation licence [R-16], updated in consideration of the chosen technology and any new site characterization information. The evaluation concludes that the DNNP site remains a suitable location for the construction and operation of a new NGS and will not create an unreasonable risk to the public, personnel or environment. For each of the hazard areas evaluated, the risk was determined to be negligible or could be reduced to an acceptable level through design mitigation.

3.1 Site Characteristics

OPG provided information on baseline characteristics of the DNNP site as part of the original PRSL application in 2009 [R-9]. Subsequently, OPG updated the baseline data in the 2020 PRSL renewal application [R-16].

As part of the PRSL renewal application, OPG validated the accuracy of the supporting documents of the original application by conducting reviews against REGDOC-1.1.1, *Site Evaluation and Site Preparation for New Reactor Facilities* [R-18] as well as new or revised codes, standards and practices. Where additional baseline data was collected, the data was evaluated to identify any changes in trends. The conclusion was that the updated baseline conditions did not alter the conclusions of the site evaluation and that the DNNP site continues to be suitable for the construction and operation of a new NGS.

Additional site characterization conducted since the PRSL renewal application include the following:

- Review and update of meteorological data;
- Execution of a geotechnical and seismic hazard investigation program and collection of additional hydrogeological data;
- Collection of additional baseline terrestrial data;
- Assessment of fish habitat for an unnamed, channelized drainage feature traversing the DNNP lands west of Old Holt Road to Lake Ontario;
- Estimation of public dose for the DN site calculated as part of the Darlington Environmental Monitoring Program;
- Completion of a comprehensive soil characterization program in 2021; and

- General land use updates.

A summary of DNNP site characteristics based on the historical and new data is provided in the sections below. Further details regarding the process and site characteristics are provided in Chapter 2 of the PSAR and the DNNP Site Evaluation Update Summary Report [R-5].

3.1.1 Topography

The DN site is situated in an undulating to moderately rolling limestone till plain bisected by the Canadian National Railway's main line in an east to west direction. The previously irregular terrain was graded in the existing DNGS powerhouse area to an elevation of about 100 m. This site elevation of 100 m is equal to an elevation of +78 metres above sea level (masl). The surface elevation rises towards the north with a mean elevation of 122 m just south of the railway tracks and irregular ground from 120 m to 128 m elevation to the north of the tracks. To the east, the site for the DNNP is composed of a gentle slope rising upward from an approximate elevation of 102 m to 112 m over a distance of 400 m. Further east, the existing ground elevations rise substantially to a height of about 130 m near the east site boundary. Note the general topography for the DNNP area has not changed significantly since the original PRSL application. The only notable changes are the reconfiguration of the Holt Road and Highway 401 Interchange and changes to adjacent road names. In summary, these changes have no impact to the conclusions of the original site evaluation.

3.1.2 Atmospheric and Meteorological Environment

Ambient air quality as well as climate and meteorological conditions were initially characterized in the original PRSL application documents [R-9]. Baseline updates were provided in the 2020 PRSL renewal application [R-16]. Some improvements in air quality and minor differences in meteorological conditions were noted at the time of the PRSL renewal, however, there were no impacts to the conclusions of the original site evaluation.

Ambient air quality in Ontario has improved as compared to the conditions documented in the original PRSL application due to the shut-down of coal-fired power plants as well as other government programs and initiatives. There have been no incidences of poor air quality index values in Oshawa since 2013, and there have been no incidences of smog and air health advisories in the York-

Durham Region since the advisories were implemented in 2015. Current air quality trends continue to indicate that smog is not a concern in the York-Durham Region.

Overall, the baseline conditions of the atmospheric environment have remained similar to those presented with the original application.

The DN site is located in southern Ontario, which displays a humid continental climate with four distinct seasons. Based on the latest Canadian Climate Normals spanning the 1981-2010 period, the highest monthly average temperatures, both regionally and locally, occurred in July and the lowest monthly average occurred in January. The regional mean precipitation was highest in August and lowest in February, while the local mean precipitation was highest in September and lowest in February. Based on the 2021 data collected from the meteorological tower located at the DN site, the average wind speed measured at a height of 10 m was approximately 2.4 m/s, and calms were reported 37% of the time. The prevailing winds were from the north-westerly quarter (9.6% of the time) and from the west (8.9% of the time). Minor differences in meteorological conditions have been identified compared to that of the original application supporting documents, however, the changes do not impact the conclusions of the original site evaluation.

3.1.3 Geology and Geophysical Data

Regional and site geology is characterized by upper and lower till layers with predominant glacial deposits between the upper and lower till layers, overlaying bedrock. The glacial deposits are associated with the Oak Ridges Moraine.

Surficial till layers include the upper Newmarket Till (silt till), followed by the lower Sunnybrook Till (fine sandy silt till with medium to coarse sand and clay and trace fine gravel), which is situated above bedrock. The glacial deposits spanning between the till layers consist of interglacial deposits of fine sand and silt layers known as the Thorncliffe Formation. Interglacial deposits also reside beneath the lower till layer, likely corresponding to the Scarborough Formation.

Bedrock originates from Ordovician-age sedimentary sequences which consists of shale and limestone associated with, in order of increasing depth, the Blue Mountain Formation, Lindsay Formation, Verulam Formation, Bobcaygeon Formation and Gull River Formation. Finally, the Shadow Lake Formation, a sandstone and shale unit, is situated above the Precambrian Basement.

Since the 2020 PRSL renewal application, a geological investigation was conducted for the DNNP's onshore power block area in 2021 and the findings were consistent with the conditions described in previous licence application documents. Overall, geology (including structural geology) and geotechnical aspects for the DNNP are consistent with the conditions described in the original PRSL application [R-9], therefore, the conclusions of the original site evaluation remain unchanged.

3.1.4 Hydrology

Surface drainage at the DN site is essentially divided by a railway line which runs east to west across the site. The area south of the railway tracks generally slopes toward Lake Ontario while the area north of the railway tracks and east of Holt Road slopes toward the east. In the developed parts of the DN site, stormwater is collected in natural channels/swales and constructed outfalls and conveyed to Lake Ontario.

In terms of lake currents in Lake Ontario, there is very little net flow along the northern shore of Lake Ontario. However, the current in the nearshore region is overall easterly and is influenced by brief patterns of strong winds exerting stress at the water surface. Lake current speeds for all directions for the 2012-2016 period typically ranged from about 9 to 18 cm/s and were typically slower in spring and early summer, May through June, than during the late summer, fall and winter seasons, August through April. Hydrological site characteristics are discussed in further detail within PSAR Section 2.8 [R-6].

No additional hydrological studies have been conducted since the 2020 PRSL renewal application [R-16] as the hydrological characteristics of the site are expected to remain unchanged.

3.1.5 Hydrogeology

<u>Groundwater Flow</u>

Groundwater aquifers at the DN site have been categorized into three hydrostratigraphic units: Shallow/Water Table, Interglacial Deposits, and Shallow Bedrock. Within the Shallow/Water Table, groundwater flows from north to south approaching Lake Ontario. Within the northeast extent of DNNP, which lies north of the CN railway, inferred groundwater flow is toward the east. General flow patterns within the Interglacial Deposits and Shallow Bedrock hydrostratigraphic units are similar to the Shallow/Water table. From the Shallow/Water Table, there is a downward vertical hydraulic gradient to the lower Interglacial Deposits and Shallow Bedrock hydrostratigraphic units. Hydrogeological site characteristics are discussed in further detail within PSAR Section 2.8 [R-6].

Since the 2020 PRSL renewal application [R-16], OPG has examined groundwater flow characteristics at the DN site as part of annual groundwater monitoring and a geotechnical investigation conducted in the DNNP's onshore power block area in 2021. The findings of these studies were consistent with the conditions described in previous licence applications.

Groundwater Quality

Updated groundwater baseline data (up to and including 2018) was reviewed as part of the 2020 PRSL renewal application [R-16]. Annual groundwater monitoring for the years 2019, 2020 and 2021 analyzed for tritium, benzene, toluene, ethylbenzene, xylene, and petroleum hydrocarbons (PHCs). Groundwater quality remains consistent with that documented in previous licence applications. One location exhibited elevated concentrations in 2020 of PHCs F2 and F3 above the 2011 Ministry of Environment, Conservation and Parks (MECP) Table 3 guidelines [R-78]. However, 2021 sample results from the same location were well below the applicable standards.

Based on the assumption that groundwater pumped during construction or in the long-term will be discharged to the natural environment, as part of the 2021 geotechnical study [R-25] for the DNNP's onshore power block area, groundwater samples were collected and submitted for analysis for comparison against the MECP's Provincial Water Quality Objectives (PWQO) Table 2 – Table of PWQOs and Interim PWQOs [R-30]. Select groundwater samples exhibited elevated concentrations of total metals, dissolved metals, phenols, and toluene above the selected PWQO. Several samples exhibited pH outside the acceptable PWQO range of 6.5 to 8.5. However, given that the water is not used for drinking and is not considered potable, these results do not change the conclusion of the original site evaluation.

No significant differences from the conditions described in previous licence applications have been identified for the hydrogeological environment of the DNNP site.

3.1.6 Biological Data

Additional terrestrial and aquatic baseline data has been collected through a variety of studies conducted since the 2020 PRSL renewal application [R-16].

Surveys for species at risk (Eastern Meadowlark, Bobolink, Barn Swallow, Least Bittern, Bank Swallow, and Bats), amphibians, reptiles, breeding birds, and pond biodiversity were conducted in the DNNP Site Study Area as part of the DN biodiversity program, providing updated information on these species.

A rare plant survey was conducted at the DN site in 2020 to update the internal rare plant management and pond decommissioning plan.

In 2021, bat snag and acoustic surveys were undertaken within DNNP lands and an area east of the Bowmanville switchyard to determine the occurrence of endangered bats and to quantify and characterize the type of bat habitat. Several species of bats, including Little Brown Myotis, Northern Myotis, Eastern Small-footed Myotis, and Tri-coloured Bat (species at risk provincially and federally) were recorded at the DN site for the first time during the 2021 monitoring. The documented occurrence of these species does not impact the conclusions of the original site evaluation, as any impacts to provincial species at risk will be addressed via the implementation of beneficial actions for the species at risk, as part of the provincial Endangered Species Act (ESA) permit. Overall, the updated baseline terrestrial data indicates that the terrestrial environment remains similar to that described in previous licence applications.

In 2022, a fish habitat assessment was completed for the channelized drainage feature traversing the DNNP lands west of Old Holt Road to Lake Ontario, which provides surface drainage for Hydro One lands north of 2nd Line Road and OPG land south of 2nd Line Road [R-5]. It was concluded that the drainage feature does not meet the definition of fish habitat as defined under subsection 2(1) of the *Fisheries Act* [R-15] as the drainage feature does not support fish and there is no connectivity to fish bearing waters (i.e., Lake Ontario).

In summary, no significant differences have been identified for the terrestrial and aquatic environment and the conclusions of the original site evaluation remain unchanged.

3.1.7 Radioactivity and hazardous substances

Baseline ambient radioactivity

Baseline radiation and radioactivity comprises the following environmental subcomponents:

- Atmospheric environment, including gamma radiation, gaseous radioactivity and radioactive particulate in air and precipitation;
- Surface water environment, including radioactivity in Lake Ontario, local streams and nearby municipal water supply plants;
- Aquatic environment, including radioactivity in sediments and fish;
- Terrestrial environment, including radioactivity in vegetation, animals and foods;
- Hydrogeological environment, including radioactivity in soils, shallow wells recharged with precipitation and deep wells;
- Radiation dose to members of the public; and
- Radiation dose to workers, including radiation doses to nuclear energy workers and other workers on the DN site.

Baseline radiation and radioactivity includes natural background, background from anthropogenic sources (fallout from nuclear testing and releases from other nuclear sites) and releases from the DNGS. OPG monitors radiation and radioactivity through its Environmental Monitoring Program (EMP) for the DN site. The results of the EMP are published annually and made available to the public. Based on the results of the 2021 EMP [R-17] the annual public dose resulting from the DN site was 0.6 uSv (represented by the adult of the Farm Critical Group) which is < 0.1% of the regulatory limit of 1,000 µSv/year for a member of the public. From 2016 to 2021, public dose estimates for the critical groups in the vicinity of the DN site are at most approximately 0.08% of the regulatory public dose limit of 1,000 µSv/year and 0.06% of the dose from background radiation (1,400 µSv/year).

Overall, the baseline conditions for ambient radioactivity have remained similar to those presented with the original application. Therefore, the conclusions of the original site evaluation remain unchanged.

Pre-existing hazardous substances

The original PRSL application documents identified areas on DNNP that are potentially contaminated with non-radioactive substances [R-9]. OPG subsequently conducted remediation and decommissioning activities for these areas (including the spoils disposal area, former DNGS concrete plant, and sandblast grit storage area).

OPG completed a soil characterization program in 2021. Sampling locations were chosen based on evaluation of current and historical use of DNNP land, a comparison of existing soil quality data against current standards and criteria, and the identification of areas of potential concern. Results of the soil characterization study identified the presence of PHCs, metals, hydride-forming metals, and other regulated parameters in soil at concentrations marginally above the MECP Table 3 standards [R-78]. The results are further discussed in the DNNP Site Evaluation Update Summary Report [R-5].

Overall, baseline conditions for pre-existing hazardous substances have remained similar to those presented with the original PRSL application and therefore there is no impact to the conclusions of the original site evaluation

3.1.8 Land use

OPG remains active in monitoring land use within 10 km of the DN site, including the review of planning applications. The focus of the monitoring is to determine whether there are any proposed land uses that would be of concern from the perspective of sensitive land uses such as day care, hospital, or retirement home locating within the vicinity of the DN site.

The review and update show that new developments are concentrated in existing urban areas (Oshawa, Courtice, Bowmanville, and Newcastle). This pattern of growth and development is consistent with the latest provincial plans, which seek to focus urban growth within existing urban areas, while maintaining limited development with the Greenbelt Plan and Oak Ridges Moraine.

Overall, there have been no substantive changes to the land use environment since the 2009 land use effects assessment was completed.

3.2 Evaluation of Natural External Events

3.2.1 Climate change

Historical assessment and future climate change predictions were performed in support of the original PRSL application. Key climatic parameters evaluated in the application included temperature, precipitation, lake water levels, and wind speeds.

For those changes that might in the future pose any hazard, mitigation will be implemented through designed barriers as committed by OPG in the DNNP Commitments Report, specifically for increasing temperatures, predicted changes in precipitation, and for lake water level [R-2].

3.2.2 Meteorological hazards

Temperature: The extreme temperature conditions defined for DNNP are +40 °C and -40 °C. The design of DNNP will demonstrate that it can mitigate the impact associated to extreme temperatures.

Humidity: The mean relative humidity in this region ranges from 65% to 80% throughout the year. Humidity evaluations showed no indications of extreme conditions that would require design mitigation.

High Winds: The average wind speed measured is 8.6 km/h, and the maximum wind speed recorded is 64 km/h for a 100-year return period. Wind is not expected to be an issue at the DNNP site since it is far less than the design basis wind speed of 232 km/h.

Wind Gusts: The readings from the closest station to the DN site measuring wind gusts, located at the Toronto Islands, are generally below 120 km/h, with the maximum historical reading of 154 km/h.

The site-specific 3-second gust wind speeds were estimated for various return periods (once in 10 to 1000 years), based on the data from four meteorological stations located between 50 to 100 km surrounding the DN site. These gust values are being used in the design to mitigate the potential hazard.

Tornadoes: In the case of a tornado, the design basis limit for DNNP is the maximum wind speed of 321.8 km/h (F4 Fujita Scale).

Hurricanes: There is a very low probability of occurrence for one to impact the DN site where the DNNP is located. The probable maximum tropical cyclone would be unlikely to yield gusts of more than 100 km/h, significantly lower than that of the design basis tornado.

Waterspouts: There is limited information on waterspouts in the vicinity of the DN site. Since the DNNP design has considered the hazards associated with design basis tornadoes, waterspouts are indirectly included as they are tornadoes over water.

Windborne Debris: An analysis of windborne debris from various categories of high wind is assessed as part of a high wind Probabilistic Safety Assessment (PSA) for DNNP. The assessment evaluates the impact of windborne debris on severe core damage and large release analysis.

Dust and Sandstorms: Neither dust storms nor sandstorms were identified as a potential hazard since the possibility of occurrence for these phenomena at the DN site is deemed to be highly unlikely.

Precipitation: The nearest monitoring stations to the DN site are located in Toronto Island and the city of Oshawa. The extreme daily precipitation predicted over a 100-year return period is 79.3 mm for the Toronto Island, and 88.6 mm for Oshawa. These precipitation predictions are significantly below the Probabilistic Maximum Precipitation value, defined as a 12-hour precipitation equivalent to 420 mm of total rainfall.

Snow / Snowpack: Similar snowfall conditions to the ones experienced at DNGS are expected to occur at DNNP due to their proximity. The DNNP highest snowfall tends to occur in January with a mean of 8.6 cm. Limiting value for snow and ice load is 3.0 kPa. As such the design of the facility must consider snowpack hazards.

Freezing Rain: Freezing rain hazard was considered for evaluation and assessment in the original PRSL application. This hazard was determined to be very low risk for DNNP due to low consequence.

Ice Storms: Ice-storms present a potential hazard for the systems located outside DNNP. However, the ice storm hazards are bounded by the Loss of Preferred Power (LOPP) event, which does not result in any temperature or pressure transient in excess of the acceptance criteria for fuel, pressure vessel, or containment.

Lightning: The DN site will likely experience a frequency of 2 to 3 cloud-to-ground flashes per year per square kilometre. This represents a very low risk for the DNNP due to low consequence.

Other Hazards: Avalanches, meteorites, geomagnetic storms and flares are very low risk for DNNP due to the low probability of occurrence.

3.2.3 Surface water hazards

In the 2009 Flood Hazard Assessment for DNNP, potential flood hazards were considered for a Probable Maximum Flood leading to surface water hazards from runoffs, storm surge, seiche, tsunamis, wave and wind effects etc. The design of the DNNP will ensure that all identified potential flood hazards are mitigated.

3.2.4 Groundwater hazards

Groundwater flow and groundwater hydrology was assessed as a part of the original PRSL application. The groundwater level for the design of the reactor foundations will be less than the highest level of 1 m below the final site grade. The DNNP RB structure is designed to withstand the effects of the highest groundwater levels specified for the plant, in conformance with REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants* [R-21].

3.2.5 Geotechnical and geophysical hazards

In support of the original PRSL application, the geotechnical and geophysical aspects, such as reactor foundation and other main earth structures, were assessed for extreme conditions including site seismic hazard, flood from surface water and high waves from Lake Ontario. The reactor foundation and these earth structures will be designed and constructed on the competent soils and bedrock present at the site to withstand the anticipated impact of the extreme conditions considered.

A geological investigation was conducted for the DNNP's onshore power block area in 2021. The findings were consistent with the conditions described in previous licence application documents.

3.2.6 Seismic and geological hazards

Earthquakes: The original seismic hazard curve used to characterize the DNNP site in 2009 bounds the recently updated DNGS curve, which is deemed applicable to the DNNP site due to their proximity. As such, the conclusion of the 2009 Seismic Hazard Assessment remains valid today.

Tsunami: Tsunami occurrences in Canada are rare and occur along coastlines with significant seismic risk (e.g. British Columbia coastline). The Great Lakes are on the edge of the Canadian Shield, a geologically stable, mid-continental region. In the absence of tsunami reports in Lake Ontario and the lack of shoreline or lakebed evidence of tsunami initiators, tsunamis are considered improbable events for the DN site.

Seiches: The maximum storm-induced surge and seiche at the Darlington shore is 0.75 m. Since historical evidence indicates that seiche events have occurred in Lake Ontario, shoreline protection at DNNP will be considered as part of the facility design.

Dams and Landslides: There are no man-made water retaining structures within the Darlington Creek watershed or other site vicinity watersheds. Hence, there are no flooding hazards associated with seismically induced failure of man-made water retaining structures.

Liquefaction Potential of Foundations: The RB foundations will be founded on sound limestone bedrock. Foundations of other structures that are not founded directly on bedrock, will be founded on competent till deposits, and/or engineered fill. Based on the currently available data and pending final confirmation from ongoing investigations, the liquefaction potential of foundations is considered to be very low.

Surface Faulting: There are no active surface faults or tectonic plates in the vicinity of the DNNP site. Therefore, there is no hazard from surface faulting at DNNP site.

Volcanic Hazards: There are no volcanic structures or active volcanoes in the farthest vicinity that impact the DNNP site. Therefore, volcanic hazard is not a potential hazard to the DNNP site.

3.2.7 Biological hazards

Biological hazards originating from Lake Ontario have the potential to contribute to biofouling of cooling water and service water supply systems. This in turn may contribute to loss of cooling capacity. The primary species that contribute to biofouling are algae, mussels, fish, and other microorganisms. For that reason, biofouling will be one of the key considerations in the design of cooling intake structure and cooling water system.

Additionally, wildlife such as birds and other airborne non-human biota such as a flock of geese or a swarm of insects could potentially block the screens of the DNNP MCR air ventilation intakes. However, the DNNP MCR air ventilation system is designed to operate at 100% re-circulation mode in order to mitigate the impacts from this hazard. The existing design provisions do not compromise the MCR ability to control the reactor and no controlled shutdown is required.

3.2.8 Natural fire hazards

Natural fire hazards such as forest fires do not pose an incremental risk to the site due to very low probability and low potential consequence of fire. The land coverage surrounding the DN site has minimal vegetation and is mostly paved in the areas proximal to site buildings. Therefore, external natural fire hazards does not pose a risk.

3.3 Evaluation of External Non-malevolent Human-induced Events

3.3.1 Aircraft Crash Events

The occurrence of large aircraft crash has been screened out due to its low frequency. Additionally, large military aviation accidents are not a concern for the DN site as there are no military aircraft flying in the vicinity of the Bowmanville airspace.

The small aircraft crash is screened out for DNNP on the basis that the facility will be designed to withstand site-specific tornado missiles, which are considered to bound the impact resulting from small aircraft crash. Likewise, accidents involving small light weight drones are also screened out as this type of event is also bounded by the tornado missile event.

3.3.2 Other Transportation Hazards

Rail Transportation Hazards: The accidents could result in cold or hot toxic gas releases, vapour cloud explosions (VCEs), boiling liquid expanding vapour explosion (BLEVE), and other types of explosions. Toxic chemicals such as chlorine and ammonia gas, could potentially be transported through wind. In a recent study for DNNP rail transportation accidents were determined not to be a threat to the design basis of the facility.

Road Transportation and Traffic Accidents: As the DNNP site is located close to Highway 401, road transportation accidents could have an impact on DNNP. The event scenario considered consists of the crash or rollover of two tractor trailers on the highway, leading to the release of toxic chemicals into the atmosphere. The potential consequences of road transportation accidents are deemed to be much less severe and bounded by rail transportation accidents.

Marine Transportation Accidents: Marine transportation accidents in Lake Ontario could involve large tankers, cargo or container ships. The potential hazards from these accidents include the release of toxic chemicals, explosive hazards, and physical damage to DNNP's water intake structure. The consequences of toxic chemical releases and explosions on a cargo ship, are similar to those events occurring due to train derailment accidents. The missiles resulting from explosions are bounded by the tornado missiles, while the chemical releases are screened out for DNNP on the basis that the appropriate mitigating measures for similar train derailment accidents are in place. The physical damage resulting from marine hazard events will be similar to the events causing loss of service water due to frazil ice or algae. Therefore, they are not a concern given that the cooling water arrangement for DNNP will be similar to the one for DNGS.

3.3.3 Fires and Explosion

Stationary non-nuclear accidents can result in accidental fire, explosions, and other industrial hazards. These potential hazards are assessed below.

Fire: The land immediately surrounding the site is expected to have minimal vegetation and to be mostly paved to prevent the possibility of external fires. Therefore, the propagation of fire to the site can be screened out. Additionally, fire hazards associated with pipeline ruptures can also be screened out as the pipelines in the vicinity of the DNNP site are beyond the screening distance.

Explosions: Explosions in the vicinity of the DN site have the potential to generate missiles and shockwaves, which could damage SSCs. However, the DN site is located beyond the screening distance from the potential sources of BLEVE and VCE explosions. In addition, DNNP is designed to be protected from missiles generated by explosions.

Industrial Hazards: The primary industrial hazards of concern are underground blasts associated with the St. Mary's Cement plant. The hazards could consist of larger shock waves which could result in turbine trips or failure to detect the larger shock waves. This is taken into account in the plant design.

3.3.4 Chemical and Radiological Hazards

Evaluation of On-site Methane Hazard: During initial site investigation, naturally occurring gas (methane) was found at/or near the bedrock/overburden interface in several boreholes. Methane gas will be monitored during excavation near the bedrock/overburden interface, and the required precautionary measures will be implemented during construction.

Evaluation of Chlorine and Ammonia Hazard: The event scenario involves a local leak in one of the water treatment plants or in the St. Mary's Cement Plant, resulting in the release of the chlorine gas (Cl_2) or ammonia (NH_3). Based on the screening distances calculated for the identified toxic chemicals, the chlorine hazards are screened out for DNNP since the potential sources are further than 5 km away from DNNP. However, the DNNP site will likely be within the screening distance (0.9 km) for the potential ammonia hazard associated with a release at the St. Mary's Cement Plant.

This hazard has been considered in the design of the MCR and SCR for DNNP to ensure that the habitability systems detect and protect control room personnel from toxic gases. The MCR habitability requirements are satisfied without the need for individuals to use individual breathing apparatuses or special protective clothing. Therefore, this hazard can be screened out for DNNP.

Evaluation of Darlington Nuclear Generating Station Hazard: Since the DNGS will be adjacent to DNNP, the ability to maintain operation of the DNNP can be potentially affected during a nuclear accident within the DNGS. In such an accident, it will be critical to be able to shut down the DNNP reactor safely. This is done by maintaining functionality of the MCR to ensure the safety of operators who will remain in the MCR for safe shutdown of the reactor following the accident

The control rooms are shielded against sources of radiation. Both the MCR and SCR have habitability design features to ensure that the control room operators will be protected against airborne radioactivity. Control room operators will not receive radiation exposures in excess of the applicable regulations. Additionally, the MCR and SCR include all instrumentation and controls necessary during safe shutdown of the plant.

Evaluation of Pickering Nuclear Generating Station Hazard: Based on DNGS hazard screening analysis, the accidental release of radioactive materials at PNGS, which is 25 km west of DNGS, can be screened out given that it is a slow developing event, there are mitigating features and enough time for operator to take proper action.

Evaluation of Hazards from Other Nuclear Facilities within the DN Site: The accidents from other nuclear facilities within the DN site, including the Tritium Removal Facility (TRF), the storage area of active liquid waste at DNGS, and the DWMF, could also have an impact on the operating staff, and in particular, the MCR operators. However, since the control rooms are shielded against sources of radiation, the MCR and SCR have habitability design features to ensure that the control room operators will not receive radiation exposures in excess of the applicable regulations

3.3.5 Electromagnetic Interference Hazards

The Electromagnetic Interference (EMI) sources such as high voltage transmission lines are an integral part of the power plant and are constantly present. These can affect the functionality of electronic I&C equipment, and such disruptions could lead to impairment of critical plant control signals.

The industry standards for EMI have been well established for transient immunity and/or protection. The DNNP I&C equipment supporting safety class systems are being designed to be electromagnetically compatible in conformance with the industry standards.

3.4 Assessment of Site Suitability

3.4.1 Evaluation against the CNSC safety goals:

As per REGDOC-2.5.2 [R-21], the safety goals used to evaluate the reactor designs include Core Damage Frequency (CDF), Small Release Frequency (SRF), and Large Release Frequency (LRF). A comparison of hazard frequency for DNNP with the numerical CNSC safety goals shows the frequencies estimated for CDF, SRF and LRF are significantly less than the safety goals. Therefore, it is concluded that the DNNP BWRX-300 design meets the CNSC requirements.

3.4.2 Evolving natural and human–induced factors

Hazards could evolve with time. Hazard evaluations for DNNP have been performed periodically at different stages of the project. Specifically, the DNGS hazard screening assessment updated in 2019 was used as a starting point for hazard re-evaluation in support of the PRSL Renewal for DNNP [R-16] due to the close proximity of DNNP site to DNGS. More recently, the DNNP-specific hazard screening, with the consideration of the BWRX-300 design, has been performed as part of OPG's commitment [R-2]. The hazard screening for DNNP will be updated periodically throughout the plant lifetime, as required by CNSC REGDOC-2.4.2, *Probabilistic Safety Assessment (PSA) for Reactor Facilities, Version 2* [R-40].

3.4.3 Hazards associated with external events

The following external hazards were reviewed and assessed for DNNP:

- Meteorological hazards;
- Surface water hazards;
- Ground Water hazards;
- Geotechnical and geophysical hazards;
- Seismic and geological hazards;
- Biological hazards;
- Fire hazards (External naturally occurring); and
- External non-malevolent human-Induced hazards; and
- Radiological hazards

The assessment took into account any changes in applicable codes, standards and regulations since the original assessment was conducted. The assessment confirmed that the site is suitable for a NGS with GEH's BWRX-300 reactor design and the hazards identified are acceptably low or will be mitigated through design.

3.4.4 Population and Emergency Planning

Section 4.10 of this Application document provides details on population and emergency planning.

3.4.6 Consideration of future life-extension

Considerations of life extension for the new NGS with the BWRX-300 reactor design will be performed at an appropriate stage in the future of the DNNP life cycle.

3.4.7 Security considerations

The site evaluation in terms of nuclear security confirmed that the PRSL basis remains valid. All security-related commitments under the PRSL were submitted and accepted by the CNSC as part of the site preparation commitments.

OPG has conducted a construction threat risk assessment [R-109]for the DNNP BWRX-300 on the DN site and confirmed that the site remains suitable, and any security risks can be effectively mitigated.

3.5 Exclusion zone determination

The exclusion zone is one component of "defence in depth" strategy, an international nuclear safety concept adopted by the CNSC and OPG for the DNNP.

The exclusion zone has a role in supporting the safety goals with respect to the protection of individual members of the public in the event of a major accident at a nuclear generating station. Specifically, permanent dwelling within the exclusion zone is prohibited in order to ensure significant dispersion of any potential radioactive release before it reaches human habitat. In addition, the exclusion zone is defined in such a way that the public can be quickly evacuated in the case of a nuclear emergency.

For BWRX-300 design, the following factors were considered in the determination of the exclusion zone:

- Land usage needs;
- Security requirements;
- Evacuation needs;
- Environmental factors; and
- Dose Acceptance Criteria.

For the BWRX-300 design the proposed exclusion zone is 350 m from the RB walls.

3.6 Conclusion

The overall conclusion of the updated site evaluation is that the DNNP site is suitable for a new NGS with GEH's BWRX-300 BWR design and would not pose any unreasonable risk to the public, personnel or environment. Further details about the site evaluation are provided in the Site Evaluation Summary Report [R-5] and PSAR Chapter 2 [R-6].

4.0 Safety and Control Measures

4.1 Management System

The Management System SCA ensures that adequate processes and programs are implemented to ensure OPG achieves its safety objectives, continuously monitors its performance against those objectives, and fosters a healthy safety culture.

4.1.1 General Considerations

OPG has a mature and effective Nuclear Management System (NMS) for its existing nuclear operating facilities, which complies with CSA N286-12, *Management system requirements for nuclear facilities* [R-22]. The DNNP will utilize programmatic elements within this NMS applicable to the DNNP licensed construction activities to support the creation of a nuclear operating facility.

The objective of this NMS is to establish the business framework and processes to support OPG in achieving its safety objectives, ensuring effective monitoring of performance against those objectives, and fostering a healthy safety culture. The NMS framework consisting of programs, standards, and other governing documents and processes collectively ensure that safety is the foremost consideration in management decisions and actions.

Every employee in the organization is responsible and held accountable for complying with the expectations of the NMS, and for ensuring their actions are deliberate and consistent with protecting worker health and safety, the health and safety of the public, and the environment.

This Application outlines programmatic details of this NMS that will be applied to provide for the planned and systematic control of the DNNP licensed construction activities, while maintaining safety as the overriding priority. Project specific documents define the framework for detailing how DNNP programs meet the intent of management system governance and will transition as the project progresses.

OPG is the Owner and will be the Licensee for the DNNP licensed construction activities that will utilize a contract model that maximizes integration and collaboration between contract partners. OPG will have the accountability to ensure ongoing and intrusive oversight for the LTC phase following the OPG NMS.

The contract agreements describe the relationship and accountabilities of the contract partners. All contract partners executing work will meet the applicable requirements of CSA N286-12 [R-22] for the activities they will perform.

4.1.2 Management System

OPG's NMS is part of OPG governance that has a hierarchical structure taking authority from the OPG-POL-0032, *Safe Operations Policy* [R-43] and the N-POL-0001, *Nuclear Safety & Security Policy* [R-44]. OPG-POL-0032 defines the requirement to establish and maintain management systems to ensure that safe operation is the overriding priority in all activities performed at OPG facilities. This policy outlines requirements and accountabilities for the organization to foster high levels of operating performance and reliability, while ensuring compliance with all legal and regulatory requirements.

The N-POL-0001 establishes guiding principles for OPG nuclear employees stating that nuclear safety shall be the overriding priority in all activities performed in support of OPG nuclear facilities and that nuclear safety shall have clear priority over schedule, cost, and production. The policy requires that everyone demonstrate respect for nuclear safety and conduct themselves in a manner consistent with defined traits of a healthy nuclear safety culture. In accordance with this policy, the Chief Nuclear Officer (CNO) is accountable to the Chief Executive Officer (CEO) and the Board of Directors to establish a management system that fosters nuclear safety as the overriding priority.

The NMS is established by N-CHAR-AS-0002, *Nuclear Management System* [R-45] taking authority from OPG-POL-0032 and N-POL-0001. This nuclear charter gives authority to the programmatic governance and details the governing document framework that satisfies the requirements of CSA N286-12 [R-22] and REGDOC-2.1.1, *Management System* [R-46]. The charter also provides direction regarding administration of nuclear licensing activities and establishes requirements to which OPG nuclear and interfacing organizations shall comply.

The charter states that every employee in the organization that executes and supports licensed activities associated with OPG nuclear facilities, and every person or entity that supplies a product or service is responsible and shall be held accountable for complying with the expectations of the charter and referenced programs, and for ensuring their actions are deliberate and consistent with protecting worker health and safety, the health and safety of the public, and the environment.

The accountabilities of the NMS are defined in the charter from overall accountability at the CNO level down to employees. The CNO is accountable for the establishment, implementation of, and effectiveness of the NMS, and ensures that a foundation of leadership exists to hold those within nuclear and interfacing organizations accountable for the implementation of, and adherence with, the NMS.

With the support of employees and the effective implementation of the NMS, OPG assures compliance with the N-POL-0001 by all who may have an impact on nuclear safety.

As the DNNP is not yet an operating nuclear facility, the DNNP will use a graded approach and only the programs applicable to the licensed activities would be in effect while others will become applicable as the project progresses. The applicable programmatic elements of the OPG NMS for the DNNP are described throughout this Application and listed in Table 4.1-1.

Table 4.1-1: Management System Documents for the DNNP Construction Phase for all SCAs

Document	Title
N-PROG-AS-0001	Nuclear Management System Administration
N-PROG-AS-0002	Human Performance
N-PROG-MP-0009	Design Management
N-PROG-MP-0014	Reactor Safety Program
N-PROG-RA-0001	Consolidated Nuclear Emergency Plan
N-PROG-RA-0002	Conduct of Regulatory Affairs
N-PROG-RA-0003	Performance Improvement
N-PROG-RA-0010	Independent Assessment
N-PROG-RA-0011	Nuclear Security
N-PROG-RA-0015	Safeguards and Nuclear Material Accountancy
N-PROG-TR-0005	Training
OPG-PROG-0001	Information Management

OPG-PROG-0005	Environment Health and Safety Managed Systems
OPG-PROG-0009	Items and Services Management
OPG-PROG-0037	OPG Business Planning
OPG-PROG-0039	Project Management
OPG-PROG-0042	Cyber Security
W-PROG-WM-0003	Decommissioning Program

To support the unique aspects of the BWRX-300 operating plant, programs to support the operations phase will incorporate the details and requirements of this technology. These programs will be phased in during fuel-out commissioning and be fully in effect for the Licence to Operate (LTO). All operational programmatic elements will comply with CSA N286-12 and REGDOC-2.1.1.

DNNP Licence to Construct Management System Report, NK054-REP-0813-00004 [R-47] further outlines program applicability of OPG's NMS to the proposed DNNP licensed construction activities.

The following describes applicable programs given authority from N-CHAR-AS-0002 [R-45] that support the implementation, assessing and continual improvement of the OPG NMS that apply to the DNNP.

The N-PROG-AS-0001, *Nuclear Management System Administration* [R-48] program receives authority from N-CHAR-AS-0002, provides the business framework and assigns roles and accountabilities for the effective implementation of the NMS. This program also establishes tools and processes that address both internal and external factors to support continual improvement and confirm the effectiveness of the NMS. Processes are created such that all applicable regulatory requirements and codes and standards are embedded and integrated within the NMS including aspects of health, safety, environment, security, economics, and quality.

The N-PROG-RA-0010, *Independent Assessment* [R-49] program provides independent assessment processes to perform comprehensive and critical evaluation. This program also ensures the NMS is reviewed with sufficient frequency to confirm its continuing effectiveness.

The N-PROG-RA-0003, *Performance Improvement* [R-50] program consists of organizational learning tools and provides a framework to guide development, assessment, and improvement of OPG performance, including DNNP. The framework of processes includes a means to prevent, detect and correct adverse conditions as well as facilitate the communication of lessons learned. This program covers the key areas of performance improvement, such as corrective action, self-assessment, benchmarking, OPEX, and nuclear safety culture.

The OPG-PROG-0001, *Information Management* [R-51] program establishes a framework and system for managing OPG's information. Supporting implementing procedures for records and document management define the record management process and associated requirements to be met. These documents are implemented to meet the records management requirements identified in nuclear regulations and codes.

4.1.3 Organization

OPG's organizational structure is established to optimize intrusive oversight and project management of all contract partners for the DNNP licensed activities. It aligns with a contract model which allows for increased integration and collaboration between the contract partners: Owner (OPG), Developer (GEH), Constructor and Architect Engineer. Owner oversight is maintained separately from the integrated teams to ensure independence and ultimate accountability as the licensee. A visual representation of the contract partner team is provided in Figure 4.1-1 below.

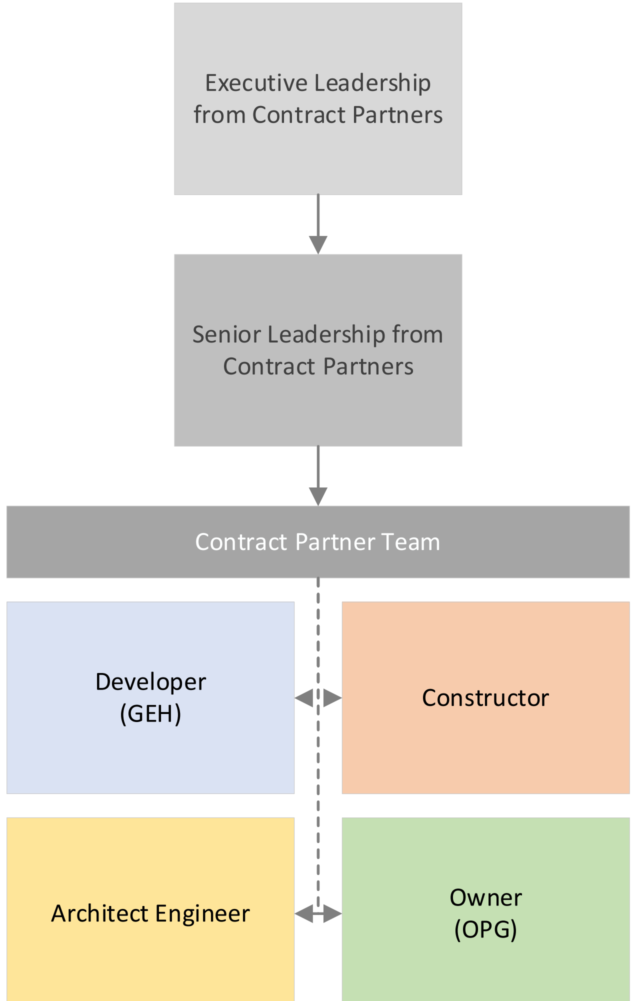


Figure 4.1-1: Contract Partner Team

Figure 4.1-2 shows the current organizational structure for the DNNP organization.

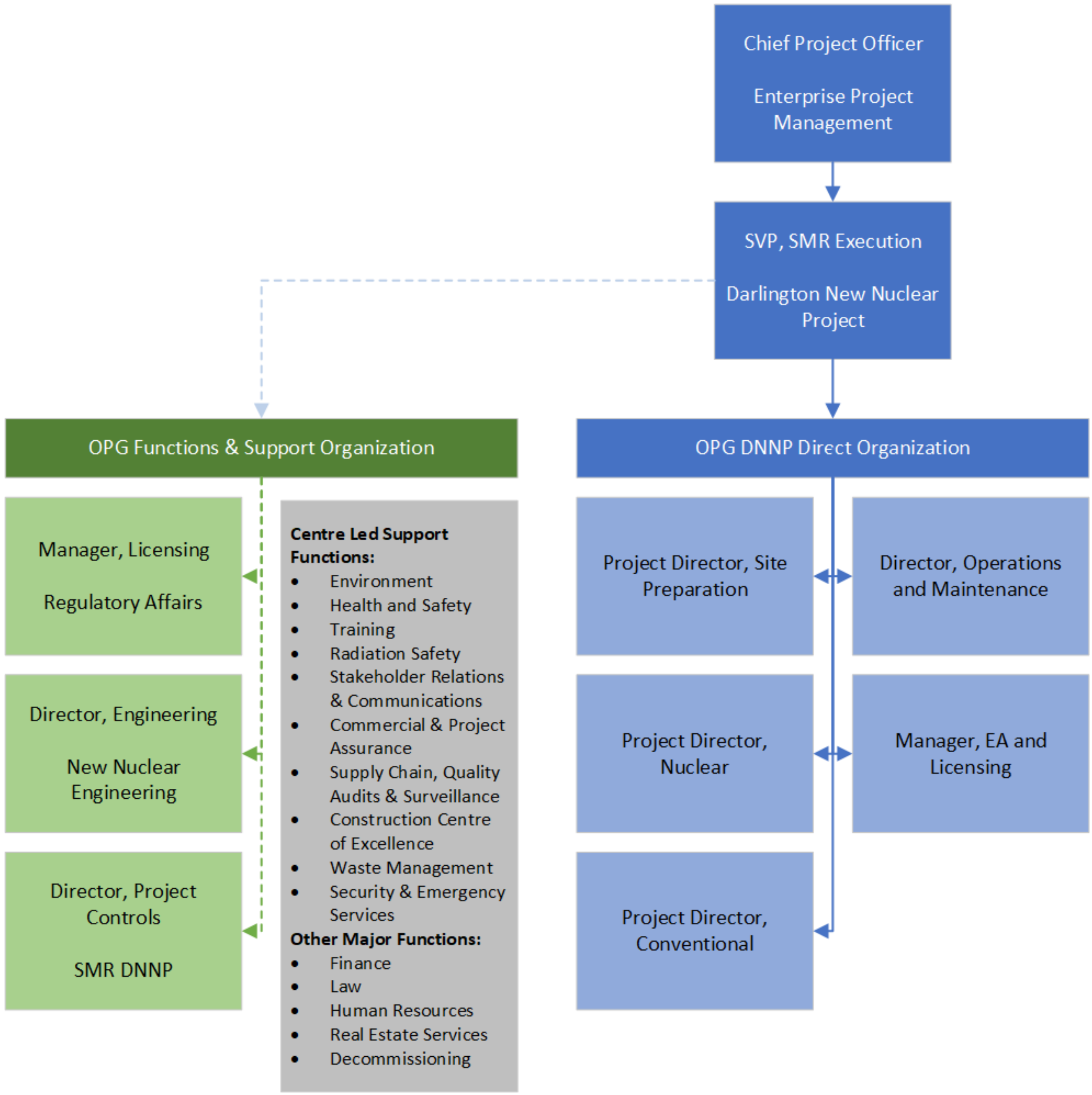


Figure 4.1-2: Organizational Structure for DNNP Organization

To ensure the overall project is successful, a contract model is being utilized to maximize integration and collaboration with the Developer and supporting contract partners.

OPG is working collaboratively with the Developer and will continue to be intrusive in the design and safety analysis process, and products. Project decision making will be prioritized at the appropriate level with escalation of decisions if contract partner members are not able to agree.

OPG will retain overall accountability for the licensed activities conducted during the LTC phase. GEH, is the Design Authority (DA) for the powerblock (nuclear and conventional systems in the powerhouse), the intake structure, and switchyard systems and structures. GEH will maintain DA through the design, construction, inspection and testing, and fuel-out commissioning phases. The turnover strategy describes the transition of the DA to OPG prior to plant start-up in accordance with REGDOC-2.5.2 [R-21]. OPG is the DA of the areas outside of GEH scope.

Staffing and resourcing needs will vary over the lifecycle of the project. DNNP will be staffed with a combination of permanent OPG and temporary, augmented, or seconded staff to fulfill project needs throughout its lifetime. OPG resource management and planning is reviewed and reconfirmed on an annual basis as part of the business planning cycle. Onboarding of staff onto the project is done in accordance with established OPG human resource processes.

Training and development of staff is prioritized on the project to ensure sufficient in-house capabilities are built and maintained for the life cycle of the facility. Knowledge development, transfer, and retention plans are in place as applicable to ensure independent and intrusive oversight is maintained during the design and construction phase of the project and to increase technical and operational knowledge of the BWRX-300. Where OPG does not have existing required experience and proficiency, especially in any areas with the potential to impact nuclear safety, third party reviews are being performed in parallel to OPG reviews to ensure high quality design products and build the knowledge within the OPG team.

OPG's Contract Partner Team for DNNP Construction

OPG is recognized internationally for excellence in nuclear operations, and in nuclear project execution as a result of the successful execution of the Darlington Refurbishment Project. This experience puts OPG in a notable position to execute this new nuclear deployment. Working with GEH, OPG is deliberately putting in place a very strong, competent, and qualified team to execute this project, OPG as the Owner and Licensee is accountable to maintain intrusive oversight and project management of all aspects of the project with increased focus when nuclear safety could be impacted. OPG's management system ensures contract partners are qualified to perform designated scope throughout the life cycle of the project per CSA N286-12 [R-22] requirements and will ensure contracted work is carried out to the required level of safety and quality consistent with REGDOC-2.3.1, *Conduct of Licenced Activities: Construction and Commissioning Programs*. OPG is an informed customer that has developed and continues to increase the in-house knowledge required to provide on-going oversight and technical challenge to ensure licence and regulatory requirements are met. OPG will ensure sufficient qualified workers for nuclear safety related positions are in place. OPG has established owner's requirements and will monitor for adherence.

OPG has chosen a world-leading, highly experienced reactor designer in GEH, which has designed BWRs that operate around the world. The BWRX-300 design is an evolution of GEH's previous BWR designs with the most recent being the ESBWR, which is certified by the USNRC.

GEH has the role of the DA for the powerblock, intake structure, condenser cooling water and switchyard scope of the project. This includes the responsibility to meet owner requirements, licence and regulatory requirements and compliance with identified applicable codes and standards. Activities to be performed by GEH include design, procurement, and commissioning support.

Also involved in facility design will be an Architect Engineering (AE) firm. The AE will be responsible for development of the conventional aspects of the design work. The activities associated with the AE scope include design, procurement, field engineering and commissioning support with oversight by GEH as the DA and OPG as the Owner.

The selected Constructor will be an experienced nuclear construction firm with a practiced nuclear construction quality program, a strong safety culture, and proven expertise in large projects. The constructor will be accountable for construction design reviews, construction planning, procurement, construction execution, and commissioning activities.

OPG is a highly experienced, qualified, competent, and excellent nuclear operator. However, in recognition of the value of learning from others, OPG has established a

Memorandum of Understanding (MOU) with a highly qualified and competent nuclear operator in the United States, Tennessee Valley Authority, which itself operates three BWRs at Brown's Ferry. Through this relationship, OPG will gain valuable advice and OPEX which will inform OPG's work throughout the project planning and construction periods and is expected to continue during operations.

By engaging other world-class expert companies across various aspects including design, engineering, construction, and operations, OPG is ensuring that design, construction, and safety of its DNNP BWRX-300 plant will meet regulatory requirements, incorporate industry-leading best practices, and establish best-in-class quality and safety.

4.1.4 Configuration Management and Change Control

This sub-section documents how OPG establishes and maintains consistency between design requirements (including safety analysis), physical and operational configuration, and configuration information (e.g., operating procedures or drawings) from initial conception until end of operating life. Within OPG, configuration management is applied via N-STD-MP-0027, *Configuration Management* [R-52] that is applicable to the DNNP.

N-STD-MP-0027 receives authority from N-PROG-AS-0001 which applies to programs under N-CHAR-AS-0002. This is important to ensure station physical configuration for Structures, Systems, and Components (SSCs) matches the configuration documents for all states of the plant, and ensures that configuration is maintained accurate, consistent, and readily accessible along with defining clear scope, responsibilities, authorities, and interfaces among organizations.

N-STD-MP-0027 complies with requirements of CSA N286.10, *Configuration management for high energy reactor facilities* [R-53] and CSA N286-12 [R-22]. Requirements are met through implementation of this standard into OPG programs, as they apply to the configuration management at the beginning of a new build project as well as configuration management from initial conception. The contract agreements describe the interface arrangements, relationship and accountabilities of the contract partners.

GEH will be accountable for configuration management and change control from design to commissioning for their scope of work as a DA. OPG is accountable for project management and oversight of GEH deliverables. The oversight activities will be applied using a graded approach such that higher level risk activities and deliverables will have more frequent and intrusive oversight. The oversight is intended to ensure ongoing conformance of the engineering products or deliverables in accordance with the GEH management system.

4.1.5 Safety Culture

Safety culture is applicable to all activities that may affect the health and safety of the workers and the public, and the environment in every phase of the facility's life cycle.

OPG promotes nuclear safety culture objectives in compliance with CSA N286-12 [R-22], REGDOC-2.1.2, *Safety Culture* [R-59], and REGDOC-2.3.1, *Conduct of Licenced Activities: Construction and Commissioning Programs* [R-54] through N-POL-0001 [R-44], and N-CHAR-AS-0002 [R-45]. In addition, OPG's NMS programs implement requirements and expectations for understanding and promoting a strong safety culture through N-PROG-AS-0001 [R-48], OPG-PROG-0005, *Environment Health and Safety Managed Systems* [R-60], N-PROG-AS-0002, *Human Performance* [R-61], and N-PROG-RA-0003 [R-50] programs.

The N-POL-0001 [R-44] establishes OPG's commitment from its employees to nuclear safety by stating:

> *"Nuclear Safety and Security shall be the overriding priority in all activities performed in support of OPG nuclear facilities. Nuclear Safety shall have clear priority over schedule, cost, and production*."

The policy also makes everyone accountable to conduct themselves in a manner consistent with the following nuclear safety & security culture traits:

Table 4.1-2: Nuclear Safety & Security Culture Traits

No.	Safety & Security Culture Traits
1	Personal Accountability
2	Questioning Attitude
3	Effective Safety Communication
4	Leadership Safety Values and Actions
5	Decision Making

6	Respectful Work Environment
7	Continuous Learning
8	Problem Identification and Resolution
9	Environment for Raising Concerns
10	Work Processes
11	Vigilance

The policy also commits OPG to conduct comprehensive, systematic, and rigorous safety culture assessments at least every five years.

Receiving authority from this policy, N-CHAR-AS-0002 [R-45] defines the framework that establishes the programs and processes required to ensure OPG achieves all safety objectives and continuously monitors performance against these objectives. This charter describes that every employee that executes or supports licensed activities and every person or entity that supplies a product or service for OPG's nuclear facilities is responsible and shall be held accountable for complying with the expectations within the charter and programs that make up the NMS.

N-PROG-AS-0001 [R-48] describes the integration of the NMS programs and processes used to demonstrate effective implementation and compliance with the requirements set out in CSA N286-12 [R-22] and the charter. This program includes nuclear safety oversight which summarizes the framework and accountabilities for programs as well as the processes used for oversight and assessment of nuclear safety.

OPG-PROG-0005 [R-60] governs the design and execution of OPG's system for protecting the environment and managing worker safety and the roles and responsibilities of the program. This program takes authority from nuclear and non-nuclear policies.

The N-PROG-AS-0002 [R-61] lays the groundwork for improving and sustaining worker performance. This program provides guidance to reduce the probability and consequence of human error associated with the worker-machine interface required to operate, maintain, and support a nuclear facility.

N-PROG-RA-0003 [R-50] consists of organizational learning tools and provides a framework to guide development, assessment, and improvement of facility performance including nuclear safety and security culture assessments and monitoring processes. An important expectation of OPG for the contractor's management systems is that they always demonstrate the attributes of a healthy nuclear safety culture.

4.1.6 Applicable OPG Documents

The OPG governance documents for the Management System SCA, which supports the licensing basis, are included in Table 4.1-3 below:

Table 4.1-3: Management System Document for the Management System SCA

Document	Title
OPG-POL-0032	Safe Operations Policy
N-POL-0001	Nuclear Safety & Security Policy
N-CHAR-AS-0002	Nuclear Management System
N-PROG-AS-0001	Nuclear Management System Administration
N-PROG-RA-0003	Performance Improvement
N-PROG-RA-0010	Independent Assessment
OPG-PROG-0001	Information Management
OPG-PROG-0005	Environment Health and Safety Managed System
OPG-PROG-0009	Items and Service Management
OPG-PROG-0037	OPG Business Planning
OPG-PROG-0039	Project Management

4.2 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 sufficient personnel are in place in all relevant job areas and have the necessary knowledge, skills, procedures and tools in place to safely carry out their duties.

4.2.1 General considerations

OPG's N-PROG-AS-0002 [R-61] program meets or exceeds all applicable regulatory requirements and related objectives. The goal of the Human Performance program is to continually reduce the frequency and severity of events through the systematic reduction of human error and the management of defences in pursuit of zero events of consequence.

In addition, OPG's N-PROG-TR-0005, *Training* [R-62] program or equivalent, provides the structure, processes, and tools for defining, developing, implementing, documenting, assessing, and improving the training required to ensure staff have the appropriate knowledge, skill, and attitudes for safe and efficient plant operations.

The principles of human performance will apply to the lifecycle of the DNNP facility throughout all stages of design, construction, operation, and decommissioning with a graded approach that is commensurate with the level of risk.

OPG workers, including contract staff, are qualified, and possess the necessary competencies to perform their work safely, meeting established standards and expectations set by OPG Management. Workers rely on OPG procedures or equivalent, which encompass a wealth of industry OPEX to execute tasks event free. Human performance is monitored using established processes and metrics.

OPG fosters a culture where individuals feel comfortable reporting mistakes without fear, in an atmosphere of healthy accountability and organizational learning. The fundamental principles of human performance are captured in training for all levels of the organization. Part of the training is understanding how individuals, the environment, organization, and technology interact, that is, the interaction between these categories is a framework for a systematic approach to support excellence in worker performance.

Events can be reduced by understanding the reasons mistakes occurred as well as learning from successes when things go right. OPG fosters a culture of open reporting of events following established processes for human performance event communications and analysis.

The description of resource management is provided in the NK054-PLAN-01210-00008, *DNNP – Program Management Plan* [R-31].

OPG expects that all workers apply themselves to exhibit nuclear professional behaviours. OPG will ensure human and organizational factors are considered throughout the design, planning and construction activities. For details please refer to PSAR Chapter 18 [R-6].

4.2.2 Personnel training

Personnel Training is governed under OPG's N-PROG-TR-0005 [R-62] program or equivalent. Training associated with the DNNP conforms with the requirements of REGDOC-2.2.2, *Personnel Training* [R-63].

The qualification and training requirements for personnel engaged in the design activities is described in Section 4.1.3 of this application.

This section describes the training system for use for personnel engaged in activities during the construction and fuel-out commissioning phases of the DNNP. It includes:

- Training for staff, supervisors, contractors and sub-contractors;
- Training specific to commissioning activities;
- Initial Training and Qualification programs for certified staff (including Simulator Training), Field operations personnel, control maintenance and mechanical maintenance, and associated OPG Instructors;
- Use of existing Nuclear Training Programs including revisions as a result of DNNP technology; and,
- Scheduling and timing of training program delivery.

The initial training and qualification programs will also support the certification of staff which will be completed as part of an application to operate the reactor, and prior to fuel-in commissioning.

Contract partners are responsible for the training and qualification of their staff, supervisors, contractors and sub-contractors under their own management system.

OPG provides personnel with site specific information where needed. Skilled trades staff are required to hold journeyperson status and certificate of qualification as appropriate. Apprentices work under the accountability of a journeyperson when performing tasks associated with the skilled trade.

Additionally, contract partners maintain records of staff certifications, licences, professional designations, training and qualification (applicable to activities conducted on the project) and make them available for OPG review upon request.

Any workers who require access to the Darlington NGS protected area and/or tie-ins to existing Darlington NGS station systems/equipment, will undergo training and qualification under the DNGS training program as required prior to commencement of work.

During the construction phase of the project, other training program activities includes:

- Development of DNNP specific initial training and qualification for certified staff, field operations personnel, control maintenance and Mechanical Maintenance
- Development and delivery of training specific to commissioning activities
- As a result of new technology, equipment, processes and procedures, revisions to the Nuclear Training Programs such as Nuclear General Employee Training, radiation protection, conventional health and safety, work protection code, engineering, applied nuclear physics and chemistry.
- Delivery of initial training and qualification programs for control room staff, field personnel, control maintenance and mechanical maintenance and associated OPG instructors to support fuel-out commissioning.

Note: Control Room staff may be selected from previously certified, currently certified at OPG or another NGS or currently enrolled in an Initial Certification Training program.

The revision, development and delivery of training for staff who will commission, operate and maintain the new station will follow N-PROG-TR-0005 [R-62] or equivalent, and all applicable standards and procedures taking authority from the program.

Development and delivery of DNNP specific initial training and qualification for workers who are expected to conduct licensed activities at the new nuclear facility will follow OPG's N-PROC-TR-0008, *Systematic Approach to Training* (SAT) [R-65] or equivalent. This includes the identification of performance requirements and definition of worker training through analysis, the design and development of training to support proper job performance and individual development, implementation of training and examination processes, as well as examination security, development, and approval processes.

Training for certification of staff required under a future operating licence includes use of a full-scope simulator. The full scope simulator is a replica of the BWRX-300 MCR hardware panels and instrumentation, telephone and communications systems, radiation and fire emergency tones and public address system. It includes the simulation computers and servers required to provide plant system modeling/simulator operation and Instructor Station functionality.

All operator related actions normally performed in the BWRX-300 MCR identified by the training analysis will be incorporated into the simulator capability. The simulator will be based on the final iteration of the plant design and the simulator and simulation models will support the training of certified staff.

Revisions that may be required for existing nuclear training programs as a result of DNNP will follow the existing systematic approach which provides training change management processes that systematically analyze procedural and equipment changes, changes in job descriptions, OPEX and training needs.

The NK054-PLAN-01210-00029 Sheet 2, *DNNP Construction and Commissioning Training Plan* [R-66] will be submitted as a supporting document to this application.

4.2.3 Personnel certification

As part of future operations, OPG will develop programs that ensure that persons seeking a certification or renewal of a certification issued by the CNSC for a designated position referred to in the DNNP operating licence are qualified to carry out the duties of that position in accordance with the Nuclear Safety and Control Act (NSCA) and the regulations made under the NSCA [R-37].

Certification of personnel is outside the scope of this licence, however, the sections below provide an overview of the programs and processes for certifying personnel at existing NGS. These will be adapted to support certification of personnel for the future operations at DNNP.

Personnel certification is governed under N-PROG-TR-0005 [R-62]. This program is mature and robust and conforms with applicable CNSC regulatory documents.

Required skills and competencies for certified positions are identified through OPG's SAT that is used to identify potential training changes, to confirm training requirements through training needs analysis, to further define specific training requirements through job and task analysis, and to design, develop, implement, and evaluate training. The objective of the SAT is to guide the development of performance-based training to support job performance requirements and individual development.

Under the training program, OPG has a list of major grained performance areas where the consequence of human error poses a risk to the environment, the health and safety of persons, or to the security of the nuclear facilities and of nuclear substances. Operations positions requiring CNSC certification are listed as major trained performance areas.

Personnel involved in development and conduct of certification training programs are required to be qualified. All qualifications are developed, implemented, approved and revised following established processes and documented in Training and Qualification Descriptions (TQDs) and guides.

TQDs define the initial training and qualification requirements for each of the positions requiring certification, and for each of the certification training program instructors and examiners. These TQDs describe each phase of the initial training programs. Upon successful completion of the training phases, including required certification examinations, candidates are eligible to receive a CNSC certification.

TQDs will be developed for each of the positions requiring certification at the DNNP. A qualification guide will be developed that describes the initial and continuing training for DNNP operations certification training instructors and examiners.

The DNNP Group 1 initial certification training program is planned to begin in 2026, with the program completed, a sufficient number of CNSC certified staff will be in role prior to the initial fuel load.

Per the *Systematic Approach to Training* [R-65], the training program will be developed as design and operating documentation becomes available. Based on simplification of the design and the advanced safety features incorporated in the BWRX-300, it is anticipated that the SAT-based certification training programs will be of shorter duration and less complex than the current programs at other OPG facilities.

Positions requiring regulatory certification will be defined prior to the LTO submission and will be based on the technology needs and safety significance. Certification training programs will be developed as part of the design process and is expected that only one certification program for the operations staff will be required to meet the regulatory requirements.

The roles and responsibilities for each designated position will be described in a series of role documents that will be developed as operating documentation becomes available.

OPG has mature and effective programs in place for initial and continuing certification training in compliance with applicable regulatory requirements.

In support of future operations, OPG will demonstrate its capability to administer initial training for certified staff, and to ensure a sufficient number of certified staff are available for the safe and reliable operation of the DNNP. This includes having sufficient trained and qualified staff available to deliver the examination and testing programs throughout the station's continued operation.

4.2.4 Work organization and job design

The organization of work is managed under the organizational structure, roles and responsibilities as described in Section 4.1.3. Staffing and resourcing needs will vary during the LTC phase.

OPG will ensure the DNNP contract partners are staffed, resourced, trained and have appropriate work allocation to meet their respective project deliverables.

For OPG, the DNNP will be staffed with a combination of permanent and temporary, augmented, or seconded staff to fulfill project needs throughout its lifetime. OPG resource management and planning is reviewed and reconfirmed on an annual basis as part of the business planning cycle. Onboarding of staff onto the project is done in accordance with established OPG human resource processes.

4.2.5 Applicable OPG Documents

The OPG governance documents for the Human Performance SCA, which supports the licensing basis, are included in Table 4.2-1 below:

Table 4.2-1: Management System Documents for Human Performance

Document	Title
N-PROG-AS-0002	Human Performance
N-PROG-TR-0005 or equivalent	Training

4.3 Operating Performance

The operating performance SCA includes an overall review of the conduct of the licensed activities for the LTC phase, and the programs and processes that enable safe licensed activities.

4.3.1 General considerations

The term 'Operating performance' in the context of this Application refers to the licensed activities taking place during the construction and fuel-out commissioning phases.

Operating performance is governed under OPG-PROG-0039, *Project Management [R-67]* during the construction phase. OPG will have the primary responsibility for the safety and security of construction and commissioning activities, including the work carried out on its behalf by contractors. OPG will have the responsibility to ensure ongoing and intrusive oversight for all phases of the facility development including aspects related to facility construction and commissioning following the OPG management system. OPG will require the constructor to have its own management system compliant with applicable current standards.

Additional details of OPG's construction oversight activities and commissioning and turnover plan to address applicable REGDOC-2.3.1 [R-54] requirements will be submitted in support of this Application via a Construction Management Plan and a Commissioning and Turnover Program Management Plan respectively [R-32]. Compliance to applicable key regulatory documents, codes and standards (including REGDOC 2.3.1) will be demonstrated in applicable documentation submissions and activity planning.

Contractors will manage the Health and Safety of all workers associated with the DNNP as per NK054-PLAN-01210-00034, *Health and Safety Plan* [R-69]. This plan establishes the framework for the management of worker Health and Safety and includes emergency response and fire protection. OPG will use this plan for the construction phase to guide the contractors involved in the construction activities in preparing their health and safety plans. Existing commitments from the PRSL, regarding health and safety issues during construction activities will remain in effect.

Construction activities will be controlled by the constructor with oversight from OPG. Development and execution of commissioning instructions will be in completed in collaboration with applicable contract partners.

Additional information or plans will be produced as required to address specific risks from construction activities. Conventional health and safety is further discussed in Section 4.8 of this Application.

4.3.2 Procedures

Construction and commissioning requirements are covered and controlled by following existing OPG's OPG-PROG-0039 program [R-67]. In support of construction and commissioning activities, the design and oversight will be controlled in accordance with N-PROG-MP-0009, Design Management and N-STD-MP-0009, *Contractor/Owner Engineering Interface and Oversight* [R-71] respectively.

<u>Construction Program</u>

The BWRX-300 design philosophy of "simplicity" and "designed for constructability" enables it to be constructed and commissioned in a short period of time.

Chapter 14 of the PSAR provides an overview of plant construction and commissioning. The Construction Management Plan that demonstrates compliance with REGDOC 2.3.1 will be submitted as a supporting document to this Application. The plan will provide the details for the management of the activities under construction, and OPGs oversight of the construction vendors and construction activities.

The following section provides a brief overview of some key elements in the overall construction strategy.

Long Lead Items

Items with long lead times, on-site manufacturing, modular assembly, and testing will be identified with provisions to ensure construction sequencing is not adversely affected. Any differences between purchasing requirements, the LTC design basis and as-built items will be evaluated and reconciled. Long lead items for the BWRX-300 construction include, but are not limited to:

- Reactor Pressure Vessel
- Hydraulic Control Units
- Fine Motion Control Rod Drives
- Steam Turbine Generator Set
- Reactor Pressure Vessel Internals – Large
- Main Output Transformer
- Steel Bricks™

During construction and fuel-out commissioning, the maintenance, surveillance and in-service testing of components and systems will be with OPG oversight.

Procurement and Receipt of Materials

Established processes and procedures will ensure equipment supplied is manufactured under a QA program that includes inspection for proper fabrication, cleanliness, calibration, and verification of operability.

Procurement packages consisting of commercial and technical sections with any required drawing manifests will be assembled in accordance with approved procurement processes that have been reviewed by the contract partners prior to release.

The product technical specifications will establish the specific product requirements including packaging and handling, and compliance verification requirements. Material and equipment will be categorized using a ranking system that sets the level of inspection requirements during manufacturing that considers the following:

- Consequence of equipment failure (including safety considerations, operational significance, economic significance, etc.)
- Probable occurrence of failure
- Complexity of the design, manufacturing process, and installation (including considerations for first of a kind designs)

At the lowest level, inspections may not be required or will be limited to a final inspection and review of the documentation as needed to satisfy purchase specification requirements (final visual and dimensional inspection, review of testing results, positive material identification, and review of vendor data and documentation).

At the highest level (most strict), an initial pre-fabrication meeting with formal notification may be required. Designated witness and hold points at key manufacturing points will be established in the vendor's quality plan/supplier inspection test plan. The inspection process may include comprehensive, progressive in-process inspections, including dimensional checks, visual examinations, witnessing of functional or performance testing, final acceptance, and pre-shipment inspection. Sub-supplied components may require inspection prior to incorporation into the final product; inspections may be required at sub-supplier fabrication locations.

Components will receive an initial on-site “receipt inspection” to ensure the components are as ordered, have not been damaged, and that the components are not fraudulent, counterfeit or suspect.

Protection of Structures, Systems, and Components

The requirements associated with maintenance and conformity of materials and equipment during on-site fabrication, over the construction cycle up to fuel-out commissioning will be documented and reviewed by contract partners.

- Measures will be established to protect safety class SSC from construction activities.

The following equipment protections controls will be established:

- Environmental condition limits for temperature, pressure, humidity, dust, dirt, airborne salt, wind, and electromagnetic conditions as determined by the component or system design criteria
- Foreign material exclusion measures that prevent the introduction of outside materials, debris, tools, and components where the pose a health and safety hazard or environment impact
- Protection requirements for installed components from personnel traffic, weather, adjacent construction activity or temporary structures
- Implementation of system specific requirements and cleaning methods
- Compatibility requirements for cleaning methods and materials with the components being cleaned to include cleanliness requirements before installation

- Chemistry requirements for layup, cleaning, flushing, and conditioning of piping systems and components
- Requirements for the removal of waste material and consumables generated during construction after completion of work

Storage

Components will be stored in accordance with design and manufacturer requirements with the following considerations:

- Cleanliness and housekeeping practices
- Fire protection requirements
- Protective coatings, preservatives, cover and sleeves
- Physical damage prevention
- Environmental control
- Preventive maintenance requirements and in-storage maintenance
- Security against theft, vandalism, and unauthorized use or alteration
- Shelf life
- Component identification

On-site Manufacturing and Testing

On-site manufacturing will be located where it will not adversely affect construction activity or safety class SSC. Rules and procedures will be established for on-site testing to ensure that applicable codes and standards are met.

Testing and Turnover

The specifics of construction testing will be defined in the installation specifications or in the documentation provided by the major equipment suppliers.

The testing will be used to demonstrate that components and systems are correctly installed, calibrated to ensure accuracy, and operational, that is ready for the application of energy. Completion of construction testing will assure systems are ready for pre-operational testing.

Process and procedures will be established to control and coordinate the turnover of work, structures, equipment, and systems when completed and the associated configuration documentation.

Turnover of SSC from one organization to another will be conducted as follows:

- One organization will be designated as the lead organization and ensures that all responsibilities and limits of authority are clearly established, documented, and communicated.
- Boundaries between SSC will be clearly identified in the field and on documents.
- System status will be defined.
- Prior to acceptance, workers will perform walk downs to the extent necessary on the SSC that are being turned over to ensure that they are in the state defined in the turnover documentation but not necessarily ready for operation.
- Incomplete items, exceptions, and completion schedules will be identified and listed for resolution prior to final acceptance.

The processes and procedures controlling turnover include the following aspects:

- Review of the facility configuration information relating to SSC, and areas by the party turning over the work and the party receiving it for completeness and accuracy
- Performance of tests to ensure the SSC have been manufactured, constructed, and installed to confirm to design specifications
- Identification and assessment of any remaining non-conformances or incomplete components, to ensure there is no safety implication during commissioning activities
- Development of inaugural or baseline inspection data for systems or components for comparative purposes for in-service inspection
- Agreement upon, planning, and scheduling of any outstanding work
- Identification of termination points of the boundaries of turned over SSC (or parts thereof) in turnover documentation with associated required configuration
- Inspection of turned over components and associated records and documents
- Verification of compatibility with information and communication technology systems when turning over electronic documents and records

- Documentation of the turnover of responsibilities including transition of maintenance
- Establishment and turnover of approved as-built plans together with adequate and precise plant configuration details
- Marking and tagging of all SSC turned over

Commissioning Program

Commissioning specifications will be provided by the designer for each system and reviewed by the DA. An integrated plant commissioning program with individual system commissioning plans to satisfy the requirements within the specifications will be developed prior to the commissioning phase in accordance with REGDOC-2.3.1 requirements [R-54].

Commissioning activities performed under the LTC will be controlled in accordance with the OPG-PROG-0039 [R-67] and N-STD-MP-0009 [R-71]. Commissioning which requires fuel-in core will be performed under an LTO and adhere to additional programs and procedures outside the scope of this Application.

The DNNP Program Management Plan [R-31] will describe the processes, procedures, and organization that will be used to manage the turnover and commissioning of the new facility. Chapter 14 of the PSAR also provides details on plant commissioning.

Chemistry Control Program

The OPG organization has oversight responsibility for the chemistry and chemical control program during the construction and commissioning phases, which will be reviewed and accepted by OPG prior to use.

A chemistry control program, specifications, and procedures, established by the constructor, will be in place to cover requirements/aspects for the construction and commissioning phases of the project. This program establishes processes used to control contaminants and operate with optimum chemistry to maintain system integrity during construction and commissioning activities.

The goals of the chemistry control program are to minimize degradation of the assets through control of corrosive impurities, protect the integrity of the nuclear fuel, minimize radiation buildup, and limit chemical and radioactive release for protection of the environment.

The program objectives will be accomplished through appropriate material selection, implementation of system design requirements, procurement and construction requirements, a chemical control program, applying industry mitigation strategies and leveraging BWR water chemistry operational experience.

The chemistry control program will include chemistry procedures, specifications, monitoring requirements, performance indicators, data management, chemical management, and training. The details of the chemistry control program will be developed based on industry standards and guidelines.

The constructor will use a quality management system planned and developed in compliance with contract requirements consistent with CSA N286-12 [R-22]. The OPG organization has oversight responsibility for Chemistry and Chemical Control programs during the construction and commissioning phases. The Chemistry Program established by the constructor will be reviewed and accepted by OPG prior to use.

4.3.3 Applicable OPG Documents

The OPG governance documents for the Operating Performance SCA, which support the licensing basis, are included in Table 4.3-1 below:

Table 4.3-1: Management System Documents for the Operating Performance SCA

Document	Title
OPG-PROG-0039	Project Management

4.4 Safety Analysis

The Safety Analysis SCA covers the development and maintenance of the safety analysis that supports the overall safety case for the BWRX-300 facility.

Safety analysis is an analytical framework used to demonstrate how safety requirements are met for a broad range of operating conditions and various initiating events. Safety analysis involves hazard, deterministic and probabilistic analyses in support of the siting, design, construction, commissioning, operation and decommissioning of a NGS.

The BWRX-300 safety analyses are documented in PSAR Chapter 15 [R-6]. Their objectives are aligned with those specified in REGDOC-2.4.1, *Deterministic Safety Analysis* [R-74] and REGDOC-2.4.2 [R-40]. The completed safety analyses demonstrate that the applicable regulatory safety objectives and the acceptance criteria for event mitigation are met by the plant design.

4.4.1 General considerations

The Safety Analysis SCA is governed by the OPG N-PROG-MP-0014, *Reactor Safety Program* [R-73] which defines the roles, responsibilities, and requirements for performing and oversight of safety analysis.

Adhering to the principles of "informed customer", as per REGDOC-1.1.2, *Licence Application Guide: Licence to Construct a Reactor Facility* [R-19], safety analysis deliverables are prepared by GEH with OPG input and acceptance. OPG's role in providing oversight, input and acceptance of contracted deliverables is further discussed in Section 4.1.3 of this Application.

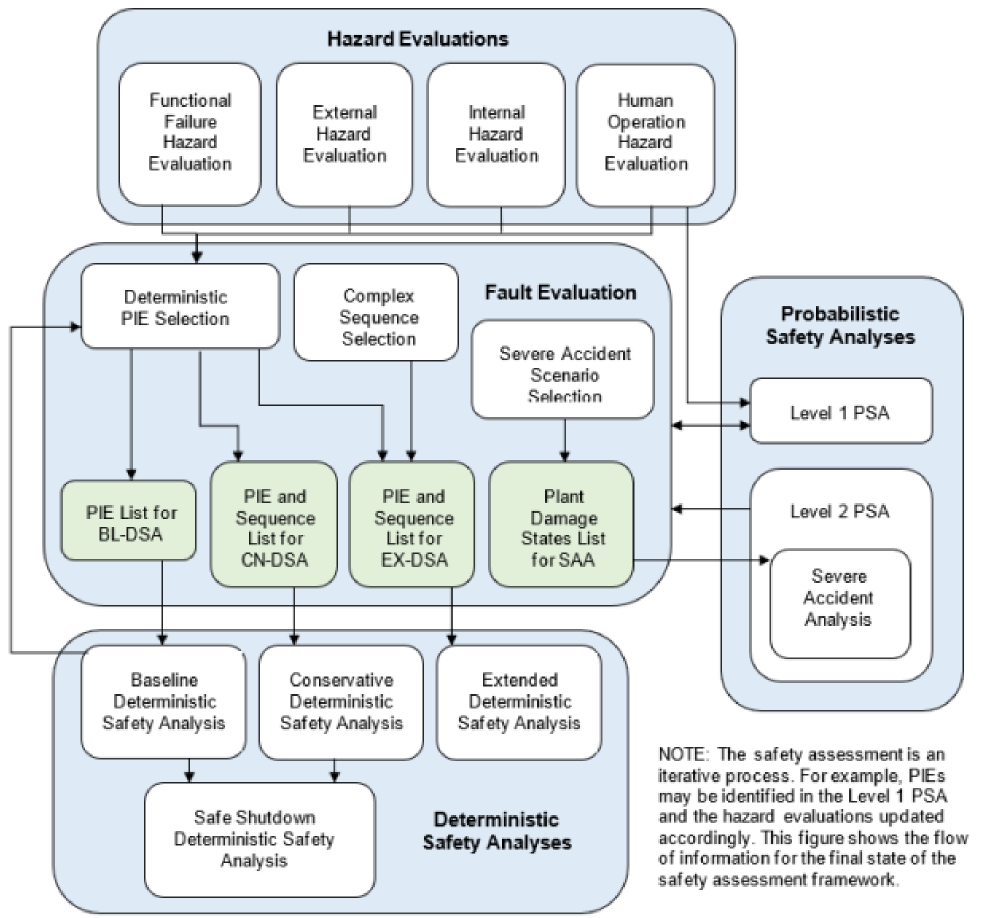
REGDOC-1.1.2 requires that the LTC Application include a PSAR. A PSAR [R-6] for the DNNP BWRX-300 has been prepared and is submitted to the CNSC with this Application.

The BWRX-300 design basis is achieved through an iterative process wherein the design is developed to meet applicable safety objectives, goals, and requirements, which is demonstrated through safety analyses. The results of the safety analyses provide feedback to the design. If indicated by the results, the design may be modified until safety objectives are met. The scope and detail of the safety analysis increase as the design matures.

The processes used for the BWRX-300 safety analysis are presented in PSAR Chapter 15 [R-6] while the results of the analyses are presented in Chapter 15 of the PSAR as follows:

- Section 15.1.2 Analysis of Hazards
- Section 15.5 Deterministic Safety Analyses
- Section 15.6 Probabilistic Safety Assessment
- Section 15.7 Summary of the Results.

The safety analysis framework is illustrated in Figure 4.4-1.



BL-DSA – Baseline Deterministic Safety Analysis
CN-DSA – Conservative Deterministic Safety Analysis
EX-DSA – Extended Deterministic Safety Analysis
SAA – Severe Accident Analysis

Figure 4.4-1: BWRX-300 Safety Strategy Implementation Process

Hazards are key contributors to the overall NGS risk identification and mitigation. The hazard analysis begins with identification of all potential hazards to the plant, which subsequently are screened in a manner commensurate with risk. These hazards are further analyzed as a part of the Deterministic Safety Analysis (DSA) and PSA to demonstrate the effectiveness of protective physical barriers. The hazard evaluation considers internal and external hazards.

A fundamental element of the safety analyses is the identification and selection of PIEs that is achieved through the systematic process of fault evaluation. The fault evaluation scope is the list of potential PIEs generated by the hazards evaluation. The fault evaluation determines which plant scenarios are to be included in the deterministic design basis safety analysis and those are to be analyzed in the PSA.

DSA is aimed on predicting the response of a NGS to PIEs considered alone or in combination with other potential failures. This is an essential component to demonstrate the safety of the plant design. Its objective is to confirm that safety functions can be fulfilled and that the credited SSCs of the plant, in combination with operator actions, are effective in keeping potential releases of radioactive material below acceptable limits.

For AOOs and DBAs, the estimated potential dose to workers and the public from PIEs are well below regulatory limits and fully meet derived acceptance criteria specified in REGDOC-2.4.1 [R-74]. Beyond Design Basis Accident (BDBA) events that could lead to early or large radioactive releases, are practically eliminated.

PSA provides an integrated review of the plant design, operational safety, and complements the results of the DSA. The PSA supports risk-informed design, development and with DSA, demonstrates the success of the design in achieving the design objectives. The PSA assesses design vulnerabilities and optimizes the design using a graded approach consistent with REGDOC-2.4.2 [R-40].

The PSA results, as documented in PSAR Chapter 15, Section 15.6, demonstrate that the quantitative safety goals specified in REGDOC-2.5.2 [R-21] have been met.

4.4.2 Postulated initiating events

The determination of PIEs is an iterative process. The PIEs and associated failure frequencies are first determined qualitatively based on system conceptual design, previous similar designs, and OPEX. A bounding set of PIEs and fault sequences that result in the most significant challenge to the BWRX-300 design are selected for evaluation in the DSA.

The classification of PIEs is related to the Defence in Depth as conceptually illustrated on Figure 4.4-2. The BWRX-300 Safety Strategy relies on five DLs consistent with the IAEA DiD concept and the REGDOC-2.5.2 [R-21] requirements. The first Defence Line (DL1) minimizes potential for PIEs to occur in the first place and minimizes potential for failures to occur in subsequent DLs by ensuring high-quality and conservatism in design, construction, and operation. The second (DL2), third (DL3), and fourth (DL4a & DL4b) lines comprise plant functions that act to prevent PIEs from leading to significant radioactive releases. The fifth line (DL5) involves off-site emergency preparedness to protect the public in case a substantial radioactive release occurs.

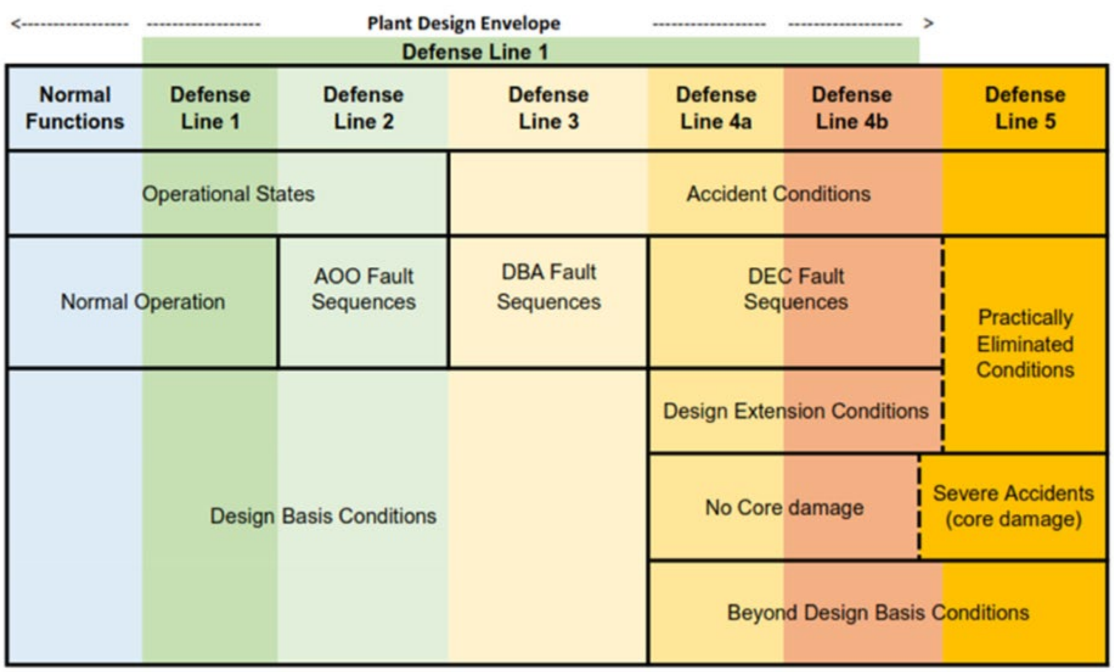


Figure 4.4-2: Conceptual Definition of Defence-in-Depth Concept

Table 4.4-1 shows a high-level description of the objectives, design means, and operational means for the five levels of DLs.

Table 4.4-1: Identification of Defense Levels

Level of Defence (DL)	Objective	Design Means	Operational Means
1	Prevention of abnormal operation and failures	Conservative design and high quality in construction of normal operation systems, including monitoring and control systems	Operational rules and normal operating procedures
2	Control of abnormal operation and detection of failures	Limitation and protection systems and other surveillance features (Safety Category 2)	Abnormal operating procedures/emergency operating procedures
3	Control of design basis accidents	Engineered safety features (Safety Category 1)	Emergency operating procedures
4a	Control of design extension conditions to prevent core melt	Safety features for design extension conditions without core melt (Safety Category 2)	Emergency operating procedures
4b	Control of design extension conditions to mitigate the consequences of severe accidents	Safety features for design extension conditions with core melt Technical Support Center (Safety Category 3)	Complementary emergency operating procedures/severe accident management guidelines
5	Mitigation of radiological consequences of significant releases of radioactive materials	On-site and off-site emergency response facilities	On-site and off-site emergency plans

PIEs are identified and classified such that all foreseeable events with potential radiological consequences or a significant frequency of occurrence are considered in the safety analysis.

A systematic process is used for the BWRX-300 selection and classification of PIEs identified as the result of hazard evaluations. This process is documented in Chapter 15 of the PSAR [R-6] and meets the REGDOC-2.4.1 requirements.

4.4.3 Deterministic safety analysis

Fault sequences consider a combination of the PIEs and additional concurrent failures of the credited mitigating functions. During this process, the PIEs and fault sequences are organized into fault groups, and within each fault group, they are classified based on their estimated frequency of occurrence. The frequency range considered for PIE categorization is consistent with that specified in REGDOC-2.4.1 [R-74], namely:

- Anticipated Operational Occurrences (AOOs) - frequencies of occurrence equal to or greater than 1E-02 per reactor year;
- Design Basis Accidents (DBA) - frequencies of occurrence equal to or greater than 1E-05 per reactor year, but less than 1E-02 per reactor year; and,
- Beyond Design Basis Accidents (BDBA) - frequencies of occurrence less than 1E-05 per reactor year.

The output of this process is a list of representative PIEs and fault sequences.

The list of faults documents the representative set of PIEs and fault sequences to be analyzed and is used as the primary tool to maintain consistency between design and analysis activities while iterations are occurring.

An important aspect of the BWRX-300 DSA is the systematic integration of Common Cause Failures (CCFs), Common Mode Failures, and single failures, both as event initiators and failures affecting event mitigation, into the DSA and DiD Concept.

The BWRX-300 DSA uses a layered analysis approach that includes three types of DSA:

- Baseline – deterministic safety analysis (BL-DSA)
- Conservative – deterministic safety analysis (CN-DSA)
- Extended – deterministic safety analysis (EX-DSA)

This approach addresses initiating and mitigating DL function failures in a more systematic and structured manner.

These DSA acceptance criteria are discussed in PSAR Chapter 15, Section 15.3 [R-6].

The primary objective of the BL-DSA is to demonstrate the effectiveness of the DL2 functions. The scope of BL-DSA includes single failure PIEs and models the expected response of the plant to demonstrate that the event meets applicable acceptance criteria. The analysis endpoint is a controlled stable state. The mitigating DL functions credited in BL-DSA are DL2 functions. If a DL2 function fails or is not effective, then the corresponding DL3 function is credited.

CN-DSA primary objective is demonstrating the effectiveness of DL3 functions. The CN-DSA scope includes PIEs due to single failure, PIEs due to spurious CCF in DL2 or DL4a, and PIEs with postulated passive CCF of DL2 functions that were credited in BL-DSA. The CN-DSA is performed using conservative initial plant conditions with established acceptance criteria and applying a graded approach in quantifying the uncertainties (see PSAR Chapter 15, Section 15.5). Single failure criterion is applied to DL3 Safety Class (SC) 1 SSCs. CN-DSA credits only DL3 mitigation function. The endpoint of the analysis is a controlled stable state.

The EX-DSA objective is assessing the effectiveness of DL4 functions. The EX-DSA scope includes events categorized as DECs, including PIEs due to spurious CCF of DL3 functions, DBA with postulated passive CCF of DL3 mitigating functions, and complex sequences identified by the PSA.

Dose acceptance criteria

The BWRX-300 design compliance with the requirements in REGDOC-2.4.1 [R-74] is shown in PSAR Chapter 15, Section 15.3 [R-6].

The estimated doses shown in Section 15.7 of the PSAR [R-6] meet the dose acceptance criteria for AOOs and DBAs specified in REGDOC-2.5.2 [R-21], with significant margin.

Trip coverage and trip setpoints

BWRX-300 trip coverage is documented through the deterministic analysis approach.

For all AOOs and DBAs where failure to trip could result in potentially unacceptable consequences, sufficient trip coverage is provided to initiate reactor shutdown. Details are provided in Chapters 4 and 15 of the PSAR [R-6]

4.4.4 Hazard analysis

The BWRX-300 hazard analysis process is described in PSAR Chapter 3, Section 3.1.6 [R-6]. It includes both internal and external hazards and the complete range of plant states that produces a comprehensive set of PIEs.

Screening criteria (qualitative or quantitative) are applied to the potential hazards to screen out the hazards that pose no adverse consequence to the safe plant operation. As noted earlier in this section, safety analysis and design follow an iterative process, so along this process screening of hazards will be further refined as the project advances through to detailed design.

The BWRX-300 safety strategy includes four types of hazard evaluations that envelope both natural and human-induced internal and external hazards:

- Functional Failure Hazard Evaluation (FFHE)
  The evaluation identifies failures of plant systems or equipment with potential to cause a challenge to any FSFs.

- External Hazard Evaluation (EHE)
  This evaluation identifies failures that originate from a source that is not under control of the nuclear power plant licence holder. External hazards include both natural and human induced hazards.

- Internal Hazard Evaluation (IHE)
  This evaluation identifies conditions originating within the boundaries of the site that can lead to a plant transient and potential equipment failures.

- Human Operation Hazard Evaluation (HOHE)
  The human operation hazard evaluation identifies erroneous decisions or actions made by operators that could result in an unintended state-change of the plant.

The hazard analysis described above for the BWRX-300 design conforms to the requirements of REGDOC-2.4.2 [R-40]. The hazards analysis methodology and preliminary results for the hazard analysis is documented in supporting documents [R-23] [R-76] and in PSAR Chapter 15, Section 15.1 [R-6].

4.4.5 Probabilistic safety assessment

The PSA objectives of the BWRX-300 are aligned with the objectives identified in REGDOC-2.4.2 [R-40]. The PSA performed as part of the PSAR submission is also consistent with the methodology submitted to the CNSC prior to this Application.

The DNNP PSA covers all relevant initiating events consistent with REGDOC-2.4.2. The event types investigated include:

- Internal initiating events
- Loss of offsite power
- Internal hazards (e.g., fire, flooding, lifting of heavy loads)
- External hazards (e.g., high wind, seismic events)

The PSA is used to inform the design and further reduce risk by identifying potential plant vulnerabilities. The methodology and preliminary PSA results documented in supporting documents [R-42] [R-55] and in PSAR Chapter 15, Section 15.6 [R-6]. The results demonstrate that the BWRX-300 design is well below the safety goals identified in REGDOC-2.5.2 [R-21].

4.4.6 Severe accident analysis

The severe accident analysis is performed as a part of the PSA to analyze accident sequences and identify possible plant vulnerabilities. It also analyzes effectiveness of the designed systems and informs the severe accident management program by improving the emergency response.

Severe accidents are a subset of BDBA events, based on a frequency of less than 1E-5 occurrences per year that may lead to severe core damage with potential for radioactive release.

As documented in PSAR Chapter 15, Section 15.1[R-6], the Severe Accident analysis was used to identify plant vulnerabilities to aid in designing mitigating systems for severe accident management. The analysis showed that the probability of early and late large release is practically eliminated as evidenced by the analysis results which are well below the safety goals defined in REGDOC-2.5.2 [R-21]. The BWRX-300 Severe Accident Management Program is in development and informed by the insights from the analysis for optimization of accident management strategies and measures as described in PSAR Chapter 15, Section 15.6 and Appendix 15B [R-6].

4.4.7 Overall summary

A summary of results of the safety analyses for the DNNP is documented in PSAR Chapter 15, Section 15.7 [R-6].

The establishment of the BWRX-300 design basis is achieved through an iterative process wherein the design is developed to meet defined safety objectives, goals and requirements as confirmed by safety analyses.

The completed analyses for bounding events have demonstrated that the DNNP BWRX-300 design meets the dose acceptance criteria and quantitative safety goals with adequate margins. This fulfills the technical and radiation protection safety objectives of REGDOC-2.5.2.

4.4.8 Event mitigation

A key aspect of event mitigation is accident management, which is described by REGDOC-2.3.2, *Accident Mitigation* [R-75] as a set of actions during the evolution of an accident to prevent the escalation of the accident, to mitigate the consequences of the accident and to achieve a long-term safe stable state after the accident.

PSAR Chapter 15 [R-6] outlines OPG's strategies for accident management. From the plant perspective, accident management is split into two parts:

- Preventive accident management integrates actions and measures needed to prevent a severe accident from developing more severe consequences and core damage and to terminate its progress if it has started. Preventive accident management deals with postulated accidents and those DECs that do not result in core damage and is usually accomplished by plant operation staff using emergency operating procedures (EOPs) in the MCR.

- Mitigative accident management, which is often referred to as severe accident management, is primarily devoted to maintaining the integrity of the containment and minimizing the release of radioactive materials. Accident mitigation also serves to achieve a long-term stable state by cooling the core when the fuel has started to degrade beyond the acceptable level.

The concept of practical elimination is also used to demonstrate that adequate design provisions have been implemented across all levels of BWRX-300 DiD to prevent an occurrence of a severe accident or arrest its progression.

The DSA analyzes the event sequences and ensures that the credited safety functions are effective in stopping the event progression. Event mitigation is also considered in the PSA to identify potential plant vulnerabilities and design improvements to reduce the probability of severe accidents or should an accident occur, mitigate the consequences. The optimization of the reactor design is further discussed in Section 4.4.5 of this Application and PSAR Chapter 15, Section 15.6 [R-6].

Furthermore, Section 4.5 of this Application and PSAR Chapter 3 provide further information on the containment design margin designated to mitigate the consequences of DECs and BDBAs. The containment design conforms to the acceptance criteria requirements in REGDOC-2.5.2 [R-21].

4.4.9 Applicable OPG Documents

The OPG governance documents for the Safety Analysis SCA, which supports the licensing basis, are included in Table 4.4-2 below:

Table 4.4-2: Management System Document for the Safety Analysis SCA

Document	Title
N-PROG-MP-0014	Reactor Safety Program

4.5 Physical Design

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

4.5.1 General considerations

OPG has a fully mature and effective design program in place to support the BWRX-300 design and meet all regulatory requirements with high safety and quality standards. The objectives of OPG's design program are to ensure that structures, systems, and components of OPG's nuclear facilities meet requirements and operate safely, reliably, effectively, and are consistent with design analysis and QC measures. OPG's design program satisfies the requirements of CSA N286-12 [R-22], and REGDOC-2.5.2 [R-21].

The N-PROG-MP-0009, *Design Management* [R-79] program sets the overall requirement for execution and control of activities that provide design support and documentation for the nuclear facility. This program provides assurance that all design activities and their resulting documentation are controlled in a manner consistent with the plant's licensing basis. The program describes processes for monitoring, and assessment of design activities to ensure that appropriate interfaces and oversights are maintained throughout the design process.

N-STD-MP-0009 [R-71] summarizes the framework for oversight, monitoring and assessment of design activities conducted by contractors to facilitate the successful implementation of tasks at OPG by ensuring that contractor activities are in conformance with OPG's QA Program and regulatory requirements or an OPG approved contractor QA Program.

The *DNNP Engineering Project Oversight Plan*, NK054-PLAN-01210-00035 [R-56], outlines the oversight to be implemented by OPG DNNP Owners Engineers. The Oversight Plan will ensure that:

- Engineering design deliverables meet design quality and procedural requirements specified by OPG and the QA programs of GEH and other contract partners.
- OPG collaboration with GEH and other contract partners is effective and all requirements are satisfactorily met.

- There is early detection of potential issues so corrective measures can be taken at the appropriate time.

GEH is the DA for scoped design activities in accordance with the contracting strategy. GEH will maintain DA through the design, construction, inspection and testing, and fuel-out commissioning phases. A turnover strategy will describe the transition of the DA to OPG prior to plant start-up. OPG is the DA of the areas outside of GEH scope.

In their respective roles as DA and oversight, GEH and OPG will meet the REGDOC-2.5.2 Section 5.1 design objectives with their respective management systems and QA programs. This includes the required administrative controls, design controls, and QA processes, procedures, and practices.

GEH will perform the role of DA in compliance with the GEH Quality Management System with OPG providing oversight. OPG has performed a QA compliance audit of GEH processes and procedures and confirmed the GEH Quality Management System is compliant with CSA N286-12 Management System Requirements for Nuclear Facilities. OPG will confirm that the program is implemented as per the GEH Quality Management System using strategic oversight.

OPG is accountable for ensuring design objectives are met as per REGDOC-2.5.2 [R-21]. OPG will ensure that GEH, as DA, is meeting these objectives with strategic oversight of the development lifecycle phases. Strategic oversight will ensure OPG personnel are aware of the areas of nuclear safety, safe operating envelope, design bases, and licensing basis. Through this oversight along with compliance audits, OPG will act as an informed customer.

The oversight activities include review and acceptance of design deliverables, as well as performing surveillance and witnessing of engineering activities performed by GEH, and other contract partners, in accordance with their management systems.

Design changes

Throughout the design and construction phase OPG will act as an informed customer and provide oversight to ensure that design changes are controlled.

Facility configuration will be maintained from initial conception to end of operating life through programmatic configuration and change control processes overseen by the Design Authorities. These configuration management processes will comply with the applicable requirements in REGDOC-2.3.1 [R-54], REGDOC-2.5.2 [R-21], CSA N286-12 [R-22], and CSA N286.10 [R-53].

At a high level, configuration management is used to:

- Control project information through identification, categorization, storage, document control, and records management
- Control the implementation of changes

Changes during the construction phase will be processed to maintain conformance with design requirements, physical configuration, and configuration documentation. As part of this process, OPG and GEH will ensure that changes are reviewed, approved, and released with the appropriate notification for field changes and non-conformance issues. Relevant configuration changes will be communicated to the CNSC in accordance with applicable license conditions.

Applicable regulations, codes and standards

Regulations, codes, and standards relevant to the DNNP design have been reviewed and evaluated to determine their applicability, sufficiency, and adequacy for the BWRX-300 design. At a high level, evaluation of the regulations, codes, and standards was performed as follows:

- Review REGDOC-1.1.2 [R-19] and REGDOC-2.5.2 [R-21] for each topic area and determine applicable requirements and guidance
- Review documents referenced in REGDOC-1.1.2 and REGDOC-2.5.2 (e.g., under Additional Information) to obtain additional requirements and guidance
- Review codes and standards referenced in the applicable REGDOCs (e.g., CSA standards) and determine whether applicable to the BWRX-300 design

- Add any relevant additional codes and standards not covered in CNSC REGDOCs or CSA standards or add any alternative codes and include justification for using alternative codes.
- Include relevant codes and standards based on the expertise of the technical engineering leads and informed by recent relevant designs

Codes and standards that differ from those used in Canada, and cases where applicable regulations, codes and standards are not met directly, have been subjected to a safety assessment. The safety significance of these deviations has been assessed to confirm that they do not negatively affect the overall level of safety for the DNNP facility.

The analysis and safety assessment of the codes and standards deviations is summarized in PSAR Section 1.11 [R-6]. Further details on applicable regulatory documents, codes and standards can be found in the NK054-REP-01210-00137, *DNNP Licence to Construct Regulatory Documents, Codes & Standards* [R-58] and Appendix B and C of PSAR Chapter 1 respectively.

<u>General Design Approach of the Systems Structures and Components</u>

The BWRX-300 leverages the ESBWR design, proven in-use materials and off-the-shelf components. DNNP's BWRX-300 uses a well-defined, robust, procedure-driven process to direct cost-effective design decisions that meet safety objectives. This design approach is focused on safety which drives value engineering principles and uses a design-to-cost approach for quality, completeness, engineering, manufacturing, and construction.

The design approach includes:

- Comprehensive, collaborative, and iterative design process
- Rigorous controlling procedures and requirements
- Documented and controlled processes and tools
- Robust configuration management
- Evolutionary development process

A rigorous process is used to optimize the design through the various phases. Technical reviews and configuration control are executed throughout each design phase, providing a high level of confidence that requirements are met for each phase. For further details, refer to PSAR, Chapter 3, Section 3.1 [R-6].

General Plans for Construction

The BWRX-300 draws from proven construction methods from within and outside of the nuclear construction industry such as the water and tunneling industry. The design allows for large modular components that can be consistently lifted, set, aligned, and fixed in place.

As part of normal project processes, a construction sequence is used to support detailed construction sequencing for determination of travel routes, laydown areas, fabrication and storage facilities, and general construction site layout.

Improved Construction Approaches

The construction of the plant is performed using a standardized construction plan. The key excavation feature is the reactor shaft development. Straight line shaft excavation will minimize material removal and backfill requirements. Standard techniques used by tunneling and hydraulic industries will be used to accomplish this excavation.

As shown in Figure 4.5-1, Steel Bricks™ is a type of Steel-Plate Composite and will be used for the walls and floors of the RB. Steel Bricks™ provide modularity, off critical path construction capability and eliminate the foundation interface challenges. Steel Bricks™ utilize formed steel shapes generally rectangular in nature that provide both strength and permanent concrete forming. No rebar or rebar assembly is required.

Figure 4.5-1: Steel Bricks™

The modularized construction will be conducted in three stages:

- Shop fabrication and pre-assembly of the largest standard shipping load to site
- Site fabrication to further assemble material to largest possible size for setting with equipment available at site
- Final in-place assembly

Approaches to Component Manufacture

Many of the components utilized in the BWRX-300 have been used extensively in the nuclear and power industries and have a well-established supply chain.

The BWRX-300 will employ factory assembly of the turbine and generator which has been effectively applied in the combined cycle industry. This approach eliminates the need for open-top or open-ended assembly of fine-tolerance critical-equipment in a construction environment. This reduces the risk of foreign particle entry and minimizes exposure of the internal components to elements. Also, assembling this equipment in a shop environment by craftsmen that routinely perform these tasks on manufacturer specific equipment daily will enhance QA.

4.5.2 Site characterization as it relates to plant design.

Site characterization has been conducted extensively for the DN site. The characterization information is used as an input for the safety analysis and facility design.

A summary of DNNP site characteristics and new data collected since submission of the PRSL renewal application is provided in section 3.1 of this document.

4.5.3 Design principles and requirements

This section summarizes the design principles and requirements that govern the overall design of the facility as well as the operation and interaction of all structures, systems, and components (SSCs). The goal is to demonstrate that the reactor facility will be reliable, robust, maintainable, and meet the required overall level of safety. To support this goal, it will be demonstrated that the DNNP facility design:

- Conforms to applicable codes and standards
- Considers OPEX and the latest R&D
- Mitigates the potential effects of common-cause events and severe accidents

<u>Safety objectives and goals</u>

The general nuclear safety objective is that the BWRX-300 is designed and will be operated in a manner that protects individuals, society, and the environment from harm. To ensure this, the BWRX-300 is designed to comply with the following three complementary safety objectives, as identified in REGDOC-2.5.2 [R-21] and PSAR Chapter 3 Section 3.1 [R-6].

- Radiation Protection Objective: Radiation exposures within the reactor facility during normal operations, AOOs or due to any planned release of radioactive material from the reactor facility are kept below prescribed limits and as low as reasonably achievable (ALARA). Provisions are made to mitigate the radiological consequences of any accidents.
- Technical Safety Objective: All reasonably practicable measures are taken to prevent accidents, including those of very low probability, and to mitigate the consequences of events should they occur.

- Environmental Protection Objective: All reasonably practicable measures are taken to protect the environment during the construction and operation of the reactor facility and to mitigate the consequences of an accident. The design includes provisions to limit, control, treat and monitor releases to the environment and minimize the generation of radioactive and hazardous wastes.

Radiation Protection Objective

The DNNP's BWRX-300 design accounts for the Radiation Protection (RP) objective.

Designing to meet the Radiation Protection Objective begins with a comprehensive identification of radiation sources and an appropriately conservative source term. Then comprehensive safety features and measures are provided to prevent unplanned exposures and limit occupational exposure during plant operation (including in potential accidents), maintenance, and decommissioning. These safety features and measures, in order of preference, include:

- Passive engineered safety features
- Active engineered safety features
- Administrative safety measures

Engineered safety features (some passive, others active) include shielding, ventilation, containment, remote handling, and interlocks. Administrative safety features which reduce radiation exposure during planned operations include restrictions on occupancy, monitoring arrangements, pre-planning of exposure and the use of barriers (including personal protective equipment) and notices.

Details on how radiation protection is considered in the design for both normal operation and accident conditions are provided in PSAR Chapter 12.

Technical Safety Objective

The DNNP BWRX-300 is designed to ensure that potential radiation doses to the public from AOOs and DBAs will not exceed dose acceptance criteria as per REGDOC-2.5.2 [R-21]. In the deterministic safety analysis, the potential committed whole-body dose for average members of the public who are most at risk, at or beyond the site boundary (defined as "critical groups"), is calculated for a period of 30 days after the analyzed event to confirm that for AOOs and DBAs, doses would be less than or equal to the following:

- 0.5 millisievert (mSv) for any AOO, or
- 20 mSv for any DBA.

Section 4.4 of this Application as well as PSAR Chapter 15 [R-6] describe how all accidents, including those of very low probability, have been considered, and demonstrate how safety goals described in REGDOC-2.5.2 [R-21] are met by the DNNP BWRX-300 design. Additionally, the DiD approach to design and analysis ensures that the nuclear safety function is maintained after PIEs (see section 4.4.2 for further discussion).

Environmental Protection Objective

With respect to environmental protection, the design includes provisions to control, treat, and monitor the main sources of liquid, gaseous and solid radioactive wastes. PSAR Chapter 11 [R-6] and Section 4.5.16 of this Application documents the measures that will be taken for the safe management of radioactive waste. Additional details on overall environmental aspects of the plant are provided in PSAR Chapter 20.

Operating Limits and Conditions

The BWRX-300's safe operating envelope and a set of OLCs for the facility will ensure compliance with the safety analysis.

OLCs are derived from deterministic safety analyses, PSAs and evaluations, and include the following:

- Safety Analysis Limits and safe operating limits (Safety limits and limiting Safety system settings)
- Conditions of operability
- Actions and action times
- Surveillance requirements (SR)

The OLCs are outlined, and their bases are documented, in PSAR Chapter 16 [R-6].

Safety assessment and engineering evaluation

Section 4.4.1 of the Application describes the safety assessment framework (SAF) applied to the safety analysis and design of the DNNP BWRX-300 facility.

The SAF aligns with the DiD concept which is described in Section 4.4.2 of this Application. The framework elements are organized into evaluation and analysis activities. Design, evaluation and analysis are performed in an iterative manner as the design progresses from conceptual design towards detailed design.

Requirements generated from all sources (safety assessments, product requirements, regulatory documents, codes & standards, customer decisions, etc.) are managed by GEH as the DA as per the following:

- Traceability: Establish relationship between requirements, and verification and validation
- Change Impact Assessment: Facilitate determination of impact on upstream and downstream requirements when design changes are proposed
- Completeness of Design: Provide confidence that stakeholder requirements are fulfilled in the body of design documentation. This results in an auditable trail of requirements in the design documentation and evidence that the completed design meets the requirements.
- Control of Scope: Help prevent introduction of functions and features that are not required. It accomplishes this by imposing discipline during the design process such that requirements are not introduced without a basis.
- Consistency: A defined strategy helps ensure that various organizations and disciplines have a common understanding about how requirements are managed, how the requirements architecture is used, and how the same tools are used in the same way.

General Engineering Design Evaluation

The requirements derived from the safety analyses for plant systems are detailed in the PSAR as follows:

- Timing of system operation is documented in PSAR Chapter 15, Safety Analysis

- Minimum system performance envelope to meet safety analysis assumptions is documented in PSAR Chapter 16, Operational limits and conditions for safe operation
- Ability of the system to perform over the lifetime of the reactor facility is documented in PSAR Chapter 13, Conduct of Operations (Section 13.3.4. Ageing management and long-term operation)
- Ability of the system to perform in any abnormal environmental conditions in accident scenarios for which the system is credited is documented in Chapter 3, Section 3.9.

The design principles of independence, diversity, separation, and redundancy are applied to safety systems ensuring high reliability, fail-safe operation, and a reduction in CCFs. The level of independence, diversity, separation, and redundancy is commensurate with the safety significance of the system and plant level Defence in Depth Requirements. The design safety principles align with the requirements of IAEA SSR-2/1, *Safety of Nuclear Power Plants: Design* [R-26] and REGDOC-2.5.2 [R-21] and further details are provided in PSAR Chapter 3 [R-6].

Identification of facility states and operational configurations

Plant states considered in the DNNP BWRX-300 design are grouped into the following four categories:

- Normal Operation is operation within specified Operational Limits and Conditions (OLCs) including all plant operating modes as specified in PSAR Chapter 16 Operational Limits and Conditions.
- AOOs are deviations from normal operation that are expected to occur once or several times during the operating lifetime of the reactor facility. With the appropriate design provisions, these AOOs will not cause any significant damage to items important to safety or lead to accident conditions.
- Design Basis Accidents (DBAs) are conditions for which a reactor facility is designed according to established design criteria.
- Design Extension Conditions (DECs) are a subset of beyond-design-basis accidents (BDBA) that are considered in the design process of the facility in

accordance with best estimate methodology to keep releases of radioactive material within acceptable limits. DECs could include severe accidents. No other BDBA plant states are considered in the design.

The basis for the categorization of plant states is the expected frequency of occurrence of PIEs. Acceptance criteria are assigned to each plant state in the design, considering the principle that frequent PIEs have only minor or no radiological consequences, and that any events that may result in severe consequences are of extremely low probability. The aim of the acceptance criteria is to demonstrate that barriers to the release of radioactive material from the plant will maintain their integrity to the extent required following a PIE.

<u>Design envelope</u>

The plant design envelope includes NOs, AOOs, DBAs and DECs. Practically eliminated conditions are very low probability events or sequences that could result in high radiation doses or large radioactive releases if they occurred and should therefore be practically eliminated. Practical elimination is applied to events leading to or involving core damage (i.e., a severe accident) and achieved by justifying that the event is either:

- Physically impossible or
- Extremely unlikely to occur with a high level of confidence (i.e., probabilistic criterion)

Practical elimination has been applied in a manner consistent with REGDOC-2.5.2 [R-21]. Preference is given to implementing design features that eliminate events rather than relying on probabilistic analysis.

Additional information can be found in PSAR Chapter 3, Section 3.1, PSAR Chapter 15 Section 15.1 and Section 15.2 [R-6].

<u>Defence-in-depth</u>

Section 4.4.2 provides an overview of the DiD concept and the associated DLs.

The concept of DiD involves the provision of multiple layers of defence against some undesirable outcome rather than a single, strong defensive layer. In the case of a nuclear power plant, the undesirable outcome is the exposure of workers, the public or the environment to radioactivity exceeding levels determined to be safe.

There are two types of defensive layering considered:

1. Physical barriers in place to prevent the release of radioactivity: The fuel matrix, fuel cladding, reactor coolant pressure boundary (RCPB), and containment. The integrity of one or more physical barriers must be maintained to prevent unacceptable releases.
2. A combination of active, passive, and inherent safety features used to minimize challenges to the physical barriers, to maintain the integrity of the barriers and, in case a barrier is breached, to ensure the integrity of the remaining barriers.

Safety functions

The BWRX-300 fulfills the FSFs described in Section 4.5.1 during all plant states (normal operation, AOO, DBAs and DECs) ensuring the design meets safety objectives.

If a manual action is credited to perform an FSF, the monitoring and display of plant parameters necessary to perform the manual action successfully are also considered part of the FSF. The FSFs are central to both the SAF and the DiD concept of the BWRX-300.

The SAF utilizes the FSFs to define which faults/failures should be considered as PIEs based on the potential to challenge one or more FSF. This is documented further in PSAR Chapter 3 Appendix 3A. The DiD concept uses the FSF to define the interface between the DLs and the physical barriers to prevent radioactivity release. This is discussed in more detail in PSAR Chapter 3 Section 3.1.6 [R-6].

Safety classification of structures, systems and components

The BWRX-300 approach to classifying SSCs important to safety is based on the classification principles contained in the IAEA SSR-2/1 [R-26] and SSG-30, *Safety Classification of Structures, Systems, and Components in Nuclear Power Plants* [R-27], and REGDOC-2.5.2 [R-21]. As such, this approach is based primarily on deterministic methods, and is directly traceable to the safety functions performed by the SSC.

This approach addresses:

- The consequences of the SSC's failure to perform its safety functions,
- The expected frequency of the SSC being called upon to perform its safety functions, and
- The time following a PIE at which, or the period for which, the SSC may be called upon to perform a safety function.

A fundamental element of the classification approach is the direct correlation between the DLs in which a function resides. In accordance with the function's safety significance and role in preserving the FSFs, functions are categorized into three Safety Categories: Safety Category 1, Safety Category 2 and Safety Category 3, with Safety Category 1 being the most important. All SSCs that support DL2, DL3, DL4a & DL4b functions are considered important to safety.

This approach is summarized in Table 4.5-1 below. The classification of SSCs is discussed in more detail in PSAR Chapter 3 Section 3.2 [R-6].

Table 4.5-1: Summary of Safety Classification of SSCs

Safety	Defense Line	Safety Category of Function	Safety Class (SC) of SCC
Increasing Safety Function	DL3	1	SC 1
	DL4a	2	SC 2
	DL2	3	SC 3
	DL4b	3	SC 3
Non-Safety Related	N/A	N/A	Non-Safety Class

REGDOC-1.1.2 [R-19] recommends specific criteria for deciding on the appropriate design requirements that the safety classification methodology should use. The criteria are as follows:

- Appropriate codes and standards to be used in the design, manufacturing, construction, testing and inspection of individual SSCs (see PSAR Section 3.2.6 [R-6])

- System-related characteristics, such as the degree of redundancy, diversity, separation, and reliability (see PSAR Section 3.2.6)
- Environmental Qualification (EQ) (see PSAR Section 3.4.6)
- Seismic qualification (see PSAR Section 3.2.5)
- Availability requirements for particular SSCs for on-demand duty and for reliability for the prescribed mission time (see PSAR Section 3.2.6)
- Quality assurance requirements (see PSAR Section 3.2.3 and 3.2.4)

Design for reliability

This section describes how designing for reliability has been incorporated into the BWRX-300 design.

Reliability Assessment Methods

The BWRX-300 Reliability Program is used to ensure that all systems important to safety function reliably in accordance with design and performance criteria. The Reliability Program includes the following:

- Identification and categorization of systems using a systematic process;
- Identification of specific failure modes and specification of reliability targets;
- Specification of minimum capability and performance level consistent with safety targets and regulatory requirements;
- Provisions for information incorporation into maintenance programs;
- Provisions for inspection, tests, modeling, and monitoring to assess reliability based on safety class; and,
- Documentation of program activities, attributes, elements, results, and administration.

See PSAR Section 13.3 [R-6] for more information on how the Reliability Program addresses fitness for service.

The classification of SSCs in the BWRX-300 design is used to determine the required reliability. This classification informs the appropriate materials and codes and standards to be used in the design, manufacture, construction, testing, and inspection of SSCs. The prescribed Safety Class defines the relative safety importance of a given DL function.

Various aspects of the safety analysis which are detailed in Section 4.4 are used to assess the reliability of the facility. System reliability requirements and associated design features for mechanical systems are discussed in PSAR Chapter 3 Safety Objectives and Design Rules for Structures, Systems, and Components, and I&C systems (DL2, DL3, DL4a, and non-classified) are discussed in PSAR Chapter 7 [R-6].

Design Principles and Features Supporting Reliability

Several design principles and BWRX-300 design features have been utilized to support the specified reliability goals. These include:

- Application of deterministic design principles in codes and standards
- Minimizing probability of failures for SSCs with deterministic design
- Deploying a simple design with minimal complexity
- Incorporating principles of independence, diversity, separation, and redundancy
- Use of passive safety features

Additional detail on these design principles and features can be found in PSAR Chapter 3, Section 3.1 [R-6].

Common Cause Failures

CCFs are functional failures of multiple components due to a single event or cause. A key impact is that the reliability of an SSC can be considerably reduced by a dependent failure, relative to random failure mechanisms acting alone. CCFs can affect several SSCs important to safety simultaneously and require appropriate design measures to minimize their effect. For the BWRX-300, protection against CCFs is achieved through the application of diversity in defence in depth and an assessment through functional hazard evaluations. Additional detail can be found in PSAR Chapter 3, Section 3.1 [R-6].

Single Failure Criterion

The single failure criterion requires that each safety group perform all the required safety functions for a given PIE in the presence of any single component failure. For the BWRX-300 design this criterion is applied in accordance with REGDOC-2.5.2 [R-21] in the following ways:

- As a design attribute typically achieved through redundancy in the system architecture of high safety classification SSCs
- As a conservative assumption made in design basis analysis (in addition to the PIE and fault sequence) to demonstrate a high degree of confidence that acceptance criteria will be met

The PSA is used to identify single failures that should be considered in design basis analysis and functions as a complementary means for demonstrating insensitivity to single failures. Additional detail can be found in PSAR Chapter 3, Section 3.1 [R-6].

Fail-Safe Design

Fail-safe design requires that systems and equipment fail to a safe state, or a to a known defined state that has been determined not to jeopardize:

- Plant safety
- Accident mitigation
- Prevention of a safety-related function

Fail-safe design features are incorporated into the BWRX-300 using safety system actuations that are not reliant on external power. This ensures that FSFs are fulfilled for accident conditions. Examples of fail-safe design in the BWRX-300 include:

- Control Rod Drive (CRD) hydraulic scram valves which open on loss of power
- Isolation Condenser System (ICS) initiation on loss of power

Allowance for Equipment Outages

The BWRX-300 maintenance program provides the framework for identification, control, planning, execution, auditing, and review of maintenance, inspection, and testing of SSCs. The BWRX-300 is designed for the performance of online maintenance and testing of safety class SSCs. SSCs that cannot undergo online maintenance and testing have been designed to ensure reliability goals are met between equipment outages. SSCs credited in the safety analysis are identified and periodically tested at a frequency related to reliability analysis results and operational experience. Additional detail can be found in PSAR Chapter 13, Section 13.3 [R-6].

Shared Systems

The use of shared systems, functions, and instrumentation is controlled in the BWRX-300 design, utilizing defence in depth principles to maintain independence in shared systems between DLs. Design features for I&C, Electrical, and Fire Protection Systems (FPS) include design components that separate shared system DLs from impacting the function of each line. Additional information on the approach to shared I&C can be found in PSAR Chapter 7, Section 7.3 [R-6].

Ageing-Related Considerations

Ageing considerations for SSCs are considered from the conceptual design phase and included in the system and component design requirements and specifications. Known ageing phenomena are quantified in the design of SSCs and consideration is given to age-related degradation to ensure that safety and performance are maintained throughout the lifetime of an SSC.

Plant programs are implemented to facilitate periodic surveillance, inspection, testing, and maintenance to assure system reliability throughout the facility lifecycle. The surveillance frequencies are based on a combination of reliability analysis, the PSA, and previous OPEX. Additional information on this topic can be found in PSAR Chapter 3, Section 3.1 and PSAR Chapter 13, Section 13.3 [R-6].

Human factors

The high-level goal of the *BWRX-300 Human Factors Engineering Program Plan (HFEPP)* [R-81] is to specify a proportionate, integrated, and effective Human Factors program for all phases of the reactor lifecycle that reduces the risks and consequences influenced by human interactions, as far as reasonably achievable. The program includes all elements of planning, analyses, design guidance, job design, verification, and validation to ensure that the human factors considerations have been effectively integrated in the design process. It outlines the detailed approach to ensure that the program's goals are achieved such that:

- Design of Human-System Interfaces (HSIs) reduces the likelihood of error and provides for timely, clear error detection.
- Tasks can be accomplished within time and performance criteria.
- Allocation of Function (AOF) and proposed job design (staff complement and job roles) are such that a suitable level of human vigilance is ensured

and acceptable workload levels that minimize periods of human underload and overload is provided.
- Presentation of information supports a high degree of situational awareness of the state of the plant and actions required.
- HSI design supports the capability of personnel to recover from previous decisions and actions that did not achieve intended results.
- Application of ergonomic principles to working areas and their environments ensure these areas are safe and designed to support performance of required tasks.

Human Factors uses a graded approach to provide the appropriate focus for analysis and design with regards to human interactions within the systems. The grading of Human Actions for the BWRX-300 project is based on three risk categories:

- Nuclear Safety
- Personnel Safety
- Asset Protection

The HFE program coordinates the Task Analysis, HSI Design, Procedures Development, and Training and Qualification Program Development with the Human Reliability Analysis with the goal of ensuring the interfaces are designed:

- To a set of encompassing criteria that have been compiled from relevant regulatory guidelines and industry standards
- With characteristics that support task needs for the given user population
- To better quantify human reliability and establish enhancements to designs and supporting procedures and training that improve reliability where they are found to be significant risk contributors

The HSI Design establishes HSI features that support monitoring to aid users in recognizing the potential for abnormal conditions and to initiate actions for recovery to prevent severe consequences. The Task Analysis also analyzes accident response to effect HSI, procedures and training aimed at mitigating the consequences of adverse events should they occur.

The BWRX-300 HSI is designed to facilitate the operator's awareness and access to important status information and critical controls. In the case of a parameter in alarm or events that trigger a series of alarms, the alarm management system will be designed to display relevant alarms and information, as developed as a part of the HSI design process. For further details, please refer to PSAR Chapter 18, Section 18.2 [R-6].

HFE inputs to the design incorporate considerations for maintenance, inspection, access and egress, labels and signs, and workplace design to address the design of working areas and environments. For further details, please refer to PSAR Chapter 18 Sections 18.1, 18.2, and 18.3.

Radiation Protection

The systematic application of the ALARA philosophy to the DNNP BWRX-300 design establishes the design criteria required to control radiological contamination and reduce occupational exposure during plant operation and maintenance, decommissioning, and in the event of an accident.

The radiation protection objectives for the DNNP BWRX-300 design are:

- Radiation exposures must be kept within regulatory limits
- Radiation exposures must be kept ALARA, social and economic factors being taken into account. Provisions are made for the mitigation of the radiological consequences of any accidents considered in the design

General design considerations and methods employed to maintain in-plant radiation exposures ALARA include:

- Reducing the necessity and amount of time spent in radiation areas for personnel
- Reducing radiation levels in routinely occupied plant areas
- Controlling contamination

Equipment and facility designs are evaluated for maintaining exposures ALARA during plant operations. Events considered include normal operation maintenance and repairs, refueling operations and fuel storage, in-service inspection and calibrations, radioactive waste handling and disposal, etc. For more details on ALARA, and the control of radiation and effluent paths, refer to Section 4.5.4 and 4.9, as well as PSAR Chapter 12 [R-6].

Robustness against malevolent acts

The BWRX-300 design provides physical features to ensure the facility is robustly protected against malevolent acts to prevent potential release of radioactivity or energy to the public and environment.

The BWRX-300 development includes a security by design approach from the early stages of design that uses sound engineering principles to demonstrate that, within an acceptable margin of confidence, sufficient capabilities are available to perform the above functions over a wide range of threats.

For further details, refer to PSAR, Chapter 3, Section 3.5 [R-6], and Darlington BWRX-300 Security Assessment [R-77].

Safeguards in the design and design process

The facility design incorporates all necessary features to comply with Canada's obligations arising from the safeguards agreement between Canada and the IAEA. These features have been described in the DIQ [R-82] which has been submitted to the IAEA through the CNSC. Further discussion on Safeguards can be found in Section 4.13 of this Application document.

Feedback into the design and design process from operating experience and safety research

The section describes how the relevant OPEX has been, and will be, considered for the BWRX-300 during all phases of design, construction, commissioning, operation, maintenance, and decommissioning.

The DNNP BWRX-300 design is the simplest BWR design since GE began developing nuclear reactors in 1955. In development of the BWRX-300 design, over 60 years of operational experience, testing, and experimental benchmarking of functional compliance has been applied in choosing the design safety functional features. GEH BWR plants operating worldwide provide OPEX that is maintained by GEH, US Institute of Nuclear Power Operation (INPO), IAEA, and WANO. The BWRX-300 design utilizes the GEH BWR and INPO BWR Operational databases in the SSC design.

As DA, GEH has established provisions for the incorporation of OPEX through Integrated Management Systems with oversight by OPG. The OPEX comes from a variety of sources including:

- Direct input
- GEH/GNF experience from the operating BWR and ABWR fleet
- INPO
- Electric Power Research Institute (EPRI)
- United States Department of Energy (USDOE)
- United States NRC
- CNSC

The operating experience review (OER) process identifies and documents operational experience related to risk-important human actions. The objectives of the OER process are to obtain information and lessons learned from experience to support design of BWRX-300 systems.

Industry Incident Reviews

Station Blackout events have historically been the most demanding for BWRs to cope with and have usually been the dominant sequence for Severe Accident scenarios. The BWRX-300 is an advanced passive reactor design that does not require active safety systems. The BWRX-300 design carries forward the passive ICS and containment cooling concepts from the ESBWR. The systems that support FSF and plant monitoring are designed to operate for a coping period of 72 hrs, without AC power, and without an intake structure that normally provides cooling water. The ICS pools and SFP have enough inventory to provide adequate decay heat removal and fuel cooling for at least 72 hrs, after which alternate water makeup sources (e.g., flexible mitigation/EME) are used to refill the pools. The PCCS is designed to passively limit containment pressure and temperature by transferring heat to the equipment pool. The demonstration of plant safety functions during a beyond design basis external event such as an earthquake that creates these conditions is typically part of the diverse and flexible coping strategies that form the basis for compliance of regulatory requirements related to the Fukushima tsunami event.

In April 2012, the INPO conducted an independent review of the Fukushima nuclear accident with the purpose of identifying operational and organizational lessons learned from the accident. The results of this review are well documented. For details please refer to PSAR Chapter 18, Section 18.2 [R-6].

Operability and maintainability

Operability is a primary consideration in the safety analysis and design which establishes design criteria and OLCs that meet applicable codes, standards and regulatory requirements to ensure the ongoing operability of safety critical systems for different plant states.

Maintainability is built into the BWRX-300 design by defining primary requirements for In-Service monitoring, tests, maintenance, repairs, and inspections and ensuring they have been followed. These requirements include accessibility, ALARA, aging management, testing, and maintenance. In cases where safety class SSCs cannot be designed to support the desirable testing, inspection, or monitoring schedules, one of the following approaches is taken:

- Proven alternative methods, such as surveillance of reference items or use of verified and validated calculation methods, are specified.
- Conservative safety margins are applied, or other appropriate precautions are taken, to compensate for possible unanticipated failures.

Operability and maintainability are fundamental aspects of the BWRX-300 facility design. For more detail, please see PSAR Chapter 3, Section 3.10 [R-6].

Control of foreign material

The BWRX-300 design will provide for the detection, exclusion, and removal of all foreign material and corrosion products that may cause component or system damage or may have an impact on safety.

Foreign material refers to any objects that are not intended to be present within a system. These objects can be introduced from different sources such as maintenance activities (e.g., tools and materials) or component degradation (e.g., failure of a sub-component such as a gasket that is then spread throughout a system).

Instrumentation is used to detect changes in system pressure and flow resulting from foreign material allowing for timely operator actions to prevent adverse conditions. Exclusion (or removal) of foreign material is achieved primarily by installation of filters or strainers at designated locations, so that as cooling water moves around the fuel and through filters and strainers, foreign material is filtered and screened out. Adequate cleanout drains will be provided to enable staff to clean out foreign material.

Some examples of design features that help to control foreign material in the BWRX-300 system are as follows:

- Each main steam turbine stop valve contains a steam strainer and screens to prevent foreign matter from entering the main steam turbine control valves and turbine
- The Condensate Filters and Demineralizer System (CFD) continuously removes dissolved and suspended solids and corrosion products from the condensate and from drains returned to the condenser hotwell, to limit accumulation of corrosion products in the cycle
- The Fuel Pool Cooling and Cleanup system (FPC) provides filtration and demineralization of the water in the Fuel Pool, reactor well, and equipment pool to remove small particulate matter and perform ionic cleanup

<u>Other Safety Functions</u>

In addition to the design principles discussed above, the BWRX-300 design incorporates the following approaches to ensure safety.

Simplicity in Design

Simplicity attributes are incorporated throughout the design of the structures, systems, and components (SSCs). This organizational commitment to simplicity for the design is made in combination with other fundamental design principles such as redundancy, independence, defence in depth, diversity, and determinism (predictability and repeatability).

An example of simplicity in design is the reduction in the number of required components. Reducing the number of components in the design improves not only safety and reliability, but also operability and maintainability. Simplicity is achieved by incorporating passive safety features. Complexity has been minimized while recognizing that complexity may be necessary in certain cases to enhance reliability or reduce the potential for human error.

Passive Safety Features

The BWRX-300 design established passive design features which do not require dependence on external sources of power or operator actions to perform their functions. For example, the ICS removes decay heat from the reactor passively without any loss of reactor coolant inventory when the main condenser is unavailable.

Safety Margin

The safety margin is the result of the conservative assumptions and rules applied to the design based upon the SSC capabilities in postulated scenarios that are more severe than those in the design basis. In addition, the DSA demonstrates that the challenges to the physical barriers do not exceed their physical capacity. Further details can be found in PSAR Chapter 15, Section 15.5 [R-6].

Decommissioning

Assessment at the design phase, of future facility decommissioning and dismantling activities, includes consideration of OPEX gained from the decommissioning of existing nuclear facilities, as well as those facilities that are in long-term safe storage. By incorporating consideration of decommissioning techniques at the design stage, OPG will achieve benefits of lower worker doses and reduced environmental impacts at the time where plant decommissioning is required.

The decommissioning process involves:

- Removing the used nuclear fuel from the reactor into the Fuel Pool for some years, and then placing it into onsite dry storage containers once the decay heat has dropped to the appropriate level
- Dismantling non-radioactive parts of the plant
- Dismantling systems or components containing radioactive products (e.g. the reactor vessel)
- Cleaning and/or dismantling contaminated materials from the facility
- Disposing contaminated materials either onsite or by shipping to an available waste processing, storage, or disposal facility

There are no unique issues for the decommissioning of the BWRX-300. The lessons learned from existing industry experience are already being considered in the BWRX-300 design, operation, and decommissioning strategies. At the time of the DNNP BWRX-300 decommissioning, the current generation of BWRs will all have been decommissioned which will provide additional OPEX. For further details, see Section 4.11.4 of this Application document and PSAR, Chapter 21 [R-6].

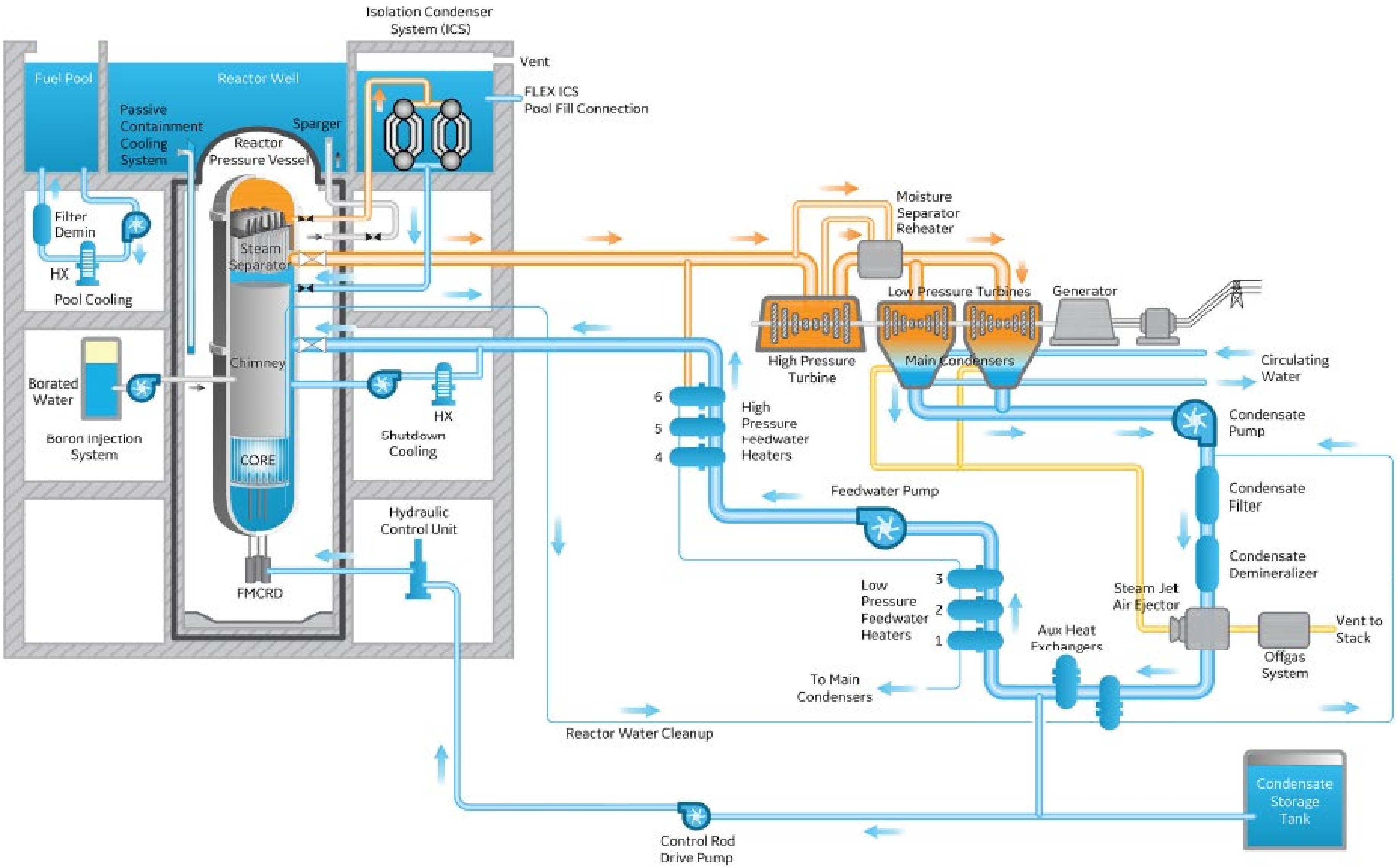


Figure 4.5-2: DNNP BWRX-300

4.5.4 Facility Design

The principal technical characteristics of the DNNP BWRX-300 SMR are provided in Chapter 1 of the PSAR [R-6]. Figure 4.5-2 shows a representation of the BWRX-300 design.

Site Access

The BWRX-300 design utilizes existing DNGS rail, roads, and waterways to the maximum extent practical. The site vehicle entrance is located on the north side of the plant. A road runs the perimeter of the power block to allow truck access to the TB.

The existing DNGS site includes a barge slip located at the far western end of the property. The largest SMR equipment load that will be off loaded at the barge slip is the RPV, and the longest single items to be transported by barge will be the RB polar crane bridges.

The existing heavy haul roadway at the DNGS facility will be used to transport heavy haul loads from the barge slip to the SMR footprint. This roadway runs west to east from the barge slip, along the southern border of the DNGS protected area, to the eastern plant boundary fence. The heavy haul road will be extended through the eastern fence along the southern portion of the SMR footprint and then turn north through the pre-assembly laydown to connect with Holt Road at the northwest end of the site. A second branch of the heavy haul roadway will continue east along the south side of the SMR footprint and turn north to the long-term construction laydown area.

Normal truck traffic, including both material deliveries and personnel transportation, will follow Highway 401 to Holt Road. Holt Road is the primary entrance to the site and closest proximity entrance to the DNNP site. There is an alternate entrance to DNNP, which can be occasionally utilized for heavy shipments to the existing facility, with permission from the St. Marys' Cement plant in Bowmanville.

Refer to PSAR Chapters 1 and 2 for additional details. Figure 4.5-3 below shows a plan view of the power block.

Main Buildings and Structures

The description of the main buildings and structures of the DNNP (e.g., power block, RB, TB, RWB, CB) is discussed below. Additional details of the buildings and structures can be found in Section 4.5.5.

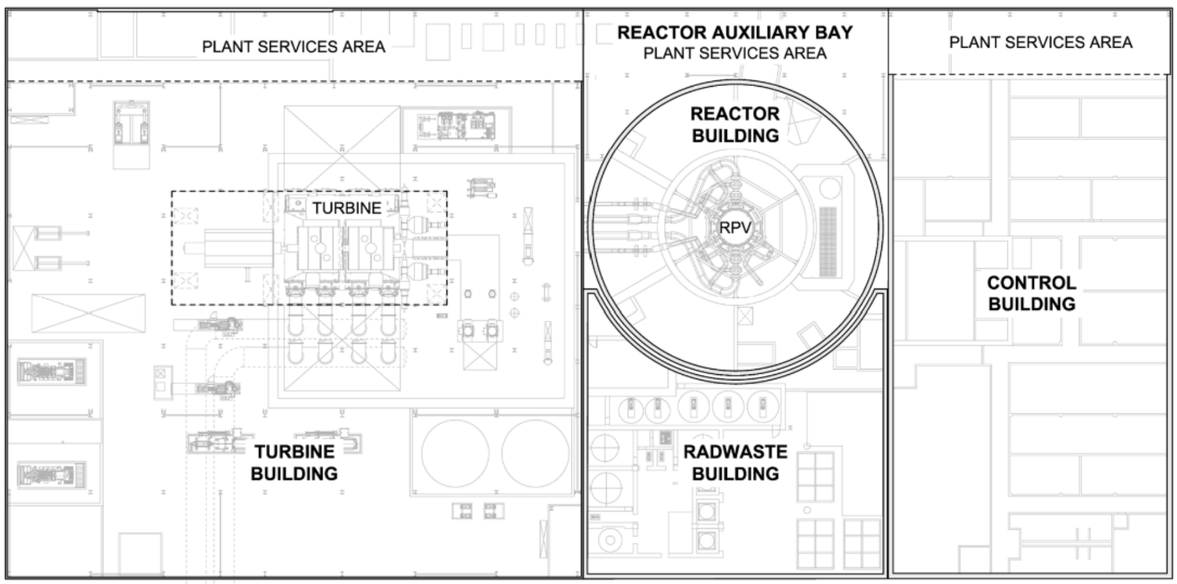


Figure 4.5-3: Powerblock plan view at Elevation 0

The **Reactor Building (RB)** is the only Safety Category 1 and Seismic Category A structure in the BWRX- 300 design. It is described in section 4.5.5 of this Application document.

The passive safety design, the embedded RB, enclosed robust containment, and included defence in depth measures provide mitigation of the effects of postulated events. Refer to Section 4.4 within this document and PSAR Chapter 3 [R-6] includes a listing of events and the associated mitigating features and analyzed effects.

To account for security and robustness against malevolent acts, shock damage footprint assessments for various potential commercial aircraft impact locations into the RB are conducted based on the sensitivity of the equipment to shock and vibration damage.

Critical equipment is in diverse locations to ensure that sufficient equipment remains operational to allow the reactor to be safely shut down and fuel cooled with containment remaining intact for all commercial aircraft impact locations. See PSAR Chapter 3 for details.

The **Radwaste Building (RWB)** houses rooms and equipment for handling, processing, and packaging liquid and solid radioactive wastes as well as the off-gas system charcoal adsorbers for radioactive wastes. The RWB is described in Section 4.5.5, and PSAR Chapter 1, Section 1.5 and Chapter 9B, Section 9B.3 [R-6].

The **Control Building (CB)** houses the MCR and some electrical, control and instrumentation equipment. It is also the entrance and exit for the BWRX-300 power block during normal operations. The CB is described in Section 4.5.5, and PSAR Chapter 1, Section 1.5 and Chapter 9B, Section 9B.3.

The **Turbine Building (TB)** houses most of the electrical equipment, power supplies and rotational and electrical power conversion equipment, including the main turbine and generator and associated support systems, and the main condenser, condensate and feedwater systems, etc. The TB is described in Section 4.5.5, and PSAR Chapter 1, Section 1.5 and Chapter 9B, Section 9B.3.

Hazards and Emergency Response

The BWRX-300 design accounts for physical protection measures against internal and external sabotage as well as provisions for protecting the capability of the monitoring and control of reactor facility parameters, of emergency management and response, and for mitigation and recovery measures to ensure the safety of workers and the public. Refer to PSAR Chapters 3 and 19.

The Security portion of the PSAR includes threat assessment, facility robustness evaluation, security program, security resources, etc. For more details on Security, refer to Section 4.12 of this Application document, and Darlington BWRX-300 Security Assessment [R-77].

To support overall emergency response, the BWRX-300 emergency response facilities meet the requirements of CNSC REGDOC-2.10.1, *Nuclear Emergency*

Preparedness and Response [R-83]. Further details are provided in PSAR Chapter 19 [R-6].

Fire protection at the plant uses a DiD concept that includes fire prevention, detection, control and extinguishing systems and equipment, administrative controls, procedures, trained personnel, and the shutdown capability. For information on actions taken in response to internal and external events, as well as egress routes, refer to PSAR Chapter 2, Section 2.10.

A robust and reliable MCR housed in the CB and SCR located in the RB provide environments to protect the operators so that they can remain at their posts and operate the reactor facility safely in all operational states and maintain the reactor in a safe condition under all accident conditions.

Refer to Section 6.6 for more details on habitability, and PSAR Chapter 6 'Engineering Safety Systems' and Chapter 15 'Safety Analysis' for details. Details on overall Emergency Response are provided in PSAR Chapter 19.

The accident monitoring details, and accident monitoring variables are discussed in Chapter 7, Table 7.3-2 of the PSAR.

Design and Control of Radiation

DNNP's BWRX-300 design utilizes the ALARA criteria to improve the layout of enclosures, accesses, and exits from controlled areas of the plant structures that confine radioactive material. The design of plant systems and components are specified to minimize personnel exposure to radiation during all modes of operation, AOOs, inspection, and maintenance.

The plant is designed to preclude the release of radioactive material to the environment that exceeds the limits of applicable regulations for normal operations, transients, and accidents. Radiation zones are established in all areas of the BWRX-300 plant as a function of both the access requirements of that area and the radiation sources in that area.

The DNNP BWRX-300 design features that minimize radioactive contamination include:

- Containment in areas where leaks and spills are most likely to occur
- Leak detection capability to provide prompt detection of leakage from SSCs

- Leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult, or inaccessible, to conduct regular inspections to avoid release of contamination
- Simplified design to reduce the number of components and structures within the plant that require maintenance and upkeep and may become contaminated during plant operation

For more details on the control of radiation, effluent paths and shielding design, refer to PSAR Chapter 12.

Engineered systems are deployed in the BWRX-300 design to monitor normal and accident conditions, as well as to provide radiological information, and alarm Operators under certain conditions. These systems include:

- The Process Radiation and Environmental Monitoring System (PREMS) consists of Process Radiation Monitoring (PRM), Continuous Air Monitors, Area Radiation Monitoring (ARM), and Containment Monitoring (CMon). The PRM system is for determining the content of radioactive material in various gaseous and liquid process and effluent streams. All radioactive release points/paths within the plant are identified and monitored by this system
- Continuous Air Monitors (CAM) are provided in the design. The portable CAMs provide a means to observe trends in airborne radioactivity concentrations.
- The Area Radiation Monitoring System (ARM) continuously measures, indicates, and records the gamma radiation levels at strategic locations throughout the plant.
- The Containment Monitoring (CMon) system is included in the design to continuously measure, indicate, and record the gamma radiation levels within the primary containment.
- Radioactive Process leak detection instrumentation is provided by the design to detect leaks from active systems.

For more details on the PREMS and other I&C systems, refer to Section 4.5.11 as well as PSAR Chapter 7, Chapter 11, Section 11.5 and Chapter 12, Section 12.4.

Systems that have been implemented in the DNNP BWRX-300 design to help control radiation include the Gaseous Waste Management System (Off-Gas

System) and Heating Ventilation and Air Conditioning (HVAC) system that are both discussed further in Section 4.5.14.

4.5.5 Structure design

This section describes buildings and structures associated with the DNNP BWRX-300.

All buildings and structures are designed to withstand internal and external hazards that have been defined in REGDOC-2.5.2 [R-21]. The BWRX-300 design considers natural and human induced external hazards as well as internal hazards. All anticipated loads, load combinations and performance design basis are considered in a graded manner commensurate to their importance to safety. The safety classification and seismic category of the structures are based on their intended functions.

Descriptions of Structures

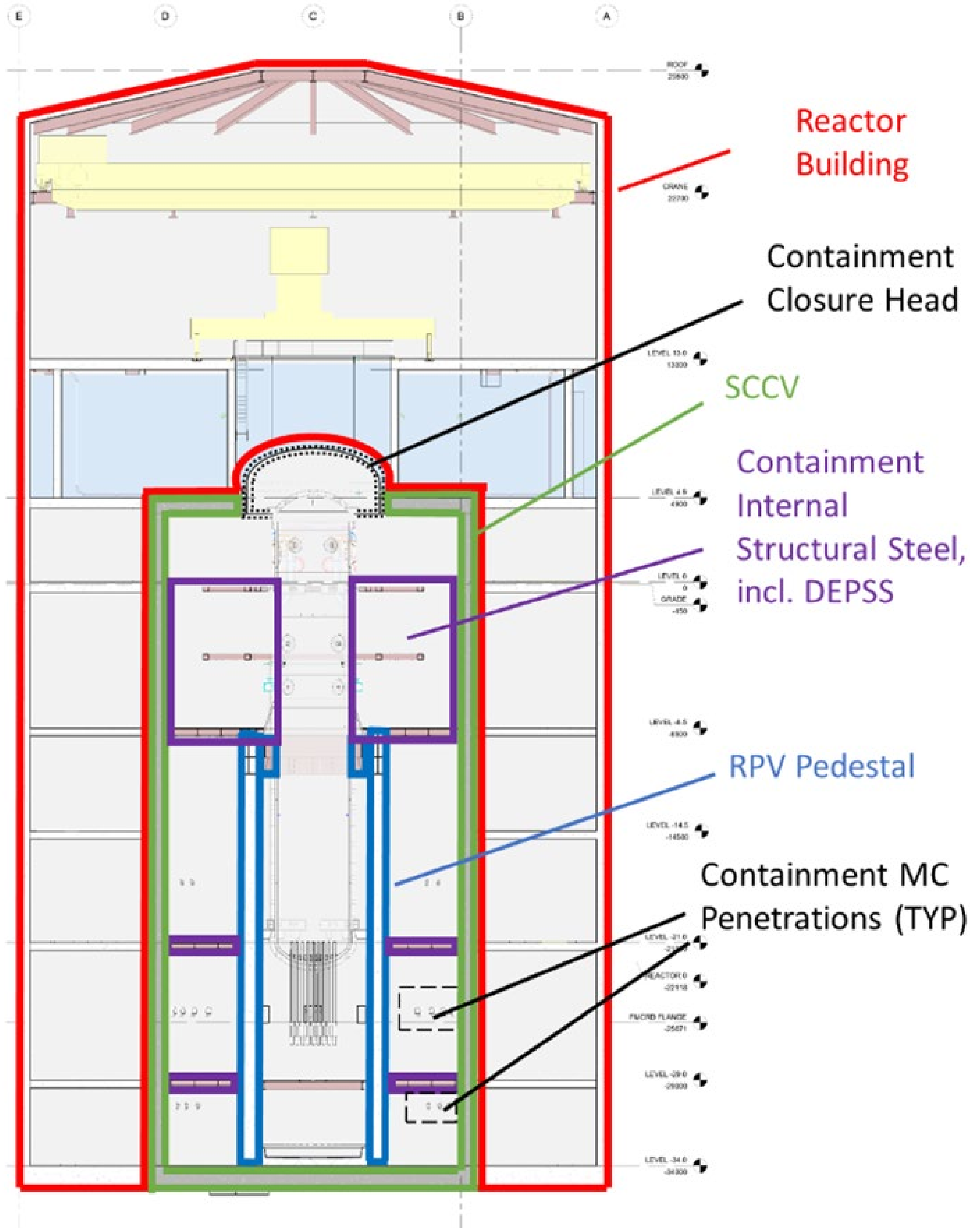


Figure 4.5-4: Structural Boundary of the BWRX-300 Containment, Containment Internal Structures and Reactor Building

Reactor Building Structure. The RB is a cylindrical shear wall building made of Steel Bricks™ floors and walls. Roof trusses supporting the Steel Bricks™ roof modules are composed of structural steel. The containment structure, containment internal structures and RB structure form an integrated RB structure.

The cylindrical-shaped, below grade portion of the RB structure houses the RPV, reactor support and safety systems, SCCV structure, containment internal structures, and most of the vital and non-vital power supplies and equipment. While a majority of the RPV and SCCV are located below grade, the top elevation of the RPV and SCCV top slab are above grade. In addition, the above grade portion of RB structure houses the refueling floor, refueling and fuel handling systems, SFP, and polar crane. The cylindrical integrated RB structure is founded on a common circular basemat that supports the containment, containment internal structures and RB. The common circular foundation mat is made of Steel Bricks™ modules, same as the supported superstructures. The containment boundary for the common basemat extends to the outer perimeter of the SCCV.

PSAR Chapter 9B, Section 9B.2 [R-6], provides the structural role and primary functions of RB and provides details of the RB design basis with performance and safety evaluation.

Containment Structure. The BWRX-300 uses a traditional containment system for the ultimate containment of radioactive materials for various postulated events. The BWRX-300 containment is a vertical cylinder comprised of the SCCV, consisting of Steel BricksTM cylindrical wall, mat foundation, and top slab, and a steel containment closure head. The containment structure is completely enclosed within the deeply embedded RB and includes personnel/equipment hatches, containment penetrations, and other safety components. PSAR section 9B.2 provides the structural role and primary functions of the containment structure and provides details of the design basis with performance and safety evaluation.

Containment Internal Structures (CIS). The BWRX-300 containment internal structures include the RPV pedestal, and the containment internal structural steel. which consists of the Drywell Equipment and Piping Support Structure (DEPSS) and the support floors at levels -21 m and - 29 m. PSAR Chapter 9B, Section 9B.2. defines the structural role and primary functions of CIS and provides the safety design basis with performance and safety evaluation.

Turbine Building Structure

The TB encloses the turbine generator, main condenser, condensate and feedwater systems, condensate purification system, off-gas system cooler and refrigerant dryer, turbine generator support system and bridge crane.

The TB structure is divided into three structural supporting systems:

- The TB shell structure which consists of a steel frame system with steel columns, beams/girders, roof bar joists, and floor/roof decks as gravity load carrying systems. The lateral force resisting system consists of braced frames and floor/roof decks as diaphragms.

- The Shielding wall area which consists of reinforced concrete shear walls surrounding and supporting the portion of the structure containing radioactive steam and water for shielding (turbines, condenser, off-gas lines, heater tanks, and associated piping).

- The reinforced concrete pedestal supporting the turbine, generator, and exciter within the TB shell structure and the shielding walls area. This turbine generator pedestal is structurally isolated from both the shielding walls and the TB shell structure for vibration control.

PSAR Chapter 9B, Section 9B.3 [R-6] provides the structural role and primary functions of the TB structure and provides details of the design basis with performance and safety evaluation.

Reactor Auxiliary Bay

The Reactor Auxiliary Bay is the structurally independent portion of the PLSA. The Reactor Auxiliary Bay structure consists of a one-story structural steel building. PSAR Chapter 9B, Section 9B.3.4 provides the structural role and primary functions of the Reactor Auxiliary Bay and provides details of the design basis with performance and safety evaluation.

Control Building Structure

The CB houses the MCR, Emergency Response Facilities (ERF), electrical, control and instrumentation equipment. The CB structure consists of a perimeter reinforced concrete wall, interior steel columns, beams/girders, roof bar joists, and roof deck as a gravity load carrying system. The lateral force resisting system consists of reinforced concrete shear walls and a composite roof deck as a diaphragm. PSAR Chapter 9B, Section 9B.3.3 [R-6] provides the structural role and primary functions of the CB structure and provides details of the design basis with performance and safety evaluation.

Radioactive Waste Building Structure

The RWB houses the off-gas system charcoal adsorbers, refueling water storage tanks, and rooms and equipment for handling, processing, and packaging liquid and solid radioactive wastes. Its main function consists of processing and housing liquid, solid and gaseous radioactive wastes. Its structure consists of reinforced concrete walls and floor slabs supported on a shallow reinforced concrete mat foundation with roof joists and composite roof decking. The lateral force resisting system of the structure consists of concrete shear walls, concrete floors, and a composite roof deck acting as a diaphragm. PSAR Chapter 9B, Section 9B.3, provides the structural role and primary functions of the Radioactive Waste Building and provides details of the design basis with performance and safety evaluation.

Pumphouse/Forebay and Tunnels

The Pumphouse/Forebay structure is designed to facilitate pump installation to support the operation of up to four SMR units at the DNNP site. The structure is composed of the forebay, pump bays, and electrical equipment rooms. Part of the Pumphouse/Forebay structure is enclosed in a metal wall panel clad steel building to provide weather protection for the pumps and electrical gear.

Tunneling includes vertical shafts, intake tunnel, and discharge tunnel and diffusers. The onshore vertical shafts are designed to facilitate the operation of up to four SMR units and facilitate the construction of the intake and discharge tunnels. The intake tunnel connects to the onshore vertical intake shaft to convey cooling water to the Pumphouse/Forebay. Similarly, the onshore vertical discharge shaft connects to the discharge tunnel to convey heated water through the discharge tunnel to the diffusers. The intake tunnel conveys cooling water from the lakebed intake structure to the onshore intake vertical shaft. The intake tunnel is located beneath Lake Ontario and is to be lined with concrete. The discharge tunnel conveys the discharge water from the onshore discharge vertical shaft to the discharge risers/diffusers. The discharge tunnel is located beneath Lake Ontario and the tunnel is to be lined with concrete.

Additional information on Pumphouse/Forebay and Tunnels structures can be found in PSAR Chapter 9B, Section 9B.3 [R-6].

Building Structures Classification and Boundary

PSAR Chapter 3 Section 3.2, provides the general classification and basis for classification, of all SSCs and includes quality-category, seismic category, safety category, and the required design rules in establishing these qualifications.

Protection Against External and Internal Hazards

The BWRX-300 design considers natural and human induced external hazards as specified in REGDOC-2.5.2 [R-21]. Chapter 3 of this application document summarizes the external and internal hazards.

The determination of the natural external hazards relies on the collection of the geotechnical, seismological, hydrological, hydrogeological, and meteorological reference data which is detailed in PSAR Chapter 2. Human induced hazards, described in REGDOC-2.5.2, have been evaluated and considered in the design. For more details see PSAR Chapter 3, Section 3.3.

Internal hazards that may have an impact on the ability of the BWRX-300 safety class SSCs to achieve a safe shutdown will be addressed in the design. For details of possible internal hazards and mitigation methods, see PSAR Chapter 3, Section 3.4.

Seismic design and Analysis

The seismic design of the DNNP BWRX-300 ensures robustness against Design Basis Earthquakes (DBE) as well as low probability seismic events that are beyond design basis. The design incorporates the site-specific geotechnical inputs.

SSCs that are classified as Seismic Category A and Seismic Category B are seismically qualified to withstand the effects of a DBE in accordance with provisions of CSA N289.1, *General requirements for seismic design and qualification* [R-84]. The seismic evaluation of BWRX-300 design ensures the ability of the CB, TB, Reactor Auxiliary Bay, and RWB to prevent adverse interactions with the RB structure and SSCs under DBE loading.

For more details on seismic design, see PSAR Chapter 3, Section 3.3 [R-6].

Extreme Weather Conditions (Wind, Tornado and Precipitation Loading)

The RB, CB, TB, Reactor Auxiliary Bay, and RWB are all designed to mitigate the impact of extreme weather conditions including temperature and humidity, rain, snow and ice, wind, tornado, lightning, extreme wind interaction for loading and effects on structures. For details see PSAR Chapter 3, Section 3.3.

Extreme Hydrological Conditions (Water Level /Flood Design)

The DNNP BWRX-300 design includes protective features that are used to mitigate or eliminate the adverse consequences of flooding due to internal and external sources. The integrated RB structure is designed to withstand the maximum external flood and groundwater levels specified for the plant. These design features meet the requirements of the applicable standard CSA N291, *Requirements for safety related structures for nuclear power plants* [R-85] with all protection measures considered for integrated RB structure against underground water relevant hazards and loads

Because plant grade is above design flood level, the Power Block structures remain accessible during postulated flood events. Thus, no emergency actions are required due to flooding to ensure the safe operation of the BWRX-300 plant. Refer to PSAR Chapter 3, Section 3.3 for details.

Aircraft crash

The BWRX-300 design includes protective features that are used to mitigate the potential impact loads due to an aircraft crashing on to the buildings. The site context conditions, and evaluation are carried out to determine the acceptable low probability and the bounded levels of aircraft loads. PSAR Chapter 3, Section 3.3 discusses non-malevolent, general aviation crashes.

Missile (internal and external) and Impact Loads

The DNNP BWRX-300 structures and barriers are designed to withstand all applicable impactive loads per the requirements of CSA N291 [R-85].

Types of missile loads considered per PSAR Chapter 3 Section 3.3 include:

- Missiles Generated by Extreme Winds
- Site Proximity Missiles (Except Aircraft)

- Structures, Systems and Components to be Protected from Externally Generated Missiles

The building floors and support structures are designed to withstand the drop of heavy loads. An exhaustive list of lifted loads and lift heights are provided in PSAR Chapter 3, Section 3.4 [R-6].

Fires, Explosions and Toxic Gases (Internal & external)

The BWRX-300 design includes protective features that are used to mitigate the effects of potential accidents such as fires, explosions and toxic gases and ensure the safety of plant personnel. When applicable, peak equivalent pressure wave from these explosions is determined and applied in the design of structures. For further details see PSAR Chapter 3, Section 3.3, for external and PSAR Section 3.4 for internal fires, explosions and toxic gases.

Other External Hazards

Biological phenomena (e.g. designing the Pumphouse/forebay structure against clogging by algae and fish, etc.), collisions of floating objects with water intakes, and malevolent acts are accounted for in the design. For further details see PSAR Section 3.3.

Other Design Considerations

Other design considerations, such as radiation shielding and leak-tightness, are taken into account in the structure design to ensure they fulfill their overall design requirements and intent.

Radiation Shielding

The shielding requirements in the plant are designed to perform the following functions:

- Limit radiation exposure to the general public, plant personnel, contractors, and visitors to levels that are ALARA and within CNSC requirements
- Limit radiation exposure to personnel, in the unlikely event of an accident, to levels that are ALARA and conform to the limits specified to ensure that plant personnel remain available to ensure the plant is maintained in a safe condition during an accident

- Limit the radiation exposure of critical components within vendor specified radiation tolerances, to assure that component performance and design life are not impaired

For the detailed information on ALARA Structures Criteria refer to PSAR Chapter 12, Section 12.3 [R-6].

Leak-tightness and Liners for Structural Components

The following measures are adopted to prevent dispersion of contamination and radiation passing through holes in design of walls and structures for leak-tightness. The enclosures that house equipment with large volumes of radioactive liquids (tanks, heat exchangers, filters, etc.) are leak tight.

The gaseous radioactive waste system is leak-tight to minimize escape of radioactive gases and potentially flammable hydrogen. Penetrations in walls and structures used to separate enclosures and buildings are located (and sealed, if applicable) to prevent through-flow of radioactive fluids. Additionally, suitable containment measures are adopted to prevent dispersion in the event of liquid overflow (curbs, sloping floor drains). For details see PSAR Section 12.3.4.

The liner plates are required for certain structural components. These are addressed in PSAR Chapter 3, Section 3.5, chapter 9B Section 9B.2 and Chapter 12 12.3.

4.5.6 System design

The System Design SCA requires relevant information for the system description, pressure-retaining SSCs, equipment environmental qualification, electromagnetic interference, seismic qualification, and fire safety/fire protection to be included in the Application.

<u>System Description</u>

BWRX-300 System Design Descriptions (SDDs) are prepared for each system. The requirements and bases in the SDDs are considered the governing design requirements to be satisfied by a system. The SDDs identify the components that support the system DL functions. System design evaluations are performed in parallel with other activities to ensure systems support operational objectives.

Pressure- or fluid-retaining structures, systems and components

The DNNP BWRX-300 leverages evolutionary BWR design centered on protecting the reactor core and ensuring accident mitigation capability through a layered approach that places high importance on pressure boundary integrity.

A pressure boundary standard for the DNNP BWRX-300 will specify technical requirements for the design, procurement, fabrication, installation, modification, repair, replacement, testing, examination, and inspection of, and other work related to, pressure-retaining systems, components, and supports over the service life of DNNP's BWRX-300 SMR. The standard will comply with applicable requirements of CSA N285.0, *General requirements for pressure-retaining systems and components in CANDU nuclear power plants/ Material Standards for reactor components for CANDU nuclear power plants* [R-87] and associated ASME codes [R-86].

The mechanical pressure retaining systems and components will be designed to meet the requirements of ASME codes and standards. Design and fabrication requirements for each quality group are provided in the PSAR Chapter 5 [R-6].

Overpressure protection for the RCPB is provided through the NBS, ICS, and the SC1 I&C system to limit peak pressure during AOOs. For other than the RCPB, systems are primarily protected from overpressure by safety valves (SVs) or relief valves (RVs). Overpressure analyses is addressed in PSAR Chapter 5.

Various leak detection methods are provided to ensure leaks are detected before break/failure and, to the extent practical, identify the location of the source of leakage. Differential pressure cells and pressure sensors are provided for line break detection. Other methods of leak detection are based on detection of radioactivity and sump level detection. The design provides for detection capability such that a potential leak can be discovered by the containment leak detection system and isolated to ensure that the integrity of the system is maintained. See PSAR Chapter 6, Section 6.5.

The plant is designed for protection against piping failures inside and outside containment to assure no loss of essential SSCs and to ensure safe shut down. The design also addresses high-energy and moderate-energy fluid system piping breaks located inside and outside of containment. For detailed information see to

PSAR Section 3.4.4 and Section 2.7 for Pressure Boundary Postulated High Energy Piping Breaks.

The BWRX-300 reactor incorporates isolation valves connected to the RPV. The RPV isolation concept consists of two RPV isolation valves in series. Each of the RPV isolation valves are independently able to isolate the line to the RPV using flanged connections. The two in-series containment isolation valves also function as RPV isolation valves and are mounted integrally to the RPV inside containment where they are protected from outside environmental conditions that may result from a failure in the closed loop outside containment (see PSAR section 6.5.2).

There are minimal interfaces between the high pressure and lower design pressure systems that include closed valves or check valves in series, to prevent exposure of low-pressure piping to high pressure sources. Automatic isolation actions are avoided as a method to protect low-pressure design piping from high-pressure sources.

GEH has a mature pressure boundary program and associated implementation processes and procedures.

As per standard OPG practice for its existing nuclear fleet, a service agreement with the Technical Standards Safety Authority (TSSA) will be established to cover pressure boundary activities.

Equipment Qualification

Equipment Qualification (EQ) is the process carried out to ensure SSC can perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform. The objective of the EQ program is to assure that all applicable SC systems, equipment, components, protective barriers, and structures are qualified to perform their safety functions under the environmental conditions defined by DNNP DBAs.

Specifically, the EQ program considers equipment located in bounding harsh environment DBAs. Harsh environment includes thermodynamic conditions (temperature, pressure, and relative humidity), radiation parameters (radiation type, dose rates, and total integrated dose) and chemical or submergence parameters, as applicable. SSCs located in mild environments, or those SSCs not required to perform an active safety function in a harsh environment are outside the scope of the EQ program.

PSAR Section 3.9 describes the process in which equipment requiring EQ is identified, procured and evaluated.

Maintenance of the EQ program will ensure that aging is managed, obsolescence is considered, and that qualification configurations are maintained through the operating life of the plant, including any life extensions and decommissioning, as applicable.

Electromagnetic interference

Internal electromagnetic interference is caused by induction or radiation from installed equipment. Complying with CNSC REGDOC-2.5.2, Section 7.5, safety class SSC are protected against electromagnetic interference to enable them to perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform.

Qualification requirements for protection against electromagnetic interference are presented in PSAR Section 3.9.5. Plant grounding, lightning protection and electromagnetic compatibility systems and their design requirements are discussed in PSAR Chapter 8, Section 8.6 [R-6].

BWRX-300 plant I&C systems are designed to prevent effects from man-made magnetic or electric fields, such as fields from radar, radio, and mobile phones, from impacting their function. Electrical power systems are also designed to cope with grid disturbances, including conditions caused by solar flares. Communication systems are shielded as necessary from the adverse effects of electromagnetic interference and radio frequency interference.

Some design features that are implemented to mitigate the effects of electromagnetic interference include equipment grounding, cable and housing shielding, physical separation, filters, fuses, suppressors, and other components and techniques.

The industry codes and standards used to establish an electromagnetic compatible environment for the EMC/SWC for equipment qualification, applicable to electrical, I&C and communication SSCs and equipment can be found in PSAR Section 1.11 [R-6].

For more information on Electrical systems for the BWRX-300, refer to Section 4.5.10 of this application, and for more information on Instrumentation, Control and Communication systems, refer to Section 4.5.11 of this application.

<u>Seismic Qualification</u>

General

The DNNP PSAR (PSAR Chapter 3, Section 3.9.3), describes how seismic qualification protects SSCs from earthquake damage.

The seismic qualification approach includes the determination of seismic categories of SSCs by safety classifications and the associated methodologies, tests and analyses.

SSC Seismic Classifications and Categories

PSAR Chapter 3 provides the seismic classification and categories for SSCs. The seismic category sets out SSC requirements during and after a seismic event and governs how the SSC is designed and qualified.

The SSCs of the nuclear facilities of the BWRX-300 are designed, qualified, and tested in accordance with equipment and component specifications commensurate with their importance based on their safety classification and the functions they support.

The DNNP's BWRX-300 design incorporates a seismic monitoring system to allow the operator to determine if a seismic event is above or below the DBE. This system supports decision-making related to operation and inspection of plant equipment following a seismic event. See PSAR Section 7.3.4.2.

PSAR Section 3.9.3 provides details on the methodologies for seismic qualification and design as well as documentation of seismic margin with fragility levels.

4.5.7 Fire safety and fire protection systems

The BWRX-300 design leverages the existing BWR fire protection design including that of external buildings and SSCs integral to plant operation. The BWRX-300 design program for fire protection addresses the following fire protection requirements:

- Prevent the initiation of fires
- Limit the propagation and effects of fires that do occur by quickly detecting and suppressing fires to limit damage and confining the spread of fires and fire by-products that have not been extinguished
- Prevent the loss of redundancy in Safety Class and safety support systems due to fires
- Ensure the safe shutdown capabilities in the event of a fire by achieving and maintaining the reactor in subcritical conditions, and achieving and maintaining decay heat removal
- Ensure that monitoring of critical safety parameters remains available in the event of a fire
- Prevent exposure, uncontrolled release, or dispersion of hazardous substances, nuclear material, or radioactive material due to fires
- Prevent the detrimental effects of event mitigation efforts, both inside and outside of containment
- Ensure structural sufficiency and stability in the event of fire

Provisions for prevention of explosions and fires are addressed in the DNNP BWRX-300 design. The design uses the concept of DiD to achieve the required degree of reactor safety for a given SC, by using administrative controls, FPSs and features, and safe shutdown capability. These DiD principles achieve the following objectives:

- Prevent fires from starting;
- Rapidly detect, control, and extinguish promptly those fires that do occur;
- Provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities does not prevent the safe shutdown of the plant and does not significantly increase the risk of radioactive release to the environment.

The SSCs are designed and located to minimize the probability and effects of fires and explosions consistent with safety requirements. Non-combustible and heat-resistant materials are used wherever practical, particularly in the containment and the MCR. Fire detection and firefighting systems of appropriate capacity and capability are provided and designed to minimize the adverse effects of fires.

Control and mitigation of fires and explosions

Provisions for control and mitigation of fires and explosions is addressed in the DNNP BWRX-300 FP design. The fire control means consist of water based manual and automatic fire suppression systems (including fire hydrants, standpipes and hose-systems, sprinkler systems), fire extinguishers, water sprays for charcoal filters, foam system for concentrated fuel oil or lube oil hazards, smoke detection and fire alarm system, fire barriers and ventilation systems.

The design bases of the BWRX-300 fire protection program and the FPS, including the design features, are discussed in detail in Section 9A.6 of the PSAR [R-6].

In support of the LTC Application, the following documents are submitted:

- Preliminary Fire Safe Shutdown Analysis Report [R-89]
- Preliminary Fire Hazards Assessment Report [R-90]
- Fire Protection System Preliminary Code Compliance Review Report [R-91]
- Independent Third-Party Review of the Preliminary Fire Protection Design [R-92]

A final Code Compliance Review (CCR) will be completed as part of the FP Program development. This document will define code compliance or list exemptions or alternatives that depart from Canadian codes and standards and how the intent of the requirements is met using equivalent or alternative means. See PSAR section 9A.6.2

4.5.8 Reactor and reactor coolant system

The BWRX-300 is an evolutionary design that incorporates proven technologies, materials, evaluation to and methods to support its reactor and coolant systems' design. The following sections provide a summary of how the BWRX-300 reactor and coolant systems meet the requirements of REGDOC-1.1.2 [R-19]. This section focuses on the physical design, design basis requirements, and performance characteristics as well as the evaluation methods and tools used in the design.

Design of the Fuel System

The following Section describes the design of the BWRX-300 fuel system. This Section is divided into the following areas:

- Fuel design description
- Fuel design basis requirements
- Fuel design methods, computer codes, and evaluations
- Fuel manufacture, testing, inspection, and surveillance plans
- In-core fuel management

Detailed information regarding design of the BWRX-300 fuel system can be found in PSAR Section 4.2 Fuel Design, which summarizes mechanical and thermal-mechanical design bases and analysis. Additional information on relation of the fuel to core nuclear design, thermal and hydraulic design, and flow stability can be found in PSAR Sections 4.3, 4.4, and 4.7 respectively [R-6]. The reference document for regulatory guidance on GNF2 fuel qualification is NEDE-24011, General Electric Standard Application for Reactor Fuel (GESTAR II) [R-93].

Fuel Design Description

The reference fuel design for the BWRX-300 is the proven GNF2 design. The GNF2 fuel is the 3rd evolution of the GNF 10x10 fuel assembly and was first deployed for forced circulation BWRs in 2007. Currently, the majority of BWRs operating in the world use GNF2 fuel in their reactors with tens of thousands of bundles replaced in more than 100 reload cycles. This diverse range of reactor applications demonstrates the robustness of the GNF2 design. The GNF2 fuel cladding integrity safety limit is calculated so that no significant damage occurs during normal operation and AOOs on a cycle-independent basis. Additionally, the core thermal-hydraulic design establishes thermal hydraulic operating limits used in assuring safety margin is maintained in accordance with REGDOC 2.5.2

The BWRX-300 fuel system itself consists of 240 fuel assemblies and 57 reactivity control assemblies (i.e., control rods). The fuel assembly is discussed in this section while the reactivity control assembly is covered in Section 4.5.8 Reactivity control system. The BWRX-300 GNF2 fuel assembly is comprised of a fuel bundle, a channel that surrounds the fuel bundle, and a channel fastener that attaches the bundle to the channel. Please refer to Figures 4.5-5 and 4.5-6 below.

The major components in a GNF2 fuel bundle are:

- Fuel rods
- Water rods

- Upper and lower tie plates
- Spacers

A GNF2 fuel bundle has 92 fuel rod locations and four types of fuel rods: standard full-length rods (FLR), part length rods (PLR), tie rods, and rods with a burnable neutron absorber.

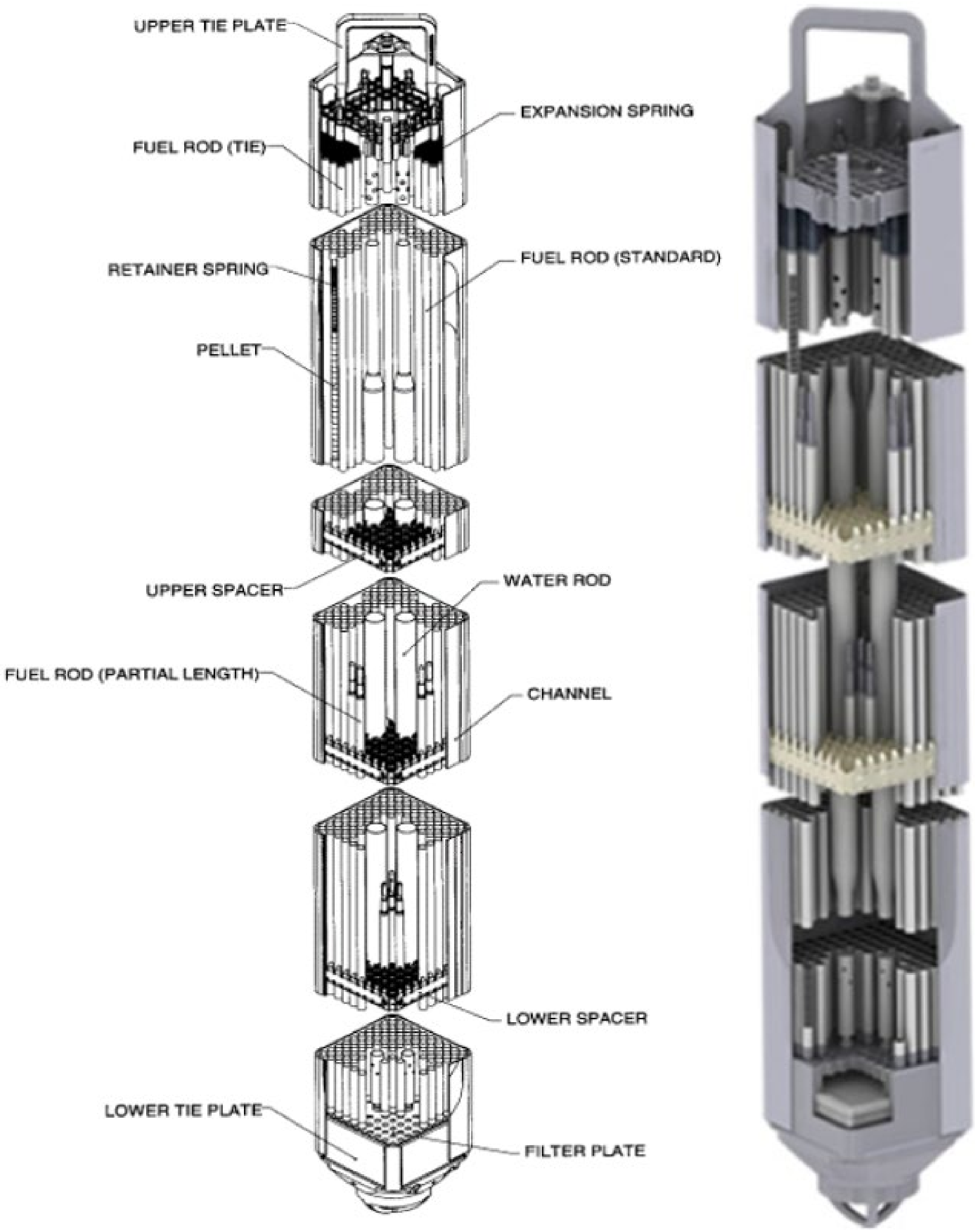


Figure4.5-5: GNF2 Fuel Assembly and Channel

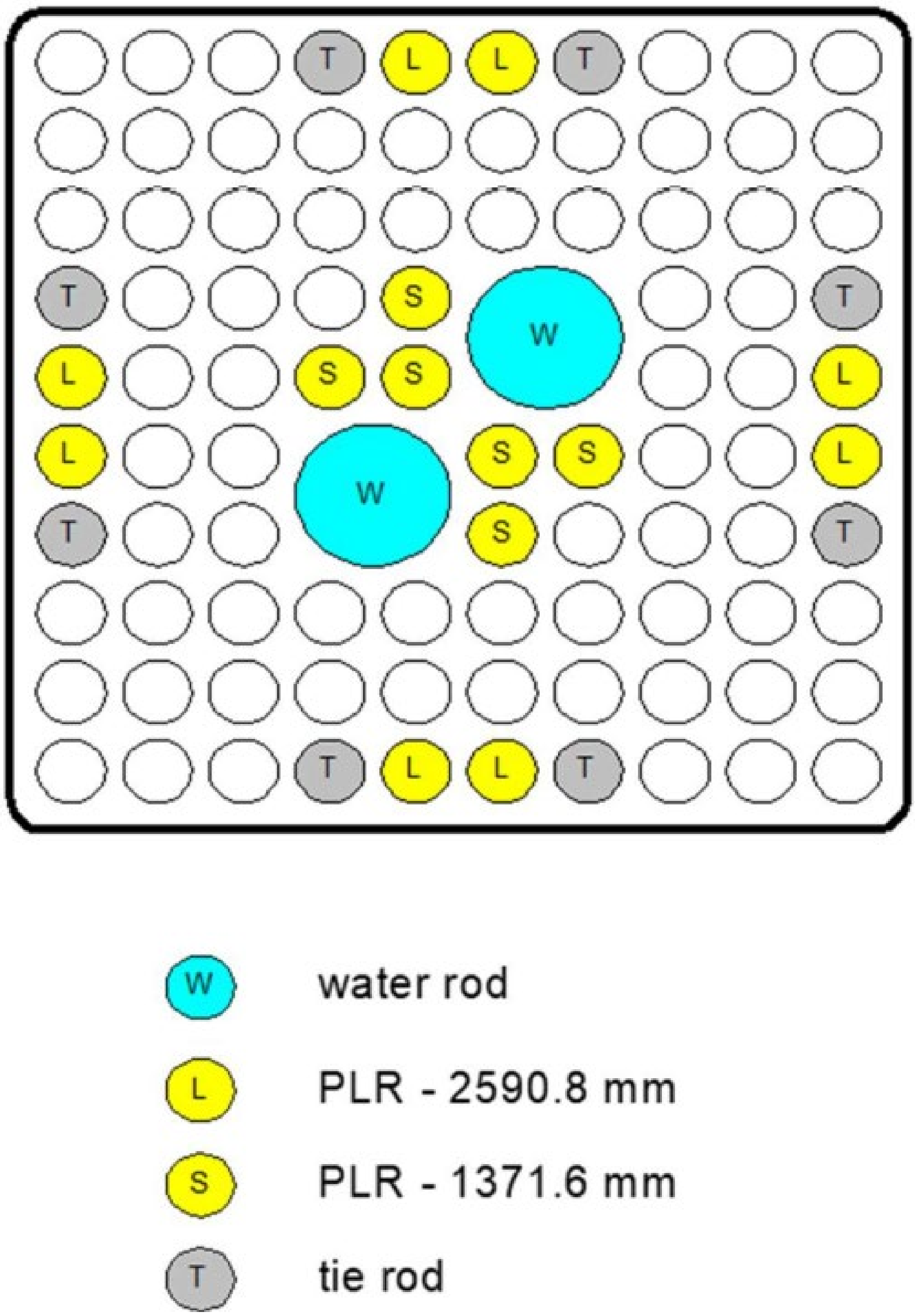


Figure 4.5-6: GNF2 Fuel Assembly Lattice

The fuel is uranium oxide (UO_2) pellets with a maximum enrichment value of less than 5%.

Two large water rods are centrally located inside the fuel bundle. Water rods are hollow Zircaloy-2 tubes with holes at the inlet and outlet to allow for coolant flow. The primary function of the water rod is to improve nuclear efficiency, however, it also improves cold shutdown margins, flattens power distribution, and acts to physically support the spacers.

Some of the fuel rods are called *gadolinia rods* since they contain both Uranium Oxide and gadolinium oxide. Gadolinium is a burnable neutron absorber that reduces as the fuel is irradiated. These rods are used as a form of reactivity control and to aid in shaping the initial power distribution of the core.

Spacers are used to prevent fretting wear on the fuel rods due to vibration. Eight spacers are distributed throughout the fuel bundle with a non-uniform axial spacing to optimize critical power performance.

The channel comprises part of the GNF2 fuel assembly. The channel is open at the bottom and slides over the lower tie plate. The top of the channel has welded tabs which support the weight of the channel on the upper tie plate. The channel performs the following functions:

- Forms a coolant flow path on the outer periphery of the fuel bundle
- Provides a surface for control rod guidance in the reactor core
- Provides structural lateral stiffness for the fuel bundle
- Controls in conjunction with the lower tie plate, coolant bypass flow at the interface of the channel and lower tie plate

Fuel Design Basis Requirements

For the BWRX-300 plant, fuel licensing criteria and fuel assembly design basis are specified in PSAR Chapter 4. Additional discussion regarding fuel mechanical, fuel thermal-mechanical, nuclear, and thermal-hydraulic design bases can be found in PSAR Section 4.2, Section 4.3, and Section 4.4 [R-6].

Fuel Design Methods, Computer Codes, and Evaluations

The GNF2 fuel design is the result of more than 50 years of design, fabrication, and operational experience. The fuel design process includes engineering methods, analyses, and tests that are mature and applicable to the BWRX-300. The PRIME computer code is the key method for performing fuel thermal mechanical design analyses and covers the following areas:

- Fuel rod internal pressure
- Fuel melting
- Cladding strain
- Cladding fatigue
- Cladding collapse
- Fuel rod stresses
- Cladding corrosion
- Fuel rod hydrogen content
- Pellet-cladding interaction (PCI)

A summary of the fuel design evaluation methods is provided in PSAR Section 4.2 [R-6], including but not limited to:

- Worst tolerance analyses
- Statistical analyses
- Fuel rod internal pressure
- Cladding fatigue analysis
- Thermal and mechanical overpowers

Additional information on fuel design methods, computer codes, and evaluations including the PRIME technical bases, qualification, and application methodology can be found in PSAR Section 4.2 Fuel Design. Information on methods, computer codes, and evaluations for areas related to fuel can be found in PSAR Section 4.3 Core Nuclear Design, Section 4.4 Thermal and Hydraulic Design, and Section 4.7 Flow Stability Evaluation.

Fuel Manufacture, Testing, Inspection and Surveillance Plans

Manufacturing of GNF nuclear fuel consists of two separate operations: fuel components operation and fuel manufacturing operation. The fuel components operation produces the fuel rod cladding, water rods, and channels used in a GNF2 fuel assembly. The raw material used in these components is provided by qualified suppliers and then undergoes fabrication into the finished part.

The fuel manufacturing operation fabricates the fuel pellets from uranium oxide powder and is then combined with additional hardware to assemble fuel rods and fuel bundles. The major processes for the fuel manufacturing operation are:

- Conversion of UF_6 to UO_2 powder
- Powder preparation for processing to pellets
- Fuel pellet processing
- Fuel rod fabrication
- Fuel bundle assembly
- Fuel bundle shipping

See PSAR Chapter 4 for additional information on fuel quality standard [R-6]

GEH has an active program for the surveillance of production of GNF2 fuel, which has been reviewed and approved by the USNRC. For the manufacture and inspection of fuel the key QC areas are:

- Material and component procurement
- Fabrication and assembly of components and systems
- Inspection and testing
- Cleaning, packaging, and shipping
- Installation and erection of systems and components
- Pre-operational and start-up testing

In-Core Fuel Management

Management of in-core fuel is summarized in PSAR Section 4.3 and 13.3.4. These sections describe how core and fuel management guidelines are implemented through operational methods to mitigate and reduce fuel performance risks.

Design of the Reactor Internals

Information regarding the design basis of the reactor internals, physical and chemical properties of the fuel components, surveillance and inspection programs and programs to monitor the behaviour of the core can be found in PSAR Chapters 4 & 5 [R-6].

The BWRX-300 RPV assembly consists of the vessel with nozzle integral RIVs and its other appurtenances, a removable head, the reactor internals and supports and instrumentation. The RPV, together with its internals, provides guidance and support for the FMCRDs.

The RPV is a vertical, cylindrical pressure vessel fabricated from forged rings or rolled and welded plate rings joined together, with a removable top head by use of a head flange, seals, and bolting. The RPV also includes penetrations, nozzles including RIVs, and reactor internals support.

The reactor internals design of the BWRX-300 is similar to several other operating BWRs. Contained within the RPV are the following major internal components:

- Core Components (fuel assemblies, control rods and nuclear instrumentation)
- Core support structures (shroud, shroud support, top guide, core plate, control rod guide , CRD housing and fuel support)
- Chimney
- Chimney head and steam separator assembly
- Steam dryer assembly

The following Figure 4.5-7 is a visual representation of the BWRX-300 and the major internal components.

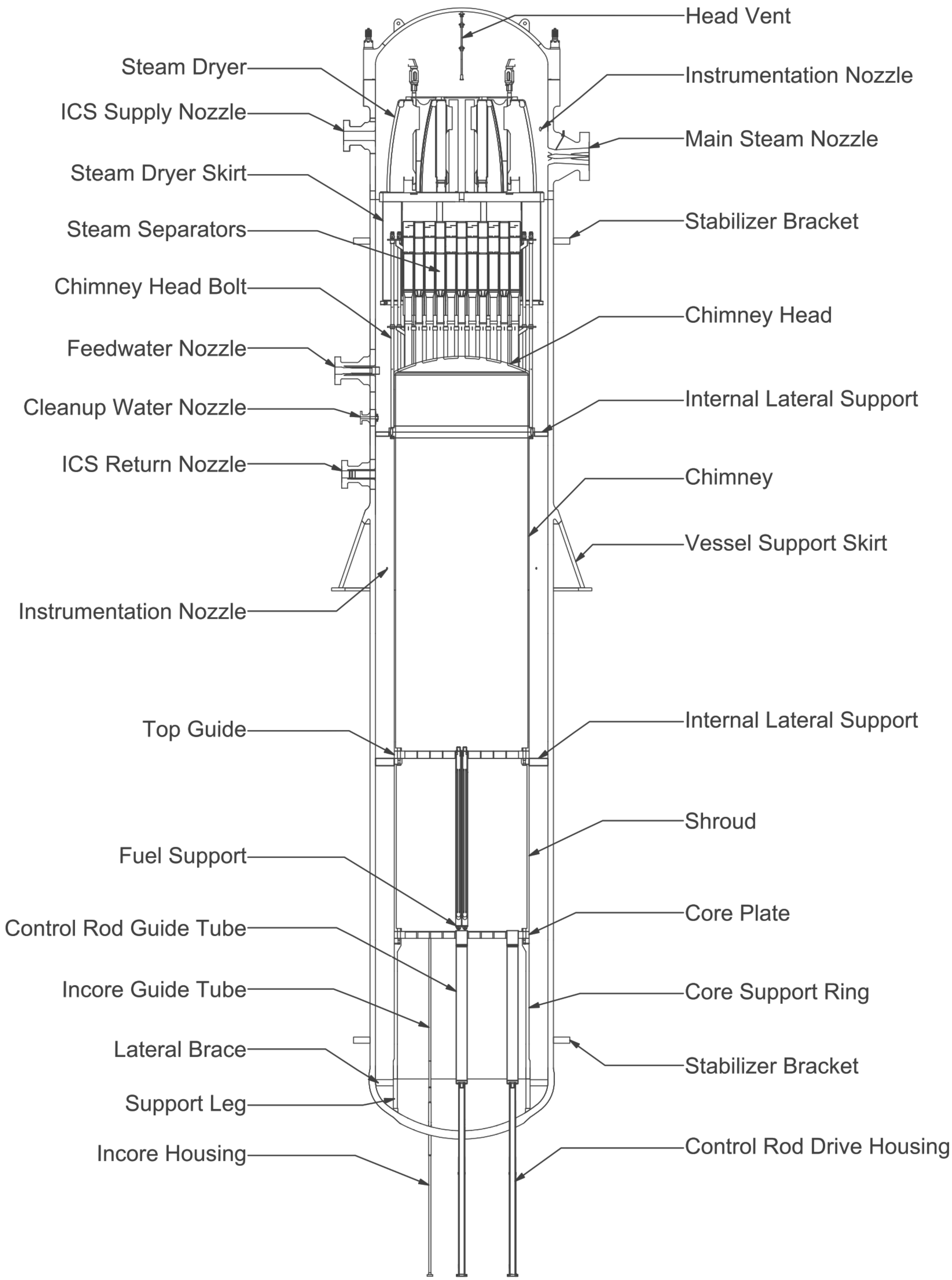


Figure 4.5-7: Overview of BWRX-300 Core Internals

The design basis requirements of the BWRX-300 reactor core support the safety design philosophy for the mitigation of Loss-of-Coolant-Accidents using inherent design features. These design features include:

- Large water inventory contained within the core to eliminate system challenges
- Reduced number and size of RPV nozzles
- All RPV nozzles located significantly above the level of active fuel
- Flanged RPV isolation valves close to the RPV

The core components consist of the fuel assemblies, control rods and the nuclear instrumentation. The control rods provide negative reactivity into the core to allow for the control of reactor power. The nuclear instrumentation in the core consists of the Power Range Neutron Monitoring System (PRNM), GTs and the Wide-Range Neutron Monitoring System (WRNM). The PRNM provides neutron monitoring power signals to the Reactor Trip system and provides signals for post-accident monitoring purposes.

Details about the PRNMs, GTs and WRNMs can be found in PSAR Chapter 5, Section 5.5.9 and Chapter 7 [R-6].

Physical and chemical properties of the reactor internal components

The alloys used in the BWRX-300 reactor (other than the RPV and steel components) are predominantly alloys with lower strength and high ductility. These alloys are not susceptible to rapid fracture and there is an extensive history of OPEX of BWR components fabricated from austenitic stainless steel and nickel-based alloys to support their selection for the BWRX-300. The only components that show loss of ductility and reduced fracture resistance are the core components, particularly the core shroud, which is subjected to irradiation. Irradiation of the core shroud results in material changes effectively increasing the strength of the material while reducing its ductility. The shroud design takes this into account to assure component robustness.

Surveillance and inspection programs for reactor internals

In the design of the RPV, consideration has been given to the effects of irradiation on beltline fracture toughness through the control of the composition of vessel beltline materials. Surveillance test program specimens are used to monitor the radiation-induced changes in the mechanical properties of the core beltline region materials of the RPV. The surveillance specimens are prepared from the actual materials used to fabricate the beltline of the RPV. Part of the specimens are in removable specimen capsules at the inside vessel wall opposite to the active core, which can be removed for analysis and comparison to the remaining baseline unirradiated specimens to monitor potential degradations to mechanical properties of the RPV.

The reactor core components are susceptible to aging degradation mechanisms, particularly Stress Corrosion Cracking (SCC) and specifically for the core shroud, Irradiation-Assisted Stress Corrosion Cracking (IASCC). The design of the components at operating BWRs incorporates ways to minimize degradation through material selection, component fabrication, and reactor chemistry control. In addition, inspection programs have been implemented to monitor key internal components for the impact of environmental degradation in currently operating BWRs.

The inspection programs include inspection and flaw evaluation approaches that provide assurance of the long-term integrity and safe operation of the reactor internal components, and provide information on component description and function, susceptible locations, and safety consequences of failure. The programs include methods, extent, and frequency of inspection. The programs are focused on managing the effects of stress corrosion cracking, intergranular stress corrosion cracking, or irradiation-assisted stress corrosion cracking, cracking due to fatigue, and loss of toughness due to neutron and thermal embrittlement. The BWRVIP program includes components such as the core shroud, core plate, core shroud support, top guide, core plate, steam dryer, and CRD housings.

These existing programs are used as the initial basis to develop BWRX-300 surveillance inspection plans for the reactor internal components.

Reactor Core Monitoring

The DNNP BWRX-300 will utilize the ACUMEN core monitoring and prediction system. ACUMEN monitors and predicts fuel performance based on neutron instrumentation inputs, which provides reactor engineers and operators the data needed to optimize the performance of the core. Further information regarding the

DNNP BWRX-300's core monitoring system can be found in PSAR Chapter 4, Section 4.5.3 [R-6].

Nuclear Design and Core Nuclear Performance

The nuclear design and core nuclear performance of the DNNP BWRX-300 is summarized in PSAR Chapters 4, 9A.1 & 16. Specifically, Chapter 4, Section 4.3 provides the bases and description for the core nuclear design, summarizing information such as the nuclear design of the fuel, reactivity coefficients, fuel enrichment distributions, burnable poison distributions and other parameters important to the design of the BWRX-300 nuclear core.

Core Thermal-Hydraulic Design

The thermal-hydraulic design of the reactor core ensures that adequate provision is made for heat transfer and heat removal based on the heat generation and distribution in the reactor core. In the BWRX-300 heat is removed from the reactor core by the light water reactor coolant during normal power operations via the NBS. Normal residual heat removal when the reactor is shutdown occurs via the SDC System or passively via the ICS, as required.

The thermal-hydraulic design of the core establishes the thermal-hydraulic safety limits to ensure that fuel design limits are not exceeded during any condition of normal operation, including during AOOs in accordance with REGDOC-2.5.2 [R-21].

In addition, conditions within the reactor are maintained such that fuel power density is kept, with sufficient margin, below levels that could cause overstressing the fuel cladding because of fuel pellet-cladding differential expansion. This is done by ensuring that the linear heat generation rate (LHGR) is maintained below a conservative maximum value during normal operation and AOOs. See PSAR Chapter 4, Section 4.4 [R-6] for further information.

Thermal-Hydraulic Design Basis, Methods and Tools

The BWRX-300 thermal-hydraulic design basis, tools, and methods used for its evaluation are standard codes that are currently in use in other BWR designs.

Details on the thermal hydraulic design basis, analysis methods and tools are provided in PSAR section 4.4 while information related to codes and qualification can be found in PSAR Chapter 15, Section 15.5.1.

Typical BWRX-300 Thermal-Hydraulic Design Characteristics

PSAR Chapter 4 Section 4.4.2.1.6 provides the typical Thermal-Hydraulic characteristics for the BWRX-300.

Core Stability

The BWRX-300 is designed so that coupled neutronic and thermal-hydraulic power oscillations are not possible throughout the whole operating region (that is, core reactivity remains stable). The BWRX-300 is designed to protect acceptable fuel design limits during AOOs. As a backup, the BWRX-300 implements a system to detect and suppress instabilities, should they occur. Justification for the thermal-hydraulic stability of the core have been summarized below and further details can be found described in Chapter 4, Section 4.7 of the PSAR [R-6].

The BWRX-300 has design features which ensure stable behaviour during normal operation and minimize the impact of potential oscillations during normal and off-normal conditions. These design features include:

- Small core
  The small core size and higher inlet orifice pressure drop of the BWRX-300 reduces the likelihood of regional mode instabilities.
  - Tighter neutronic coupling removes regional mode of oscillation
  - Core-wide oscillations are the dominant mode
- Natural circulation
  - No recirculation pump trips that result in significant change from stable to unstable conditions
  - Loss of Feedwater AOO eliminated as a stability concern
- Tall chimney
  - Increases volume of water
  - Increases driving head and natural circulation flow
  - Dampens oscillations more quickly
- Large downcomer area
  - Reduces core flow restrictions
- Tight core inlet orifices
  - Improves 2-phase to single phase loss ratio
- Feedwater temperature requirement
  - Neither thermal margins nor decay ratios are compromised

    - Minimized normal operation inlet subcooling
- Less subcooling

The BWRX-300 uses the same acceptance criteria for core stability as implemented for the current GEH feet of BWRs. These acceptance criteria consider uncertainties arising from both operating state and modeling. The BWRX-300 meets the established design acceptance criteria for core stability. See PSAR Chapter 4, Section 4.8.5 for further information.

Reactivity Control System

The following section describes the reactivity control systems for the BWRX-300 design, which consists of the Control Rods and the FMCRD system, feedwater level control system, gadolinia fuel rods, as well as the BIS. Further information regarding these systems can be found in PSAR Chapter 4, Section 4.2 and 4.5 [R-6] and Section 4.5.9 of this Application document.

Design Requirements and Description

The CRD System is the primary mechanism by which reactivity is controlled within the BWRX-300. It comprises of three major elements: the Fine Motion Control Rod Drive (FMCRD) mechanisms, HCUs assemblies, and the control rod drive hydraulic (CRDH) subsystems. The CRDs are used for power shaping, power level adjustments, and insertion of negative reactivity to achieve shutdown. CRDs are the primary means of achieving shutdown in normal operations, AOOs, DBAs, and DEC scenarios.

The CRDs provide diverse sources of control rod motive force and diverse sets of control and actuation logic to provide high confidence that the control rods can be inserted into the reactor core when necessary. The control rods are used for power shaping, power level adjustments and insertion of negative reactivity to achieve shutdown.

Further details and summarized descriptions of the CRD and FMCRD design can be found in Section 4.5.9 of this Application, and PSAR Chapter 4, Section 4.5 [R-6].

The BIS is a complementary design feature that provides a separate and diverse means for manually inserting negative reactivity (enriched boron-10 solution) into the reactor core. The BIS is not a back-up to the automatic shutdown function of the CRD. Details for this system and its basis can be found in PSAR Chapter 4, Section 4.5 and Chapter 15 Section 15.1.

Qualification and commissioning tests

The design of the CRD system has significant available OPEX due to its wide-spread use in operating BWRs since 1961. The design has not changed significantly over the past few decades and has proven to be a highly reliable and effective means of reactivity control in BWRs. The well understood fleet operational data has been utilized in the methods to design, evaluate, and analyze the control rods in their role as the primary means of reactivity control for the BWRX-300. Further information regarding the qualification and commissioning tests of the CRD system, as well as how their performance is credited in support of the Safety Analysis described in PSAR Chapter 15, can be found in PSAR Chapter 4, Section 4.5.

Diversity and physical separation in design

The BWRX-300's reactivity control systems achieve diversity and physical separation through a variety of designed features within each of the systems.

Examples of the BWRX-300's reactivity control diversity include the FMCRDs, which utilize two diverse motive forces (motor & hydraulic) with their own diverse sets of control and actuation logic for the insertion of control blades.

Further information regarding the diversity and physical separation of the reactivity control system designs, including the rate of reactivity insertion and the depth of each reactivity control system, is provided in PSAR Chapter 4, Section 4.5 [R-6].

Reactor materials

The reactor materials being used in the construction of the BWRX-300 are described in detail in PSAR Chapters 4 and 5.

Materials used in the DNNP BWRX-300 are specifically selected to be capable of maintaining reliable operation of plant systems and equipment during the design life of each component. For the reactor internals, austenitic stainless steel will be primarily used to minimize levels of radiation from corrosion products generated in-core. Where higher strength requirements than those for standard austenitic stainless steels are needed, nickel base alloys, nitrogen strengthened austenitic alloys, or precipitation hardening stainless steels will be used. Materials selection with composition controls and defined fabrication processes is used to address materials degradation issues in the reactor system; specifically stress corrosion cracking and irradiation stress corrosion cracking, see PSAR Chapter 5, Section 5.2 for more details.

The primary materials used in the BWRX-300 RPV and Pressure Boundary Components are listed in Table 5.2-1 of PSAR Chapter 5.

Basis of Material Selection and Component Fabrication

The BWRX-300 fabrication and material selection are based on the long history of material experience, refinement of material alloy chemistry and well-established fabrication processes to assure the best characteristics in service to preclude long term material degradation. These materials include wrought and cast alloys as well as weld metals used in the construction. Materials selection with composition controls and defined fabrication processes are used to address materials degradation issues in the reactor system, specifically SCC and IASCC. The processing activities used in fabrication of the RPV and reactor components, can be found in PSAR Chapter 5, Section 5.2 [R-6].

The BWRX-300 will employ proven BWR materials and processes which have been refined to meet reactor requirements.

These non-metallic engineered materials used in association with reactor system components are controlled to minimize or eliminate potential for detrimental effects on metallic reactor components. These materials which include gaskets, packing, bushings, etc., that are installed within the reactor system such that they are in contact with reactor water, will conform to the applicable chemistry controls specification. See PSAR Chapter 5, Section 5.2 and Chapter 3, Section 3.9 for more details.

PSAR Sections 5.2.3 and 5.2.5 discuss low alloy steel and austenitic high strength materials that are used as fasteners in the BWRX-300.

Core Structural Component Materials

It is important that the materials to be used in the BWRX-300 will be capable of maintaining reliable operation of plant systems and equipment during the design life of each component.

Selection of the materials is directed toward minimizing their susceptibility to degradation mechanisms. Design guidelines and controlled fabrication processes are used to manage the component stresses, with emphasis on surface stresses.

See PSAR Chapter 5 for more detailed information on materials used for the typical internal components including the shroud, chimney, steam dryer, top guide, core support, CRD housings and instrumentation housings.

Surveillance Program

Surveillance programs are established to address several aspects of the materials used in monitoring of the BWRX-300 including:

- the integrity of the pressure retaining RPV and its attachments;
- the time limiting aging impact of the RPV;
- the integrity of the reactor internals;
- the piping components for degradation;
- the water chemistry program for its mitigation effectiveness; and,
- Flow Assisted Corrosion (FAC).

See PSAR Chapter 5 Section 5.11 for more details [R-6].

Design of the reactor coolant system and reactor auxiliary system

The BWRX-300 NBS and associated cooling systems are primarily responsible for the removal of heat from the reactor core and generation of steam for the high- and low-pressure turbines.

The NBS is comprised of three primary subsystems: RPV, MS, and RPV instrumentation. In addition to these subsystems, the NBS also provides RPV head venting and RPV head flange seal leak detection features. Within the NBS are the redundant with diverse instrument and electrical features, safety class reactor isolation valves interfacing with the subsystem and systems listed below:

- Main Steam (MS)
- RPV head vent
- ICS steam supply
- ICS condensate return
- Condensate and Feedwater Heating System (CFS)
- Reactor Water Cleanup (RWCU)

The design of OPG's BWRX-300 NBS is simplified when compared with other operating BWR reactor coolant systems, as it eliminates the use of recirculation pumps and associated piping. Circulation of the reactor coolant through the NBS is accomplished via natural circulation, which is enabled mainly by the addition of a high chimney between the top of the core to the bottom of the steam separator shroud head.

Isolation of the main steam lines in the NBS is achieved using two Main Steam Reactor Isolation Valves (MSRIVs) and one Main Steam Containment Isolation Valve (MSCIV) on each main steam line. The main function of the MSRIVs is to close and isolate the RCPB at the RPV when a pipe break occurs either inside or outside of containment. If a main steam line break occurs inside containment, closure of the MSCIV outside of containment results in the sealing of containment.

The RPV also has a head vent subsystem that during normal operations vents non-condensable gases in the steam dome area to one of the main steam lines. During shutdown, air can be released from the RPV into a quench tank in containment, when the RPV is filled with water for leak testing. Due to the nature of the design being a BWR, there is no need for a separate steam generator/heat exchanger or a pressurizer for maintaining reactor coolant pressure control as steam and coolant pressure is maintained within the RPV.

Under normal operation, the normal heat sink (NHS) for the NBS is the main condenser. However, the RWCU, SDC and the ICS can be used to cool the RCS under a variety of situations. During normal shutdown and reactor servicing, the RWCU and SDC removes residual and decay heat. The RWCU and SDC (in conjunction with the ICS) allows decay heat to be removed whenever the main condenser is not available (e.g., during hot standby conditions). The RWCU and SDC recirculate a portion of the reactor coolant through a demineralizer to remove dissolved impurities with their associated corrosion and fission products from the reactor coolant, as well as removing excess coolant from the reactor system under controlled conditions. Further details regarding the specifics of the RWCU and SDC systems can be found in PSAR Chapter 9A, Section 9A.2.

The ICS provides cooling of the reactor if the RCPB becomes isolated following a scram during power operations. The ICS automatically removes residual sensible and core decay heat to limit reactor pressure when reactor isolation occurs. Over a longer duration, the ICS provides a way to remove excess heat from the reactor, if the normal heat removal path is unavailable. The ICS consists of three independent condenser trains. Only one of the three ICS trains is required to mitigate AOOs and two of three are required to mitigate LOCAs. The ICS is the Emergency Core Cooling System (ECCS) which removes sufficient heat to limit the pressure increase below the design pressure of the RPV and is the primary means of depressurizing and providing overpressure protection of the NBS. Further details regarding the ICS can be found in PSAR Chapter 6, Section 6.2 and Section 4.5.9 of this Application document.

Details regarding the NBS Component Supports and Restraints can be found in PSAR Chapter 5, Section 5.9.

The preservice and in-service inspection and system pressure test programs for the NBS are described in PSAR Chapter 5, Section 5.5.

Integrity of the reactor coolant system pressure and fluid boundary

PSAR Chapter 3, Section 3.6 General Design Aspects for Mechanical Systems and Components provides the framework for which the detailed analytical and numerical stress evaluations, as well as the engineering mechanics and fracture mechanics studies for all components comprising the reactor coolant system pressure or fluid boundary are completed.

4.5.9 Safety systems and safety support systems

Means of shutdown

The BWRX-300 design provides independent and diverse means of quickly reducing the nuclear reactivity within the reactor core to subcritical levels from all normal and postulated AOOs and DBAs with adequate margins as described in the PSAR Chapter 4, Section 4.3 [R-6].

The BWRX-300 design provides means of shutdown using a CRD system.

Control Rod Drive System

The CRD system provides the primary means of achieving shutdown in the BWRX-300 design.

The BWRX-300 reactor core design uses 57 control rods that are inserted from the bottom of the RPV. The control rods are cruciform shaped, as shown in Figure 4.5-8, with absorber blades used to control reactivity, travelling vertically in the space between 4 adjacent fuel assemblies.

The BWRX-300 CRD system uses two diverse motive forces for positioning the control rods. The CRD system achieves this with three major elements: Fine Motion Control Rod Drive (FMCRD) mechanisms; HCU assemblies; and the CRDH subsystem.

The CRD system design is based on the designs used in the currently operational BWR fleet. The methods to design, evaluate and analyze the control rods, in their role as the primary means of reactivity control, are well established. They have been exercised repeatedly, improved over time, and remain in use today.

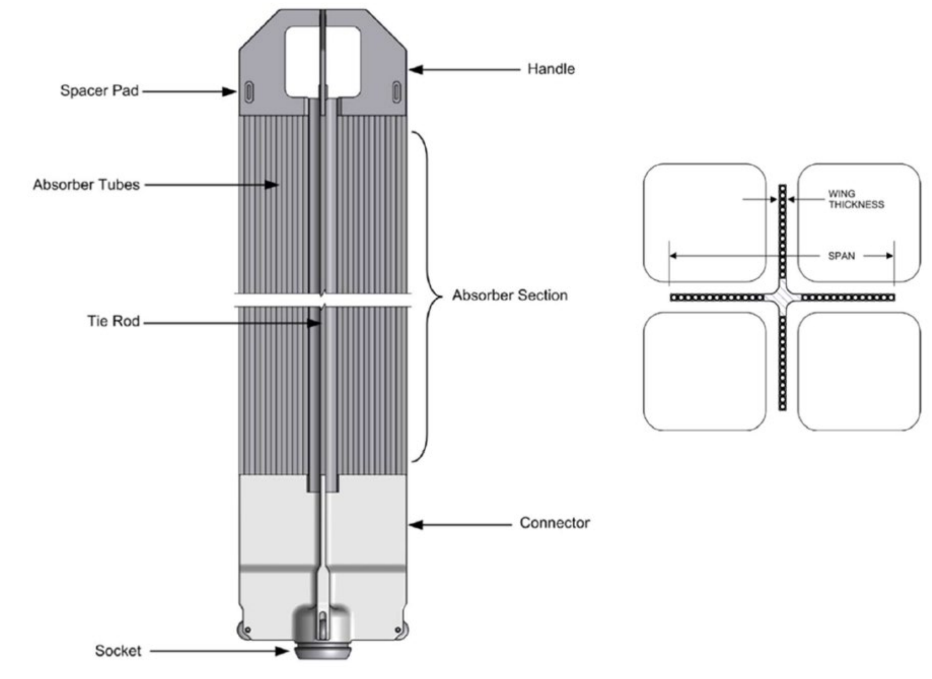


Figure 4.5-8: BWRX-300 Control Rod and Core Cell Arrangement

Control Rod Drive Hydraulic Scram

In BWR designs, a fast insertion of control rods using stored hydraulic energy is referred to as a scram or hydraulic scram shown in Figure 4.5-9.

The BWRX-300 design uses a fast shutdown consisting of hydraulic scram of the control rods by the CRD system. The system uses stored energy in redundant accumulators, actuated by multiple diverse control systems with redundant channels and sensors within each control system. The CRD hydraulic scram meets the AOO and DBA acceptance criteria with adequate margin. This is described in the PSAR Chapter 4, Section 4.5 [R-6].

The force required for hydraulic scram is provided by 29 HCUs that include nitrogen charged accumulators. A hydraulic scram is initiated by opening 29 scram valves, one on each accumulator water discharge path.

The CRDH subsystem uses high-pressure demineralized water, which is regulated and distributed to provide charging of the HCU scram accumulators. The stored energy is used in actuation of hydraulic scram to drive all control rods into the core, as discussed in PSAR Chapter 4, Section 4.5. The effectiveness of the means of shutdown, in terms of speed of action and shutdown margin, will be demonstrated in layered deterministic analyses as described in PSAR Chapter 15, Section 15.5 and applied as OLCs.

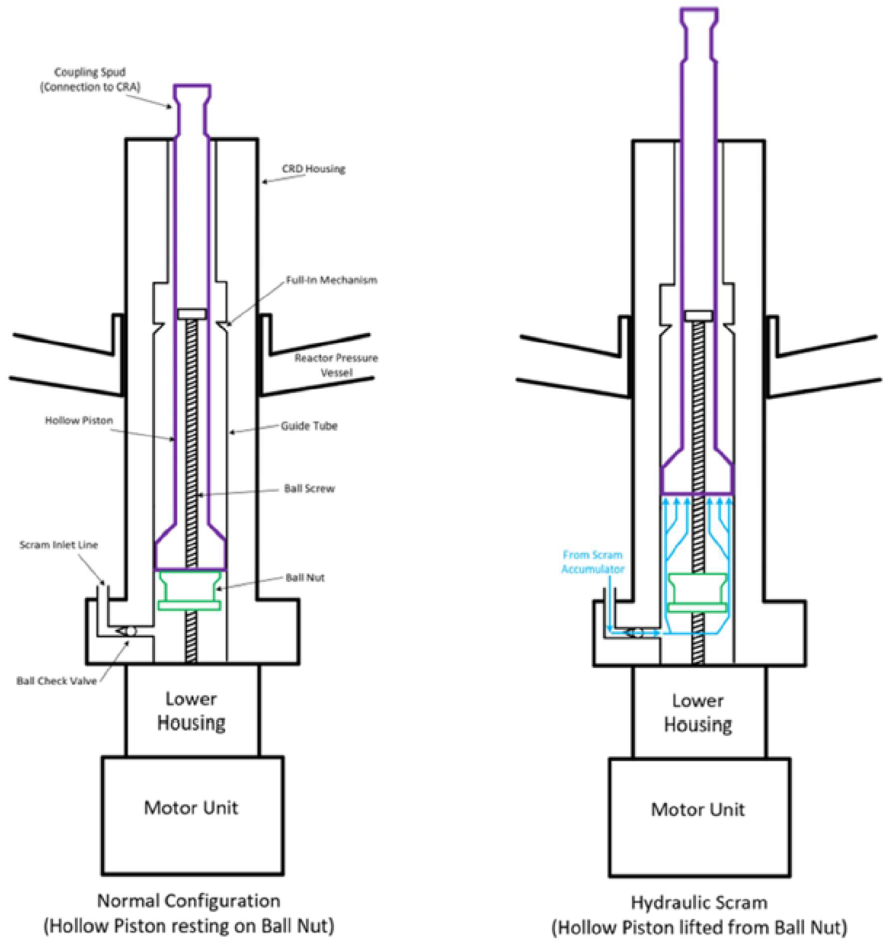


Figure 4.5-9: Hydraulic SCRAM

Fine Motor Control Rod Drive System

The BWRX-300 provides a second means of shutdown consisting of electrical motor insertion of control rods by the CRD system. This system uses stored energy in redundant batteries, actuated by multiple diverse control systems with redundant channels and sensors within each control system, which meets the DBA acceptance criteria with adequate margin. This is described in the PSAR Chapter 4, Section 4.5 [R-6]. The FMCRDs (see cross-section in Figure 4.5-10) are designed to provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) in response to manual or automatic signals from the Reactor Protection System. The FMCRDs also provide electric-motor-driven fast run-in of all control rods as a path to rod insertion that is diverse from the hydraulic-powered scram. This design addresses the low probability situation of the failure of the hydraulic scram function. Diverse sensors, logic and actuation of the shutdown means are provided in the design. In the event of a PIE that requires a rapid reactor shutdown, and if the reactor scram fails or is delayed, then the reactor is shut down by the electric motor fast run-in of FMCRDs function.

Besides the normal off site derived electrical supply, each FMCRD motor can receive power from standby diesel generators and backup batteries, which makes a situation in which control rods cannot be inserted due to lack of electrical supply very unlikely.

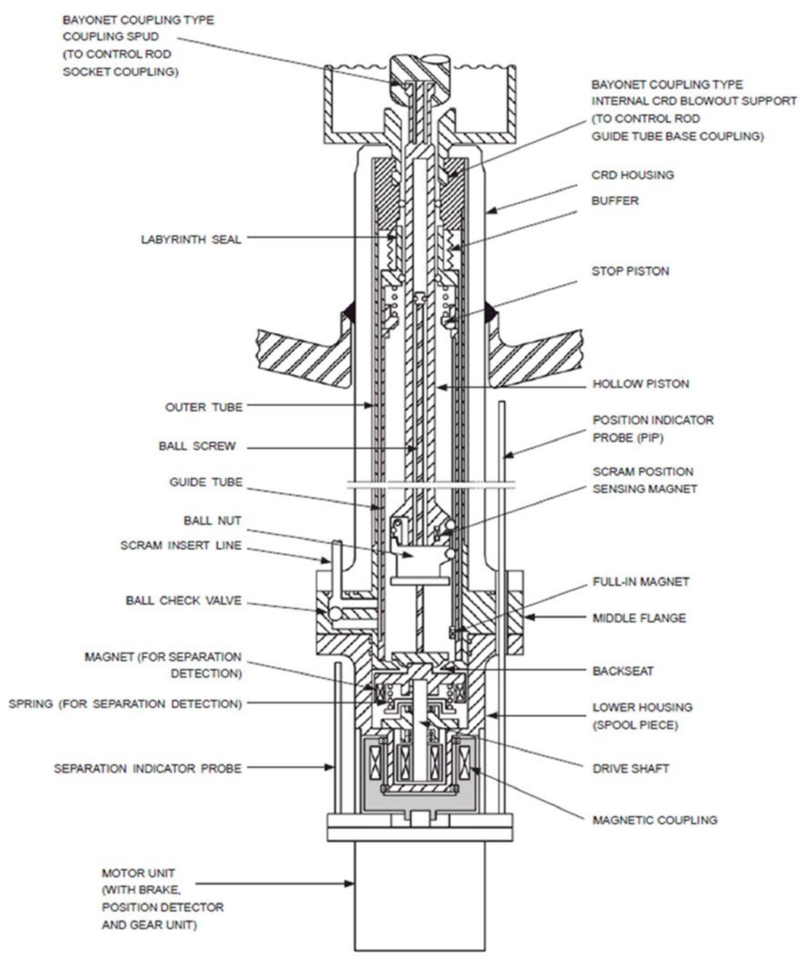


Figure 4.5-10: Fine Motor Control Rod Drive

The BWRX-300 has design and mitigating features for ensuring that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded for rod withdrawal error events. The BWRX-300 rod control system employs redundancy to limit the effect of single failures. If a malfunction of the rod control system during operation results in a rod withdrawal error, nuclear instrumentation is used to generate a rod block or reactor scram.

Means of Shutdown Actuation and Defence Lines

Three diverse actuation systems are provided: Reactor Trip System (RTS), DPS and Anticipatory Trip System (ATS). The summary descriptions of RTS, DPS and ATS are outlined in PSAR Chapter 7, Section 7.3 [R-6].

The extent of trip coverage will be documented through the layered deterministic analysis approach as per Chapter 15 of the PSAR. Potential failure modes of trip parameters are documented in failure mode and effects analyses, as part of the normal course of design.

When an automatic shutdown actuation function is initiated, it proceeds to completion (e.g., control rods reach their full-in position) without the possibility for an operator to stop it, and the actuation signal cannot be reset until the initiating condition has cleared.

The means for manual initiation of shutdown are provided in the MCR and at a secondary control location, along with the display of variables to monitor shutdown status.

Systems and Components Supporting Emergency Core Cooling

The BWRX-300 design accomplishes the Emergency Core Cooling (ECC) function through the ICS. The BWRX-300 does not require coolant injection to handle pipe breaks and transients. Large pipe breaks will be isolated by the RPV isolation valve before there is a significant loss of coolant. Small pipe breaks are sized with a limited flow area which reduces the loss of coolant inventory. The ICS pools have enough inventory to provide adequate decay heat removal such that injection is not required for at least 72 hours.

Inherent Safety Features Supporting ECC

The BWRX-300 leverages the engineering design for inherent safety features to support mitigation of LOCA. Two of these features include the inventory of water in the RPV and the location of the RPV nozzles significantly above the Top of Active Fuel (TAF).

RPV water inventory

The BWRX-300 design utilizes the RPV volume, along with the chimney region to provide a reservoir of water above the core. The normal water level is approximately 12.5 m above the TAF.

The RPV inventory ensures the core remains covered following transients involving feedwater flow interruptions or LOCAs. The RPV is also equipped with RPV isolation valves attached directly to the reactor vessel. These design features preserve reactor coolant inventory ensuring adequate core cooling is maintained following a LOCA.

The RPV volume also reduces the rate at which reactor pressurization occurs if the reactor is suddenly isolated from its NHS.

Reduction in number and size of RPV nozzles and pipe lengths relative to predecessor design

The BWRX-300 larger-bore pipes are subject to rigorous design, quality, material, fabrication, inspection, testing, and construction requirements, as well as OPG's ongoing quality, inspection, testing, and monitoring programs and processes throughout the plant lifecycle. These processes and the reduction in pipe size (length and diameter) relative to predecessor designs, provide a high degree of assurance of the low probability of line breaks that could result in a LOCA.

The BWRX-300 design places RPV nozzles as high on the RPV as possible to preclude the effect of a potential pipe break at or below the elevation of the core. The lowest fluid system nozzles are located approximately four meters above the TAF.

Isolation Condenser System

The BWRX-300 ICS, together with the stored RPV water inventory removes heat from reactor assembly and core decay heat from the reactor fuel, in a passive way, when the normal heat removal system is unavailable. Reactor heat is transferred from each IC heat exchanger tubes to the surrounding IC pool water by condensation and natural circulation. Each IC has its own large, separate pool positioned immediately above (and outside) the containment. The ICS is shown on Figure 4.5-11.

No forced circulation equipment or coolant injection is required as the ICS provides sufficient water inventory for reactor decay heat removal for at least 72 hours, even when assuming one ICS train is unavailable.

The IC units located in the RB are submerged in an IC pool as shown in Figure 4.5-11. The ICs condense steam on the tube side and heat is transferred to the IC pool water which boils, and steam is vented to the atmosphere.

The steam side connection between the RPV and the IC is normally open, and the condensate return line is normally closed. This allows the IC and condensate return piping to fill with condensate that is maintained at a subcooled temperature by the ICS pool water during normal reactor operation.

The ICS is initiated automatically on either a high RPV pressure indicating an overpressure event, or on signals indicating a LOCA (low RPV level or high containment pressure). The ICS can also be initiated manually by the operator from the MCR by opening the ICS condensate return valve.

To place an ICS train in service, one of the two condensate return valves must open. The two valves installed in parallel ensure that no single active failure can cause the loss of any single train. The parallel valves fail to the fully open position on a loss of pneumatics, control power, or control signal. Once the valves fully open, they stay fully opened until they are reclosed intentionally by the operator.

When a condensate return valve is opened, the subcooled condensate that is stored in the system during the standby state enters the RPV chimney interior providing additional inventory while quenching steam and lowering pressure at the reactor core exit. Simultaneously, steam from the RPV enters the IC where it is condensed in the tubes and returned to the RPV in a continuous cycle.

There are three independent ICS trains available for RPV heat removal. The limiting factor for determining heat removal time is ICS pool water inventory. Decay heat removal can be continued, if needed, by replenishing the IC pool inventory. The IC pools are located at ground level and are not pressurized, so replenishment can be accomplished using readily available sources.

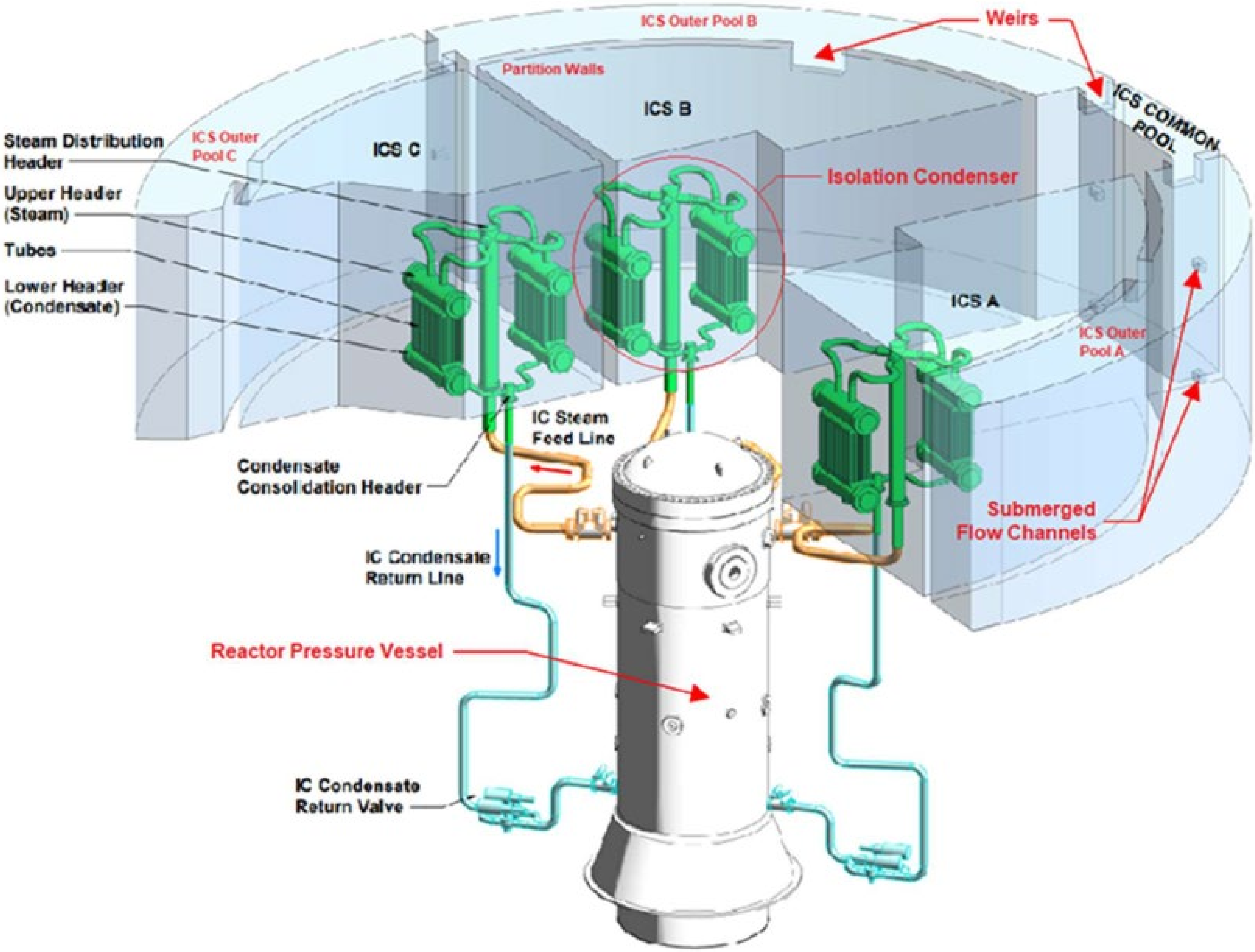


Figure 4.5-11: Isolation Condenser System

Systems and Components Supporting Emergency Heat Removal

BWRX-300 emergency heat removal is performed passively by the ICS, which is designed as a Safety Class 1 system. See PSAR Chapter 3, Section 3.2 [R-6] for further information. The ICS provides emergency heat removal for both LOCA and non-LOCA accident events.

Normal reactor cooling following a shutdown or scram is achieved by the main condenser via the Turbine Bypass Valve (TBV) with long term heat removal and controlled depressurization achieved by the Shut Down Cooling (SDC) system. The ICS represents the ultimate heat sink for protecting the reactor core for any emergency or off-normal event where the main condenser is not available, and the RPV is isolated.

When the RPV water level and pressure control systems are not available, or during certain PIEs and DECs, actuation of the ICS system provides fuel cooling and long-term decay heat removal.

Interfacing Requirements

As the ICS is a passive system, correct operation following an accident is not dependent on the electrical grid or any off-site power source. The fail-safe design of the system ensures that a loss of power event will not prevent placing the system in service.

For the ICS, normal operation is defined as the system in a standby state ready to be placed in service and fully operable. Off-normal operations for ICS are defined as the system in operation which is initiated by opening of one or both condensate-return isolation valves. The ICS can be intentionally and manually placed into service for "routine" shutdown cooling after a normal reactor shutdown.

The ICs are normally in standby mode. Inadvertent Isolation Condenser Initiation (IICS) resulting in the introduction of cold water into the reactor has been considered in the safety analysis in PSAR Chapter 15 [R-6]. IICS PIEs are considered reactor coolant inventory increase events and the analysis has demonstrated that there is no impact to plant safety.

Isolation Condenser Cooling and Clean-up System

The Isolation Condenser System Pool Cooling and Clean-up (ICC) System is designed to remove heat from the ICS (PSAR Chapter 6, Subsection 6.2) pools such that the bulk temperature of water in the pools is maintained below Technical Specifications limits and thereby ensure the readiness of the ICS to perform its Safety-Category function.

Secondary functions of the ICC include maintaining the cleanliness of the ICS pool water and providing the capability to add clean makeup water during normal reactor operations to offset the minor routine loss of water inventory due to evaporation. For further details see PSAR Chapter 9A, Section 9A.2.6 [R-6].

Systems and Components Supporting Confinement and Containment Functions

Systems and components supporting confinement and containment functions, including functional requirements, mechanical features and leakage testing requirements, are addressed in PSAR Chapter 6, Section 6.3. Figure 4.5-12 below shows the general containment arrangement.

The BWRX-300 containment pressure boundary provides a Safety Class 1 (SC1) leak-tight barrier to prevent the release of radioactive material to the environment in the event of a failure of the RCPB. The containment structure maintains its functional integrity during and following peak transient pressures and temperatures that would occur following postulated DBAs. The containment integrity is assured by the isolation of mechanical systems penetrating containment for PIEs that could result in the uncontrolled release of radioactive materials to the environment. Other containment systems support the passive removal of heat generated from AOOs, DBAs, and DECs, and the control of hydrogen and other combustible gases from SAs or BDBAs.

The BWRX-300 uses a traditional Primary Containment System (PCS) completely enclosed within the RB for the ultimate containment of radioactive materials for various postulated events. The RB protects the containment structure from external hazards (e.g. wind loads, tornado loads, aircraft hazard, missiles) and external beyond design basis scenarios.

The PCS specifically provides the FSF of confinement that achieves the following confinement functions:

- Control of pressure and temperature
- Isolation of the confinement boundary
- Leak-tightness of the confinement boundary
- A controlled point of release (which is usually elevated)
- Control of combustible sources
- Reduction of the concentration of free radioactive material in the confinement boundary
- Protection against external events
- Radiation shielding

The PCS is supported by a PCV that is an SCCV structure to maintain its structural integrity and allows the PCV to perform the pressure retaining and leak-tightness functionality during and following postulated DBAs.

The PCS, its boundary and interfaces are illustrated in PSAR Chapter 6, Section 6.3 [R-6]. The PCS is comprised of the following components, see details in the corresponding sections of PSAR:

- Steel-Plate Composite Containment Vessel – Section 6.3.2.1
- Containment Closure Head – Section 6.3.2.1
- Personnel and Equipment Airlocks – Section 6.3.2.1
- Mechanical Penetrations – Section 6.3.2.1
- Electrical Penetrations – Section 6.3.2.1
- Refueling Bellows Seal – Section 6.3.2.1
- Integrated Leak Rate Penetration – Section 6.3.2.1
- Containment Corium Shield / Core Catcher Equipment – Section 6.3
- Core Region Biological Shield / Bioshield - Section 6.3
- Passive Containment Cooling System – Section 6.3
- Containment Support Structures – Section 9B2.2
- Maintenance Structures – Section 6.3.2.1
- RPV Vertical and Lateral Support Structures – Section 6.3.2.1

The PCV design limits are established and confirmed by analysis to envelope postulated DBA scenarios. For severe accident or BDBAs, containment is provided with a pressure relief vent pathway that is routed to mitigate the release dose consequences that includes a provision for capture and retention of radioactive particulate and gases to the extent practicable and to protect from hydrogen gas detonation (see PSAR Chapter 6, Section 6.3.5).

The leak-tight boundary is a primary safety class function of the BWRX-300 PCS that prevents radioactive contamination released from the NBS and connected systems inside containment.

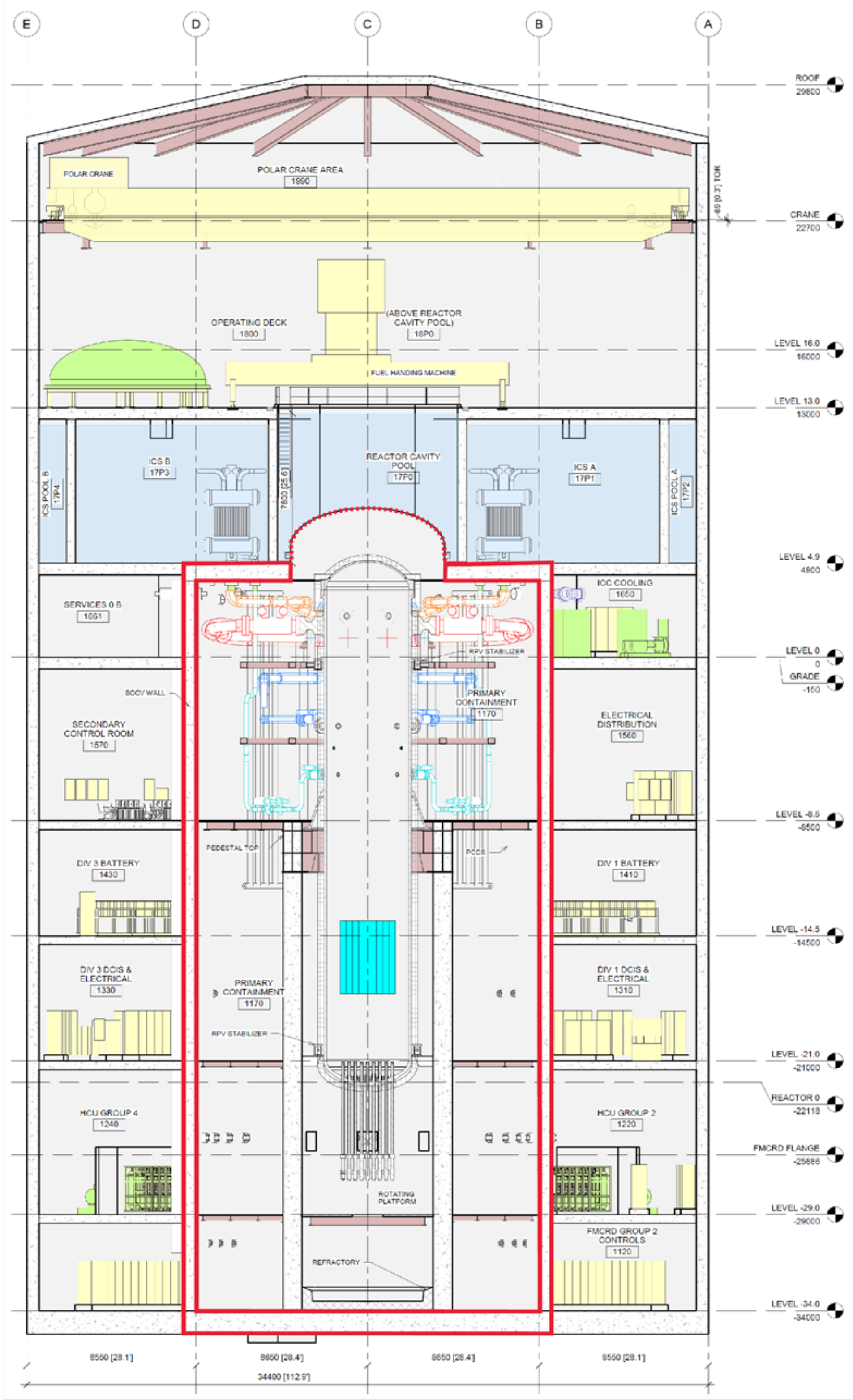


Figure 4.5-12: PSAR Figure 6.4.2-1: General Containment Arrangement – Reactor Building Section View

The PCV or more specifically, SCCV is designed such that the periodic Integrated Leak Rate Testing (ILRT) can be conducted as required. The preoperational and periodic integrated leak rate tests of the SCCV (and major hatches/penetrations) are to demonstrate leak tightness integrity of the containment boundary. The flange seals of the containment closure head and metal containment components that have potential for significant contribution to leakage are designed to be individually testable. Where resilient seals such as elastomeric seals are used, they have the capability for performing leak testing at the containment design pressure in accordance with Section 8.6.3 of REGDOC-2.5.2. Additional information can be found in PSAR Chapter 6, Section 6.3 [R-6].

The Containment Inerting System (CIS) establishes the dry pressure-containing atmosphere within containment. The inerted atmosphere is maintained in all operating modes except shutdown for refueling, maintenance and low power inspection. CIS provides dilution of hydrogen and oxygen gases released in a post-accident condition by radiolytic decomposition of water and the released hydrogen from water and fuel cladding (zirconium) reaction during a severe accident condition. The dilution precludes the development of a combustible atmosphere by maintaining an oxygen deficient atmosphere inside PCV as well as minimizing long-term corrosion and degradation of the PCV inner steel faceplates and other contained components.

The CIS provides containment with the following ventilation flow paths:

- Nitrogen supply
- Containment exhaust
- Containment overpressure venting

The Containment Supply Flow path provides all start-up and makeup inerting with one line that runs from the liquid nitrogen storage tank to a penetration in lower containment. The containment supply flow path consists of a pressurized liquid nitrogen storage tank, a vaporizer, an inerting line, and associated valves, controls and instrumentation. Additional information on CIS is captured in PSAR Chapter 9A, Section 9A.4.

The containment venting process is carried out by an exhaust line that will be opened to provide a passageway for containment gases during inerting and de-inerting. All valves in this subsystem are air operated. All components are in the RB.

Containment over-pressurization is mitigated with a pressure relief line that connects to the cavity pool. For more details refer to PSAR Chapter 9A, Section 9A.4.

Heat Removal from containment is provided by either one of the following systems depending on the operational state.

- Containment Cooling System (CCS)
- Passive Containment Cooling System (PCCS)

The CCS provides a closed loop active cooling system that recirculates air/nitrogen with no outside air introduced into the system except during outages. The system includes Air Handling Units (AHUs) that interface with the chilled water system for cooling water to provide cooling of the PCV atmosphere during normal modes of operation. The system does not perform any safety function for PCV transient or accident mitigation. For more detailed information more information refers to PSAR Chapter 9A, Section 9A.5.6.

The PCCS is based upon proven concepts and simple thermosyphon principles. Heat is rejected to the equipment pool above containment by natural circulation using PCCS pipes as depicted in the configuration on Figure 6.4.3 of PSAR Chapter 6. There are no active components or actuation signals required, as the PCCS is always in service. PCCS is in service during normal operation. However, PCCS does not contribute to the heat removal significantly during normal operation. PCCS is a passive containment heat removal system that maintains the containment within its pressure limits for DBAs such as a LOCA.

The isolation of the reactor containment pressure boundary is an off-normal FSF of DL 3 Safety Category 1. These design features preserve reactor coolant inventory to ensure that adequate core cooling is maintained following a LOCA and to mitigate energy released into containment.

The containment isolation function supports maintaining containment integrity by providing protection against the uncontrolled release of radioactive materials from the containment to the environment as the result of an accident. Redundant CIVs are included in each system line near the containment boundary, that close on predefined parameters to prevent release from containment. Penetrations are included at the boundary between the line and the containment to limit leakage at the connection. Penetrations are designed to maintain structural integrity during the extreme environmental conditions as the result of an accident to protect against leakage. The CIVs, piping between the CIVs and penetrations, and the penetrations are included as a part of the containment boundary. For information on PCS penetrations see PSAR Chapter 6 Section 6.3 and for DBA LOCAs and pipe breaks inside containment see PSAR Chapter 15, Section 15.5.

In addition to CIS containment over-pressurization protection and the strength of the containment structure, overpressure protection for the entire Reactor Containment Pressure Boundary is provided through the NBS and ICS. The ICS in conjunction with a reactor trip provides primary RCPB overpressure protection to limit peak pressure during AOOs. The ICS overpressure protection is detailed in PSAR Chapter 5, Section 5.6.

The PCS has provisions for personnel access and for habitability during plant outages to perform maintenance, inspections, and tests required for assuring PCV integrity and reliability, and the integrity and performance reliability of interfacing SSCs contained inside the PCV enclosure.

The PCS is provided with shielding structures within the containment walls and reactor shield wall (Bioshield). See PSAR Chapter 6, Section 6.3 for further information on personnel and equipment airlocks as well as containment, RPV and maintenance support structures.

Safety Support Systems

The BWRX-300 safety support systems ensure that the FSFs associated with control of reactivity, fuel cooling, long-term heat removal and containment of radioactive materials are available in operational states, DBAs, DBDA's and DECs. Furthermore, the safety support systems provide services such as electrical power, compressed air, water, and HVAC to systems important to safety. Where normal services are provided from external sources, backup safety support systems shall also be available onsite. As described in the following sections, the BWRX design incorporates emergency safety support systems to cope with the possibility of loss of normal service and, where applicable with concurrent loss of backup systems.

The BWRX-300 incorporates safety support systems that include:

- Electrical Power system, see Section 4.5.10 of this document
- Fire Protection System, see Section 4.5.7 of this document
- Habitability Systems, see Section 4.5.12 of this document
- Fuel Cooling System, see Section 4.5.15 of this document
- Isolation Condenser System, see Section 4.5.8 of this document
- Control Rod Drive Hydraulic Scram System, see earlier in this section

Electrical Power System

The BWRX-300 electrical system meets the Safety Support System needs under normal plant power via the main generator or offsite power, backup power provided via Standby Diesel Generators (SDGs), and emergency power provided by safety class 1 (SC1) and Uninterruptible Power Supply (UPSs), for more detailed information see Section 4.5.10 and PSAR Chapter 8, Section 8.1 [R-6].

Electrical power for safety related equipment including Emergency Filter Units (EFUs), dampers, valves and associated I&Cs is supplied from the safety-related uninterruptible power supply. Active safety-related components are redundant, and their power supply is divisionally separated such that the loss of any two electrical divisions does not render the component function inoperable. The EFUs operate during an emergency to ensure the safety of the SCR operators and the integrity of the SCR by maintaining a minimum positive differential pressure inside the SCR habitability area. More detailed information on electrical power for safety support for EFUs can be found in the PSAR Chapter 6, section 6.4.

Fire Protection System

The FPS is detailed in section 4.5.7 of this Application document and PSAR Chapter 9A, Section 9A.6 [R-6].

Habitability Systems

The BWRX-300 does not require immediate operator responses or monitoring for safety since reactor shutdown with passive vessel and containment heat removal are automatically initiated. A robust and reliable MCR housed in the CB and SCR located in the RB provide environments to protect the operators so that they can remain at their posts and operate the reactor facility safely in all operational states and maintain the reactor in a safe condition under all accident conditions. See section 4.5.12 for Control Facilities habitability systems.

4.5.10 Electric Power Systems

The electrical system is arranged into the following five sub-systems and are described in detail in PSAR, Chapter 8:

- Non-Safety Class (SCN) Electrical System,
- Safety Class 2 & 3 Electrical Distribution System,
- Safety Class 1 Electrical Distribution System,
- Generator including exciter system, and
- Switchyard System.

Under the BWRX-300 electrical power systems design, power is supplied to the plant from two independent offsite power sources. These power sources are designed to provide reliable power for the plant's auxiliary loads. The Preferred Power Supply (PPS) consists of the normal preferred and alternate preferred power sources along with those portions of the offsite power system and the onsite power system required for power flow from the offsite transmission system to the Medium Voltage (MV) busses.

As for grid connections, the Generator Step-Up (GSU) Transformer is connected to the Switchyard (S21 System), while the Unit Auxiliary Transformer (UAT) connects to the plant. The generator connects to the GSU and the UAT via the Iso-phase bus with a circuit breaker.

Non-Safety Class Electrical System:

In this system, a 300 MWe generator is connected to the GSU Transformer also called the Main Power Transformer (MPT) and to the UAT, through the Isolated Phase Bus (IPB), via a generator breaker. The GSU Transformer connects to the grid through the Switchyard, while the UAT Transformer connects to the plant supplying normal power to the station.

Safety-Class 2 & 3 Electrical Distribution System

The SC 2 & 3 Electrical distribution system is powered from two separate windings of the UAT to ensure separation of power supplies. Each supply has separate and independent protection schemes.

The System also has two 3.25 MWe Standby Diesel Generators (SDG) poised. In addition to SC2 and 3 loads, the system provide power to some Non-Safety Class loads (plant investment protection loads). Important loads on this system include Fine Motion Control Rod Drives and other load centers. The system has SC2 batteries independent of the SC1 batteries and the off-site diesel AC power.

Safety-Class 1 Electrical Distribution System:

The Safety-Class 1 Electrical Distribution System (EDS) provides uninterruptible power to SC1 loads, including automatic shutdown and decay heat removal. The SC1 EDS also powers the DCIS.

The system also has the SC1 batteries, battery systems and UPSs. They provide backup for at least 72 hours, following loss of AC power. The batteries are in separate rooms that are fire and flood barriered, in addition to being seismically and environmentally qualified.

Generator System:

The generator system consists of a Generator and Exciter (GEN) system that converts the rotational energy of the BWRX-300 turbine into electricity and distributes that to the grid. The system also includes the exciter subsystem (GES) and automatic voltage regulator (AVR), which controls the characteristics of the electrical output of the generator. The generator shaft is connected to the main turbine equipment system.

Switchyard System:

The switchyard system provides connection of the plant's on-site electrical systems to the utility transmission grid. There are two main connections to the switchyard, one line to output power from the Generator Step Up (GSU) Transformer and one line to supply the Reserve Auxiliary Transformer (RAT). Protective relaying schemes used for protection and monitoring of the offsite power circuits and transformers are redundant and include backup protection features. For further details, refer to PSAR Chapter 8, Section 8.4 [R-6].

Safety Support System:

The BWRX-300 normal plant power is provided via the preferred power supply, backup power via SDGs, and emergency power by SC1 Direct Current (DC) and UPS system. Normal and backup power supplies are available onsite. Emergency power supply can cope with the possibility of concurrent loss of normal and backup power supplies. The Standby Power System is provided by the SC 2&3 electrical system, and the Emergency Power System is provided by the SC1 electrical system.

The electrical system is designed with sufficient capacity to meet load requirements to support FSFs. The design incorporates an appropriate DiD strategy to ensure availability and reliability of the supported systems.

Emergency Power Supply:

The emergency support systems have been designed to:

- Be independent of normal and backup system
- Support continuity of the FSFs until long-term (normal or backup) service is re-established:
    - Without the need for operator action to connect temporary on-site services for at least eight hours
    - Without the need for offsite services and support for at least 72 hours
- Have a capacity margin that allows for future increases in demand
- Be testable under design load conditions, where practicable

As described in PSAR Chapter 8 Section 8.4, the standby and emergency power systems are represented in the SC1 EDS and the SC2 and 3 EDS. The design of these systems ensures sufficient capacity to support associated safety functions, and that availability and reliability of the system is commensurate with the safety significance of the connected loads.

The BWRX-300 electrical systems are designed to accommodate grid disturbances within the range of those shown to be possible by transmission operator studies at the point of interconnect. Such studies include maximum and minimum transient and steady-state grid voltages, available fault current, and consideration of the effect of geomagnetic induced currents resulting from coronal mass ejections. The design includes appropriate mitigations based on the results of studies such as the application of automatic load tap changes, surge suppression, transformer zero sequence current monitoring, and blocking capacitors.

OPG's DNNP BWRX-300 plant does not require either onsite or offsite medium voltage electrical power to achieve any safety-related/SC1 requirement. Although the BWRX-300 does not require any on-site or offsite AC power for safety, the permanently installed SDGs provide standby AC power and generally meet all requirements of REGDOC 2.5.2. The SC1 batteries are sized for 72 hours which allows for either of the two SC3 SDGs to start, portable generators to be connected, or off-site power to be restored.

The SC 2&3 electrical system includes permanently installed SDGs that can provide standby AC power and have sufficient onsite fuel reserves for 7 days. This ensures enough fuel is available to operate standby and emergency power sources while supplying connected loads.

The BWRX-300 electrical system design provides provisions to address any occurrence of common-cause failures involving loss of normal power supply and standby power supply, as the SC1 batteries are electrically independent, physically separate and diverse from other electrical systems. The SC1 batteries also mitigate any challenges to the standby and emergency system due to electrical system disturbances or transient conditions. Furthermore, if off-site power is not restored AC power is provided via the SDGs or via permanently installed EME connections for use with portable generators.

For further details, refer to PSAR Chapter 8, Section 8.4 [R-6].

4.5.11 Instrumentation and Control

Instrumentation and Control (I&C) Function Allocation to Defence Lines

The BWRX-300 I&C functions are provided by a DCIS. The DCIS is an integrated control and monitoring system, for the power plant, designed to support the plant safety strategy.

The DCIS consists of several systems, arranged into three safety classified segments. Each system is designed to the appropriate level of hardware and software quality corresponding to its function and DL location. The DCIS provides control, monitoring, alarming and recording functions. The various components of the DCIS can operate independently. For more information see PSAR Chapter 7, Section 7.1 [R-6].

The BWRX-300 has specific safety functions that must be provided by the I&C system. In general, these include reactor shutdown (making the reactor subcritical), isolation of the reactor vessel and containment, initiation of normal and emergency heat removal, and control and monitoring. The safety systems are generally autonomous, but can be initiated manually, and monitoring is required (especially post-accident monitoring for both monitoring the course of an accident and enabling appropriate operator intervention).

To address these safety functional requirements, the plant I&C includes an integration of three systems allocated to the plant DLs as follows:

- Safety Class 1 I&C System, allocated to DL3 and comprised of SC1 Structures, Systems and Components
- Safety Class 2 I&C System, allocated to DL4a and comprised of SC2 SSCs
- Safety Class 3 I&C System, allocated to DL2 and DL4b comprised of SC3 SSCs

Additionally, a fourth partition of the DCIS (Non-Safety I&C System) performs functions that are not needed for the plant safety functions. For further details, see PSAR Chapter 7, Section 7.1.

Defence Line 2

DL2 contains plant functions designed to control or initiate responses to PIEs and AOOs before any parameters reach the DL3 actuation setpoint.

The DL2 function is assigned to Safety Class 3. Safety Class 3 functions are implemented on the Safety Class 3 System I&C hardware and software platforms; these functions and their support systems can use active equipment. Additional detail of the DL2/SC3 system architecture and safety functions can be found in PSAR Chapter 7, Section 7.3 [R-6].

Defence Line 3

DL3 contains plant functions that act to mitigate the impact of a PIE, and those credited to maintain the plant in a safe condition following mitigation of PIEs, until normal operations are resumed. DL3 functions are needed when DL2 is not effective at intercepting a PIE, or when a PIE is beyond the capabilities of the DL2 functions. The DL3 function is assigned to Safety Class 1. The Safety Class 1 functions are implemented on the Safety Class 1 System I&C hardware and software platforms.

The main safety functions of DL3 logic are the initiation of the scram, reactor and containment isolation, and ICS initiation functions. The systems that encompass DL3 for safety functions are:

- Reactor Trip System
- Isolation Condenser Isolation
- Emergency Core Cooling System
- Control Room Habitability
- Communications

Additional detail of the DL3/SC1 system architecture and safety functions can be found in PSAR Chapter 7, section 7.3 [R-6].

Defence Line 4

DL4 consists of DL4a and DL4b subsets of functions.

DL4a functions are those that can place and maintain the plant in a safe state in case of PIEs with complete failure (i.e., CCFs of the DL3 functions (e.g., DECs without core damage). The DL4a functions are needed in the event of specific, postulated CCFs occurring in DL3. Thus, DL4a complements DL3 Safety Category 1 systems by providing a high availability for plant safety functions by providing comparable DL3 safety functions using diverse hardware and software platforms from DL3.

The DL4a function is assigned to Safety Class 2 and is performed by at least Safety Category 2 equipment, except for SSCs that are only needed after the first 7 days of the event, which are classified as Safety Category 3. DL4a functions must be performed diversely from corresponding portions of functions in DL3 providing protection for the same event.

Safety Class 2 functions are implemented on the Safety Class 2 System Instrumentation & Control hardware and software platforms; these functions and their support systems are allowed to use active equipment. Additional detail of the

DL4a/SC2 system architecture and safety functions can be found in PSAR Chapter 7, Section 7.3 [R-6].

DL4b functions are those that can prevent or mitigate a severe accident while keeping radioactive releases to acceptable levels, and protect for extreme events, multiple events, or multiple failures that can defeat DL2 – DL4a.

The DL4b function is assigned to Safety Class 3 and is generally performed by at least Safety Category 3 equipment. DL4b is for severe accidents and is not incorporated into the DCIS system except for monitoring and alarm. Additional detail of the relationship of DL4b functions to plant level lines of defence can be found in PSAR Chapter 7, Section 7.1.

I&C Functions Not Allocated to a Defence Line

Non-safety classified I&C functions are implemented on the Non-Safety Instrumentation & Control System hardware and software; these may be vendor supplied, packaged systems, connected to the DCIS through a Non-Safety Instrumentation & Control System gateway. This equipment may be classified as Safety Category 3, despite not being in a DL, because the controller design for non-safety equipment is reliable enough that they will not fail often enough to cause an AOO.

The Non-Safety Class Instrumentation & Control system controllers feed the Safety Class 2 and Safety Class 3 Instrumentation & Control System BOP segment, which contains the control and monitoring systems associated with power generation and support systems. For further details, see PSAR Chapter 7, Section 7.3 [R-6].

Human Factors Consideration

Safety Class 1 Instrumentation & Control System data are available on their appropriate divisional displays and these displays are subject to a Human Factors Engineering (HFE) evaluation to facilitate operator response, by presenting information that is necessary and appropriate, in a usable format, and that is compatible with task needs in the context under which the controls and information will be used.

Each division has its own displays in both the MCR and SCR and locally in the Safety Class 1 Instrumentation & Control equipment rooms.

To reduce the likelihood of human errors and time needed to resume monitoring and control tasks within the SCR, the layout and HSI of the SCR match the available subsections of the MCR to the extent possible.

Each divisional display is dedicated to monitoring or controlling its associated division.

The Safety Class 1 Instrumentation & Control displays also include an alarm system to provide for operator awareness and to prompt to the displays containing further information relating to the alarm. The displays are menu driven and it is possible to switch between related displays without going through the main menu.

Some Safety Class 1 Instrumentation & Control monitoring signals provide necessary information to achieve or verify correct plant response to transients and accidents. The specific accident monitoring variables that the control room operator should monitor to ensure safety during an accident and the subsequent long-term stable shutdown phase are determined through the normal BWRX-300 design process using the HFE process.

The Safety Class 1 Instrumentation & Control signals are alarmed to prompt the operator to an appropriate display (e.g., range check problems or alarm setpoints exceeded). The Safety Class 1 Instrumentation & Control signals and the various Safety Class 1 Instrumentation & Control components are alarmed for self-diagnostics. See PSAR Chapter 7, Section 7.3 [R-6].

The Human Factor Interfaces are integrated into the design architecture for each system in the control and monitoring areas including the MCR, supplementary control points (e.g., SCR), local control panels, and any emergency control location. The HFE program is described in Section 4.5.3 of this Application document and PSAR Chapter 18.

Reliability and Sharing

Design principles for the DCIS include independence, diversity, separation, and redundancy. These principles are applied to systems important to safety on a graded approach, commensurate with the safety significance of the SSC to ensure high reliability, fail-safe operation, and reduction of CCFs. The single-failure criterion is also incorporated, meaning that the safety groups are still functional in the presence of a single failure. See PSAR Chapter 7, Section 7.3 [R-6].

PSAR Chapter 3 Section 3.7 provides design information on the I&C systems and components related to reliability. Additional information on the DCIS system design and the reliability of the I&C system is also discussed in PSAR Chapter 7, Section 7.3.

4.5.12 Control facilities

The BWRX-300 control facilities include the MCR, SCR, and the Emergency Response Facility (ERF).

Main Control Room

The MCR is designed to be optimal for nominal shift complement. The MCR is sized and designed to accommodate an expected maximum number of people on a continuous basis. This maximum number will be based on the MCR nominal staffing plus additional staff (e.g., trainees, observers, etc.). The capability for having additional people provides the necessary facilities to support plant conditions and evolutions that require more than the normal operations complement. The size and layout of the MCR is driven by the staffing analysis as part of the HFE activities, described in PSAR Chapter 18, Section 18.2. The required MCR personnel have workstations designed to support their specific control, information, communication, and work coordination needs. Each MCR workstation also contains space for a role trainee. See PSAR Chapter 7, Section 7.5 for further details.

The MCR layout design ensures tasks can take place in a position that does not result in blocking views of the Group-View Display. The arrangement of the workstations in the MCR also facilitates communication between all MCR personnel, including direct conversation and visual contact. MCR design also fosters the oversight, command and control responsibilities of the required supervisory roles. The layout of workstations provides sufficient clearance for foot traffic and maintenance between consoles. The layout of workstations provides enough space for the proper storage, placement, and use of tools and procedures in the MCR. For further information on MCR layout, see PSAR Chapter 7, Section 7.5.

The MCR operator workstation is made up of three sections: two SC3 workstation sections and one SC1 workstation section. The MCR operator workstations contain the controls and indications needed to perform the assigned MCR tasks. VDU displays are designed using the HFE processes described in PSAR Chapter 18, Section 18.3.

Secondary Control Room

The SCR is utilized to perform the functions required to keep the plant in a safe state when the MCR is unavailable. The SCR includes suitable facilities for habitability and well as workspace for tasks to support required usage. The SCR has a suitable supply of food and water. The SCR also contains adequate space and provisions for sleeping as required by the postulated scenarios in which it is used. See PSAR Chapter 7, Section 7.6 for further details.

The SCR is designed to accommodate the expected staffing based on HFE staffing analysis for the expected usage conditions, as described in PSAR Chapter 18, Section 18.2. A suitable number of workstations are provided to support the specific task, communication, and work coordination needs for expected personnel.

The SCR is accessible via a qualified, redundant and protected access path that is available under required emergency conditions to permit the operation staff to safely leave the MCR and reach the SCR in the event of fire, toxic gas, or aircraft impact security event, damaging earthquake or tornado, or extended loss of all AC power in the MCR. See PSAR Chapter 6 Section 6.4, Chapter 7 Section 7.6 and Chapter 19 Section 19.1 for further details.

The layout of the workstations and HSI in the SCR provides the personnel with adequate information to assess the plant state and perform actions to maintain the plant in a safe state, if required for the expected usage scenarios. To reduce the likelihood of human errors and time needed to resume monitoring and control tasks within the SCR, the layout and HSI of the SCR are designed to be consistent with the MCR to the extent possible.

The SCR SC3 workstation section contains two VDUs to support monitoring, control, and alarm management. The SC3 workstation VDUs can display any provided plant and system information, as supported by the I&C architecture. SC3 VDUs allow for flexible use and are not divisionally separated or dedicated. These VDUs are controlled by a wired keyboard and mouse. See PSAR Chapter 7, Section 7.6 for further details.

Emergency Response Facilities

The Site Management Centre (SMC) is separate from the plant MCR and SCR and located onsite, but outside of the protected area.

SMC is provided for use by technical support staff and emergency support staff in the event of an emergency. The SMC includes equipment, facilities, and communication for trained staff to control and coordinate any emergency response and provide technical support to operations, emergency response organizations, and severe accident management evaluation.

Further details on this ERF and other facilities to support emergency response are provided under PSAR Chapter 7, Section 7.7 and Chapter 19 [R-6].

Control Facility Habitability

The control facilities contain habitability systems to enable essential workers to operate the reactor facility safely in all operational states, or to maintain the reactor facility in a safe condition under all accident conditions considered in the safety case.

The major systems that make up the habitability systems are:

- The Control Room Heating and Ventilation System (CRHVS) provides climate control. See PSAR Chapters 6 and 9A, Sections 6.4, 9A.5.1 and 9A.5.2 [R-6].
- The Process Radiation Monitoring System monitors for radiation and actuates air filtration with CRHVS. See PSAR Chapters 6, 11, and 12, Sections 6.4, 11.5 and 12.3.13.
- Lighting System provides normal and emergency lighting. See PSAR Chapter 9A, Section 9A.9.2.
- Fire Protection System (FPS) provides smoke detection, protection against fire, and appropriate alarms. See PSAR Chapter 9A, Section 9A.6.

Identification and Minimization of Threats to Control Facilities

The BWRX-300 design considers and provides protections against natural and human induced external and internal hazards for the control facilities. Such hazards include seismic concerns, extreme weather conditions, extreme hydrological conditions, aircraft crashes, missiles, fires, explosions and toxic gases, flooding, high energy line breaks, etc. For example, cabling for the I&C equipment in the MCR is separated from the cabling in the SCR to achieve fire isolation.

For identification and evaluation of external hazards see PSAR Chapter 3, Section 3.3, and for internal hazards, see PSAR Section 3.4 [R-6].

Safety Parameter Display System

The Safety Parameter Display System (SPDS) provides big picture overview display of plant parameters during transients and accidents. The SPDS are a pair of redundant computers that have access to plant data through the plant networks. SPDS information is displayed on dedicated displays, but the operator can call up any SPDS display on local Visual Display Units (VDUs) in either control room. Reference PSAR Chapter 7, Section 7.3 [R-6]. The SPDS is designed to use the same accident monitoring information that is provided to the ERF.

Communication to Emergency Response Facilities

For information on communication to the ERF, see PSAR Chapter 7, Section 7.7, Chapter 19 Sections 19.1 and 19.2, and Chapter 9A 9A.9.1 [R-6].

Lighting System

Normal lighting provides illumination during all plant operating conditions. Emergency lighting provides illumination during all plant operating conditions, including fire, transient, and accident conditions. The control room lighting equipment is designed to support reliable human performance, based on HFE requirements. The MCR and SCR lighting over the SC1 video display units and the communications console are powered from 3 divisions of the SC1 Electrical Distribution System. The remaining MCR and SCR lighting is split between the redundant diesel backed SC2 and SC3 power system. Reference PSAR Chapter 9A Section 9A.9.2 for more information on lighting [R-6].

Fire Protection System

The CRHVS detects and limits the introduction of airborne hazardous materials (radioactivity or smoke) into the control room. This system is designed with the capability to exhaust smoke, heat and gaseous combustion products from inside the Power Block to the outside atmosphere in the event of a fire. Further details are provided in PSAR Chapter 9A, Sections 9A.5.1 and 9A.5.2.

In addition, the FPS provides smoke detection and appropriate alarms. For further details see PSAR Chapter 9A, Section 9A.6 [R-6].

Shielding

The MCR/SCR is shielded against radiation from both normal operation and accident sources. Shielding is to ensure that personnel exposure in the MCR/SCR, following an accident, does not exceed dose limits. See PSAR Chapter 12, Section 12.3 for more information on control room shielding. For the radiological analysis demonstrating the shielding requirements for the control room habitability areas, reference to PSAR Chapter 15, Section 15.5.

Food & Water Supplies

The SCR has provisions for emergency food, water, storage, and air supply systems for 7 days. The SCR also contains adequate space and provisions for sleeping.

Additional Equipment to Support Habitability

There is specific equipment in place to support the habitability functions of the Control Facilities.

The Emergency Filter Units (EFUs), EFU Fans, and Shutoff, Balancing, and Backdraft Dampers provide filtered air to the control room habitability areas and provide positive differential pressure. The EFU related ductwork provides low leakage components necessary to maintain the habitability area. The Isolation Dampers and Valves isolate the air supply to the control rooms when there is a loss of AC power or radiological isolation event. Additionally, Tornado Protection Dampers mitigate the effect of a design basis tornado.

Specific design information regarding these components and others can be found in PSAR Chapter 6, Section 6.4 [R-6].

DCIS software

The characteristics of computer software vary across the DCIS is designed with diverse platforms and functional allocation based on Safety Requirements.

Software diagnostics include watchdog timers, sensor range checks, power supply monitoring, communications, actuator monitoring and inconsistencies between redundant signals.

See PSAR Chapter 7, Section 7.3 for hardware and software qualification details.

Data and Displays in the Control Rooms

The BWRX-300 I&C segments are interconnected through a redundant network to support common services like alarming, control room displays, recording and sending data. Alarm information is available in both control rooms and in the SMC.

Long term storage is accomplished using Plant Historians on the SC3 system. Plant Historians provide archiving of all the plant control, alarm, and monitoring system information. Specific detail on Plant Historians and Monitoring can be found in PSAR Chapter 7, Section 7.3 [R-6].

All alarms are available to the operator and all alarms are recorded. The operator can display all currently active alarms, acknowledge the alarms and monitor sequence of alarms. The data is recorded at least once a second, depending on the parameter, some are recorded at higher rates, and some are recorded at lower rates. The alarm monitoring system is also capable of trending data on control room displays and hardcopy printouts.

The alarm systems are redundant pairs of industrial PCs that each monitor all plant data. Each alarm system has redundant UPS and power feeds. The alarms are available even with a single alarm system failure.

The SCR has an emergency communications workspace located and designed with suitable provisions to minimize disturbance to the operating crew. The emergency communications workstation contains VDUs or Wi-Fi-enabled tablets for connection to business LAN with at least one VDU with displaying SPDS indications (i.e., no control) to monitor and report externally on plant conditions. The emergency communications workspace also contains space for communications equipment, procedures, and drawings in support of performing administrative tasks. See PSAR Chapter 7, Section 7.6 for more details.

Implication and mitigation of the failure of reactor facility computers

The reactor facility computers are designed to be fault-tolerant throughout the entire design of the computer architecture. Any single failure will not compromise the system from performing its safety related functions, and essential information for operators will be maintained.

The inherent design of the I&C system ensures safety critical functions are performed even with a complete failure of a safety related computer. The diverse and physical separation of SC1 and SC2 safety related computer systems are designed to operate independently.

Process data, alarms, and diagnostic information are transmitted back to control room operators through a diverse and redundant communication network. See PSAR Chapter 7, Section 7.2 [R-6] for more information on the data network.

Synchronization of computer systems

The computer systems within the DCIS are designed to operate independently minimizing the need for data processing synchronization. However, the divisions of the SC1 I&C system, which is made up of three divisional controllers, communicate their trip decisions in real time and each division determines if at least two have tripped.

4.5.13 Steam supply system

The Main Turbine Equipment (MTE), steam supply system and associated piping, and valves, are located within the TB. The MTE begins at the two main steam turbine stop valves and control valves, which admit steam through separate steam leads to two inlet nozzles of the single flow High Pressure (HP) turbine. Steam within the HP turbine is directed to the Moisture Separator Reheater (MSR) where it is reheated before entering the Low Pressure (LP) turbines. Some of the steam from both, the HP and LP turbines, is utilized for feedwater heating. Exhaust steam from the LP turbines is directed to the condenser. The Generator, Exciter, and Isophase Bus Duct (GEI) system converts the rotational energy of the main turbine into electricity and distributes that to the grid through the isophase bus duct to the Generator Step-Up (GSU) transformer and Unit Auxiliary Transformer (UAT). See PSAR Chapter 10, Section 10.2 for further details.

Turbine Generator

The turbine is a two-stage reheat, condensing steam turbine. The turbine converts the steam energy from the reactor into rotational motion that the generator turns into electrical energy. Steam flow to the turbine is controlled via stop valves, control valves and other valves which allow power output control.

Radiation shielding for personnel protection is provided for all required components of the MTE as discussed in PSAR Chapter 12.

The turbine is connected to a standard 60hz generator. The generator system has no safety equipment required for the safe and orderly shutdown of the plant or to mitigate a DBAs. This includes the excitation subsystem generator exciter subsystem (GES).

For a more detailed description of the MTE and generator system, refer to PSAR Chapter 10, Section 10.2 [R-6].

Turbine generator protection systems

The Turbine Generator is installed on the turbine deck within the TB to minimize the potential of turbine missiles damaging any safety-category function equipment or structures.

A separate and redundant turbine overspeed protection system is included to minimize the possibility of turbine rotor failure and potential turbine missile damage. Additionally, the valve arrangements and closure times are such that a failure of any single valve to closure will not result in unsafe turbine overspeed in the event of a trip signal. The overspeed sensing devices are in the turbine front bearing standard and are therefore protected from the effects of missiles or pipe breakage. See PSAR Section 10.2.3.5.1 for further details.

Turbine integrity is discussed in PSAR Chapter 10, Section 10.2. The effects of potential high energy missiles are discussed in PSAR Chapter 3 Section 3.5 and Chapter 10 Section 10.2.

Turbine Bypass System

The Turbine Bypass System (TBS) takes excess steam, expands it to lower pressure, and delivers it to the condenser. The system is used during low power load rejections and during plant start-up. The detailed SDD including its piping and components is provided in the PSAR Chapter 10, Section 10.7.

Moisture Separator Reheater (MSR) System

The normal function of the MSR is to remove moisture from and reheat the HP turbine exhaust prior to admittance to the LP turbines. Removed moisture and condensed steam from the MSR drain into drain tanks. The MSR is a passive system and does not perform any active Safety Category function or active non-safety category (NSC) functions.

The MSR consists of one stage of moisture separation and two stages of reheating and is in the TB. The MSR system is protected by safety relief valves that ensure that it is not over pressurized under the most adverse operating conditions of the turbine generator.

The detailed SDD including its components is provided in PSAR Chapter 10, Section 10.6 [R-6].

Main Condenser and Auxiliaries System

The Main Condenser and Auxiliaries (MCA) System is the heat sink for the power generation and normal reactor cooldown and plant start-up activities.

The primary function of the main condenser is to receive and condense the turbine exhaust steam and turbine bypass steam during various phases of plant start-up, normal operation, and shutdown. The MCA are not required to operate during or after a design basis event. For a detailed description on MCA, refer to PSAR Chapter 10, Section 10.5.

Main Steam Piping/Lines (MSL) and Valves

The MTE, piping and vessels are separated and/or isolated from Safety Category function SSCs and electrical and control systems, as applicable.
The turbine system piping includes the main steam leads from the main steam control valves to the HP turbine steam inlet nozzles, and HP turbine and LP turbine exhaust hoods instrumentation piping.

The turbine system pipe thicknesses are calculated with the appropriate erosion-corrosion allowance to preclude failure during design life, ensuring that there is sufficient margin in the design such that pressure boundary limits are not exceeded in operational states and DBAs.

The design pressure and temperature of the MSL piping and components shall not exceed the established pressure–temperature rating for the components and material used in the main steam system. The main turbine is designed and qualified to full reactor design conditions. See PSAR Chapter 10, Sections 10.2 and 10.4 [R-6] for further details.

4.5.14 Auxiliary systems

Circulating Water System

The Circulating Water System (CWS) is a recirculating system that supplies cooling water to the main condenser from the NHS. The circulating water pumps located in the intake structure of the NHS, take suction from the intake basin and discharge through a common underground pipe. In the TB, the piping comes above grade and routes into the tube side of the main condenser to act as the cooling flow for condensing main turbine exhaust steam. Circulating water exiting the main condenser is piped back underground to the NHS.

A hot circulating water return line is also provided to recycle water returning from the condenser in cold weather conditions as required to prevent freezing in the NHS intake area and moderate cold water into the condenser to improve turbine performance.

As part of DL2, the CWS has an integral support function to cool PCW. The CWS provides a means to reject heat from the MCA to the environment through the NHS. The design of the CWS meets CNSC requirements specified in REGDOC-2.5.2 as it relates to heat transfer to an ultimate heat sink. See PSAR Chapter 10, Section 10.8 for further information.

Plant Cooling Water Systems (PCW)

The Plant Cooling Water (PCW) system provides cooling water to Non-Safety and Safety Class 3 components and provides a barrier against radioactive contamination of the CWS as per PSAR Section 10.8. It consists of two piping subsystems, RCCW Piping Distribution and TCCW Piping Distribution, that provide cooling water to various heat exchangers.

The safety classification of the PCW as well as interfacing SSC is consistent with the requirements of CNSC REGDOC-2.5.2.

Portions of the PCW necessary to support Safety Class 3 functions to provide temperature control of the fuel pool, and cooling for the SDC heat exchanger, are designed to Non-Seismic Category, Quality Group D requirements. The redundant design elements of the PCW design ensure that the safety function of the system is maintained. See PSAR Chapter 9A, Section 9A.2 [R-6] for further information.

Condensate Filters and Demineralizer System

The Condensate Filters and Demineralizers System (CFD) purifies the condensate to maintain reactor FW purity. The CFD uses filtration to remove suspended solids, including corrosion products, and ion exchange resin to remove dissolved solids from condenser leakage and other impurities. The CFD is a full-flow system that consists of high efficiency backwash type filters followed by mixed bed demineralizers. Refer to PSAR Chapter 10, Section 10.3 for further information.

Condensate Feedwater Heating System

The CFS provides high purity feedwater to the RPV at the required flow rate, pressure, and temperature. The CFS interfaces with the NBS just before injection into the reactor vessel located in the RB.

The CFS system is primarily SC3. The CFS provides for closure of the Containment Isolation Valves (CIV)s and system isolation valves when signaled to do so as discussed in PSAR Section 10.3.2.1. The containment isolation components, system isolation components, leak detection components, and flow measuring devices are SC1. Containment isolation is provided in accordance with REGDOC-2.5.2. See PSAR Chapter 10, Section 10.3 for further information.

Reactor Water Cleanup System

The RWCU provides blowdown-type cleanup flow for the RPV during the reactor power operating mode. Cleanup or filtration and ion removal is performed by the CFD as per PSAR Chapter 10 Section 10.3.1. The RWCU provides an over-boarding flow path to the condenser hotwell or LWMS as per PSAR Chapter 11, Section 11.2.

The RWCU Safety Class 1 functions include continuous monitoring for leakage utilizing density compensated differential flow measurements. If a leak is detected, in either normal or off-normal conditions, the system is isolated from the RCPB using system isolation and containment isolation valves.

The arrangement of SSCs minimize the possibility of compromising the functionality of both the SC1 functions and their backup systems and equipment. The portions of the RWCU that perform containment isolation are in the RB. This provides protection against natural phenomena ensuring the ability of the RWCU to perform its Safety-Category functions. See PSAR Chapter 9A, Section 9A.2.2 for further information.

Shutdown Cooling System (SDC)

The SDC System provides for decay heat removal when shutting down the plant. The SDC comprises two independent pump and heat exchanger trains. These trains together provide redundant decay heat removal capacity. The major components of each train are a pump and a heat exchanger along with valves, piping, I&Cs, and power inputs.

The SDC system also reduces RPV inventory and can be used in conjunction with RWCU piping to reduce RPV thermal stratification. See PSAR Chapter 9A, Section 9A.2 for further information.

The SDC has two Safety-Class functions: Decay Heat Removal, and Leak Detection and Isolation. Plant systems retain sufficient heat removal capacity to prevent boiling in the reactor during periods of plant shutdown with the RPV head removed, and to simultaneously prevent boiling in the fuel pool, assuming a LOPP event and the most limiting single active failure. See PSAR Chapter 9A, Section 9A.2 for discussion of SDC Safety-Category functions. A detailed description of the SDC system can be found in PSAR Chapter 9A, Section 9A.2.3 [R-6].

Chilled Water Equipment

The CWE provides chilled water cooling to the Heating Ventilation and Cooling Systems throughout the plant, to the offgas cooler condenser, to the charcoal adsorber vault Fan Coil Units (FCUs), and to the CCS in the RB.

The safety classification of the CWE as well as interfacing SSC is consistent with the requirements of REGDOC 2.5.2.

As part of DL3, CWE containment isolation valves on piping that penetrate the containment boundary are designed to close upon receiving an isolation signal from the DCIS.

A detailed description of the CWE system can be found in PSAR Chapter 9A, Section 9A.2.

Normal Heat Sink

The NHS is a once-through cooling water system design. The NHS includes Intake Tunnel (located deep in Lake Ontario), Discharge Tunnel and diffusers and Pumphouse/Forebay.

Water flows into the intake structure forebay from Lake Ontario by means of an intake tunnel. Water flows through the CWS system where it absorbs the heat from the main condenser and the PCW heat exchangers. The water is discharged through the discharge tunnels to the diffusers and back into Lake Ontario. The use of diffusers limits the temperature increases thus mitigating impacts to aquatic habitat. A recirculation line from the CWS discharge line to the intake is provided to moderate NHS temperature at the Forebay of the NHS, as needed.

A detailed description of the NHS system can be found in PSAR Chapter 9A, Section 9A.2 [R-6].

Fire Protection Water Supply System

The FPS provides early fire detection and suppression to limit the spread of fires. The FPS is part of the overall fire protection program including the plant design and layout to prevent or mitigate fires and includes administrative controls and procedures.

The type of fire suppression is based on the combustible loading and the safety function of the equipment within a fire area.

The FPS includes two firewater storage tanks. Physical separation of redundant storage tanks ensures protection against CCFs such as seismic events and tank rupture. Each tank is equipped with a freeze protection system as necessary based on local climate conditions.

An automatic fire detection, alarm, supervisory control, and indication system is also provided in selected areas of the plant, as required by the Fire Hazards Assessment [R-90] for personnel safety and fire brigade notification.

A Main Fire Alarm Panel (MFAP) located in the MCR, monitors, and receives system actuation, supervisory, and trouble alarm signals from the individual local panels.

Detailed information of FP water supply system can be found in PSAR Chapter 9A, Section 9A.6 [R-6].

Potable Water System (Interfacing Water systems)

The Potable Water System (PWS) provides makeup water to the demineralized water trailers if needed. The PWS consists of potable water supply piping from the Municipality of Durham. The supply line includes isolation valves, flow totalizer and instrumentation. Heat tracing is provided for all outside above ground piping and instrumentation. A distribution header supplies the power block and other onsite buildings. Provisions are made in the distribution header to accommodate future demand.

A detailed description Potable water system can be found in PSAR Chapter 9A, Section 9A.9.

Makeup Water System

The MWS provides normal makeup water to the unit as well as additional capacity to refill the ICS pools in a reasonable timeframe following a typical initiation.

Demineralized water of the proper water quality specification is supplied to the single demineralized water storage tank located in the site yard. The tank is heated to prevent freezing. Two 100% pumps supply the demineralized water system.

The MWS consists of supply piping from the DNGS site, trailer hookup piping, storage tank, transfer pumps, and distribution piping.

A detailed description of the MWS can be found in PSAR Chapter 9A, Section 9A.9 [R-6].

Sanitary Water System

The Sanitary Water System collects sewage from the facility and transfers it to the existing DNGS East Sewage Lift Station. The sewage system is a non-radiologically contaminated system and collects sewage only from areas outside of any radiologically controlled areas in the CB.

The Sanitary Sewage Handling Subsystem is designed to prevent raw sewage overflow in the event of a power outage. The Sanitary Water System is designed for operation in all modes.

A detailed description of the Sanitary Water system can be found in PSAR Chapter 9A, Section 9A.9.

Process Auxiliaries

Plant Pneumatic System

The Plant Pneumatics System (PPS) provides a continuous supply of compressed air for applicable plant air demands. In addition, the PPS supplies oil-free air to the portable breathing air filtration systems. The PPS also distributes gaseous nitrogen for valve actuators and other SCCV users inside containment.

As part of its Safety-Category 3 functions, the PPS provides dry, oil-free, filtered, compressed air for valve actuators, Safety-Category instrument control functions, and general instrumentation services outside of the SCCV.

A detailed description of the Plant Pneumatic system can be found in PSAR Chapter 9A, Section 9A.4.

Equipment and Floor Drain System (EFS)

The Equipment and Floor Drain System drains, collects, and transports liquid waste from the floor drains to the Liquid Waste Management (LWM) system for processing.

The floor drain system consists of a clean (non-radioactive) drain subsystem and contaminated subsystems that are designed to transfer liquid waste to the LWMS for processing as either non-radioactive or radioactive waste. The EFS consists of drainpipes, collection sumps, sump pumps, interconnecting piping, and instrumentation to provide for the collection and removal of liquid waste in the plant. The EFS is provided for the RB, TB, RWB and PLSA.

The EFS is designed to accommodate the maximum anticipated normal volumes of liquid including the anticipated water flow from a fire hose and other fire suppression water discharges to the area floor drains.

To preclude inadvertent transfer of radioactive liquids to non-radioactive systems, the radioactively contaminated or potentially contaminated liquids are collected by separate systems (e.g., no cross connections) from those that collect non-radioactive liquids. Redundant sump pumps are included to increase the reliability, availability, and maintainability of the EFS.

A detailed description of the Floor Drain system can be found in PSAR Chapter 9A, Section 9A.9 [R-6].

Hydrogen Water Chemistry (HWC)

The Hydrogen Water Chemistry System adds hydrogen into the feedwater and oxygen (as a constituent of air) into the Off-gas system via the Service Air system. HWC is included in the BWRX-300 design for the purpose of reducing and mitigating Intergranular Stress Corrosion Cracking (IGSCC) in reactor vessel internals.

The mitigation of IGSCC is achieved by the reduction of oxygen and other oxidizing species (oxidants) in the reactor coolant by injecting hydrogen into the feedwater at the feedwater pump suction. The injected hydrogen suppresses the radiolytic formation of oxidants in the reactor core. To compensate for any excess hydrogen which may travel downstream and be removed from the main condenser by the Off-gas System, a corresponding amount of oxygen, as a constituent of injected air provided by the Service Air System, is injected into the Off-gas System prior to the Hydrogen Recombiner.

The BWRX-300 design employs NobleChem™ technology to provide the injection of noble metal(s) into the reactor to aid in the protection of reactor vessel internals from IGSCC in combination with the addition of hydrogen by the HWC. This is an established technique for reducing and preventing the growth rates of IGSCC in reactor vessel internals.

A detailed description of the Hydrogen Water Chemistry system can be found in PSAR Chapter 9A, Section 9A.9 [R-6].

Containment Inerting System

The Containment Inerting System (CIS) is designed to establish and maintain an inert atmosphere (nitrogen) within the SCCV. An inert atmosphere is maintained in all operating modes except during outages. Prior to reactor shutdown the CIS can de-inert the SCCV allowing for safe personnel access without the need of a breathing apparatus. The principal objective of the CIS is to preclude the development of a combustible atmosphere by maintaining an oxygen deficient atmosphere inside SCCV.

A detailed description of the Containment Inerting system can be found in PSAR Chapter 9A, Section 9A.4.2.

Process and Effluent Radiological Monitoring and Sampling Systems

The PREMS provides continuous monitoring to allow determination of the content of radioactive material in various gaseous and liquid process and effluent streams. Radiation monitoring is performed during normal, abnormal and accident conditions.

The purpose of PREMS is to ensure the environmental safety and radiation protection instrumentation functions are in place, and to initiate protective actions to limit the potential release of radioactive materials to the environment if process or effluent stream radiation limits are exceeded. A second function is to provide plant personnel with alarms in support of personnel exposure prevention and potential environmental issues. As such, this system is required to be in operation prior to initial fuel load. More details about the PREMS system are given in PSAR Chapter 11, Section 11.5 [R-6].

More design information related to subsystems and components that provide radiological monitoring functions are described in PSAR Section 11.5.

Process Sampling System

The Process Sampling (PS) System is a sub-system of the PREMS. The PS collects representative liquid and gaseous samples for analysis and provides the information required for monitoring plant and equipment performance. This subsystem is designed to function during all plant operational modes under individual system requirement.

A detailed description of the Process Sampling system can be found in PSAR Chapter 9A, Section 9A.3.

Heating Ventilation and Air Conditioning Systems

The HVAC System consists of the following subsystems: RB HVAC, CB HVAC, RWB HVAC, TB HVAC, and Annex Building HVAC.

The HVAC system is designed to provide a controlled environment for personnel comfort and safety and equipment operation. Sufficient outside air is provided to meet the ventilation requirements for acceptable indoor air quality.

The HVAC design includes electric unit heaters which provide heat when required to maintain minimum space temperature. The HVAC systems are designed to minimize exposure to personnel during inspection and maintenance by locating equipment and instrumentation as far as practical from potential sources of contamination. A positive pressure in the HVAC systems is utilized to minimize the infiltration of outside air. The HVAC systems are designed to reduce the potential spread of airborne contamination by maintaining airflow from areas of lower potential for contamination to areas of greater potential for contamination.

The HVAC is not required to operate during a loss of normal AC power except to support equipment and areas provided with standby diesel power designated to support plant shutdown during a LOPP.

In off-normal operating modes HVAC performs the function of exhausting hydrogen gas build up in the safety-related battery rooms and maintains the RB habitability envelope.

For the off-normal condition of an extended loss of offsite power, the electrical space heaters can be powered by a diesel generator to maintain a minimum temperature in the power block area. See PSAR Chapter 9A Section 9A.5 for further information [R-6].

Communication Systems

The communications systems provide effective intraplant communications and effective plant-to-offsite communications during normal operation, maintenance, transients, fire, and accidents conditions including Loss of Offsite Power (LOOP).

The communication system allows security personnel to maintain continuous communication with personnel in staffed alarm stations, and offsite/onsite agencies. A detailed description of the Communication system can be found in PSAR Chapter 9A, Section 9A.9.

4.5.15 Fuel handling and storage

The fuel handling and storage systems, described in PSAR Chapter 9A Section 9A.1, are designed to address nuclear safety, security, and IAEA safeguards, as well as prevent inadvertent criticality and maintain shielding and cooling of irradiated fuel as necessary to meet operating and off-site dose constraints.

The design of fuel handling and storage systems is consistent with the DiD approach applied to the reactor core. It leverages design and safety features based on OPEX and proven BWR applications.

Criticality is prevented through use of geometrically safe configurations. The design of fuel storage systems applies physical means and processes to increase subcriticality margins.

Fuel handling and storage systems are designed to maintain adequate fuel cooling capabilities for irradiated fuel ensuring that the fuel cladding temperature limits and/or the coolant temperature limits, as defined for operational states and accident conditions, are not exceeded. See PSAR Chapter 3, Section 3.1.10.2 for further information.

New Fuel Storage and Handling

The Fuel Handling System and equipment is described in PSAR Chapter 9A, Sections 9A.1 and 9A.8.

New fuel is initially stored on the fuel handling floor prior to the first fueling and subsequent refueling outages. When needed, new fuel is moved into the fuel preparation machine using an overhead crane. The fuel preparation machine is on the side of the fuel pool wall and is used to lower fuel assemblies down to an elevation where the refueling machine is located. The refueling machine is used for fuel movement and servicing within the reactor core and storage racks. The mast grapple includes a redundant load path so that no single failure will result in a fuel drop. Additionally, the following interlocks ensure that fuel is safely handled:

- Fuel cannot be lifted unless permissive for all control rods inserted into reactor is present
- Fuel cannot be lifted without grapple hook and load engagement
- Fuel grapple vertical travel is limited to ensure fuel is shielded during transit

The refueling machine lifts the new fuel assemblies by the handle and places them in an indexed Fuel Pool rack location for storage. Storage racks are used for both new and spent fuel and are designed to prevent inadvertent criticality. The Fuel Pool is located in the RB and contains a large volume of light water used to store both new and spent fuel as well as refueling equipment.

Additional information one new fuel handling and storage can be found in PSAR Chapter 9A, section 9A.1.1, and Safeguards Annex Section 5.1.2.

Core Fuel Repositioning and Refueling

Refueling is performed during an outage. The basic activities associated with refueling the reactor are as follows:

- Disassemble the containment head and RPV head
- Remove RPV internals (e.g., dryer, separator)
- Offload of spent fuel

- Repositioning of in-core fuel assemblies
- Installation of new fuel
- Replace RPV internals and reassemble RPV head and containment head

The refueling machine is used for all movement of fuel during core fuel repositioning and refueling. Additional information on new fuel handling and storage can be found in PSAR Chapter 9A, Section 9A.1.2, Safeguards Annex Section 5.1.3 [R-6].

Spent Fuel Storage and Handling

Spent fuel is removed from the reactor and initially stored in the Fuel Pool until the decay heat reduces to the appropriate level. The spent fuel is then loaded into a dry canister in the pool. Using the RB crane, the loaded and sealed canister is lifted out of the Fuel Pool and moved to the refuel floor cask washdown pad for processing and then transfer to an interim storage facility.

The construction of the on-site dry storage facility will be part of a separate licence application; however, the general approach is briefly described here. The RB Polar Crane is used to lift a transfer cask to the refuel floor and subsequently to the truck bay. Once the dry, sealed, and decontaminated canister is placed in the transfer cask, the transfer cask is picked up by the hauler and moved to the dry storage installation, also called as Independent Spent Fuel Storage Installation (ISFSI).

Long-term management of DNNP's used nuclear fuel will be planned as part of the Nuclear Waste Management Organization's (NWMO) Adaptive Phased Management program.

Additional information on the storage and handling of spent fuel can be found in PSAR Chapter 9A, Section 9A.1 and PSAR Safeguards Annex Section 5.1.4 [R-6]. Section 4.11 of this Application document provides additional information on internal waste-related programs.

Fuel Pool Cooling and Cleanup

The Fuel Pool Cooling and Cleaning (FPC) system provides continuous cooling of the water in the fuel pool. This removes decay energy from spent fuel and provides replacement coolant inventory from a variety of sources, to ensure spent fuel is kept cool and submerged. In addition, the FPC system includes demineralization and particulate filtration to maintain coolant quality and to reduce general area dose.

The FPC System consists of two trains of equipment, each with a pump, demineralizer, and heat exchanger. Each set of components are placed in parallel to provide single train operation and cross connected to provide redundancy. A single train is sufficient to prevent bulk boiling in the Fuel Pool. If both trains are rendered inoperable, the Fuel Pool is sized to retain sufficient coverage of the fuel for at least 72 hours, and the FPC system can provide makeup capacity independently of the forced cooling trains.

Each demineralizer train of the FPC system contains a particulate filter and deep bed mixed resin demineralizer. The filter may be backwashed to the LWMS to remove accumulation and reduce dose. Additional information on the FPC can be found in PSAR Chapter 9A, Section 9A.1.3.

Criticality Protection and Radiation Monitoring

The fuel storage racks are designed to maintain the fuel at a physical distance and orientation to ensure subcriticality. Additional information on criticality protection can be found in PSAR Chapter 9A Section 9A.1.2 and Chapter 12 Section 12.3 [R-6].

Radiation monitoring is provided for the fuel transfer and storage area and associated ventilation paths. If high radiation is detected, these monitors will provide indication and alarms to the operator and close the associated ventilation path. The refueling machine, hoist, and grapple feature interlocks to prevent refueling errors and ensure radiation exposure to workers is kept ALARA. The fuel pool, reactor pool, and surrounding concrete walls are designed to ensure the area dose rate is maintained within specification. More information can be found in in PSAR Section 9A.1.2 and Chapter 12.

<u>Detection of Failed Fuel</u>

The radiation monitoring capabilities of the plant are designed to detect the presence and location of failed fuel in the reactor core.

Fuel failures are first detected via the plant's Offgas System. Nuclides are measured and the sum of these is reported as the total offgas release rate. Xenon-133 (Xe-133) is the primary isotope for determining the presence of fuel failures.

Fission product activity in the reactor coolant is also monitored (e.g., iodine, strontium, and cesium isotopes) to detect more significant failures.

See PSAR Chapter 9A, Section 9A.1.2.3.8, for further information.

4.5.16 Waste treatment and control

Liquid Waste Management System

Liquid radioactive waste is generated by cleanup activities in the RB and leakage from process streams. Liquids are collected by drains, drain headers and sumps and pumped to the LWMS for processing.

In the LWM, radioactive contaminants are removed, and the bulk of the liquid is purified and either returned to the CST or reprocessed.

The LWMS is divided into subsystems where liquid wastes are segregated and processed according to the impurity type and chemical content. Segregating liquid waste minimizes the total waste produced through efficient processing. Liquid wastes removed and concentrated in filter media, ion exchange resins, and other forms is further processed in the SWMS.

The system has significant holdup capacity in waste collection tanks and sample tanks that allows for reprocessing and minimizes effluent releases to the environment. See PSAR Chapter 11, Section 11.2 [R-6], for more information.

Off-Gas System

Gaseous radioactive waste is generated as a result of activation of the reactor coolant during normal reactor operation and the release of fission gases.

The Off-Gas System (OGS) minimizes and controls the release of radioactive material into the atmosphere by delaying and filtering the off-gas stream containing the radioactive isotopes of krypton, xenon, iodine, hydrogen, nitrogen, and oxygen sufficiently to achieve adequate reduction before discharge from the plant.

The OGS contains a recombiner, condenser, and activated carbon beds. The OGS provides sufficient holdup until the required fraction of the radionuclides has decayed, and the daughter products are retained by the charcoal to reduce the radioactivity in the effluent released from the facility to levels well below regulatory limits. See PSAR Chapter 11, Section 11.3 [R-6], for more information.

Solid Waste Management System

Solid waste is created through plant operations, decontamination activities and through processing liquid radwaste. It includes the filter backwash sludges, reverse osmosis concentrates, charcoal media, and bead resins. Contaminated solids such as High Efficiency Particulate Air (HEPA) and cartridge filters, rags, plastic, paper, clothing, tools, and equipment are also managed by the SWMS.

The SWMS is designed for controlling, collecting, handling, processing, packaging, and temporarily storing solid waste generated by the plant prior to offsite shipping. See PSAR Chapter 11, Section 11.4 [R-6], for more information.

Waste Minimization

The BWRX-300 LWMS is designed such that under normal conditions the need to release liquid effluent to the environment is minimized. LWMS tanks are designed with significant holdup capacity and filtration skids are capable of filtering water to a sufficient quality for return to plant processes.

The Off-gas System (OGS) minimizes and controls the release of radioactive material into the atmosphere by delaying and filtering the off-gas process steam.

The BWRX-300 design further minimizes wastes and contamination by:

- Providing atmospheric purging of the internal portion of air sampling skids as necessary.
- Providing the ability for liquid flushing of the internal portions of liquid sampling skids as necessary.
- Designing the interior portions of liquid and gaseous sampling chambers to minimize the plate out of radioactive materials.
- Designing sample extraction points such that that they minimize the potential for spillage and contamination of adjacent areas.
- Minimizing the amount of sample that needs to be extracted, consistent with laboratory and sensitivity requirements.

The BWRX-300 design facilitates decommissioning by providing equipment, where feasible, that reduces the need for decontamination during the use, removal, and disposal of the equipment or that is easier to decontaminate. See PSAR Chapter 11, Section 11.5 [R-6], for further information.

Disposal of Waste

There are various waste disposal paths that may be deployed under a future operating licence depending on the characterization of the waste. Note that none of these are required for the requested construction licence.

The waste disposal paths include but are not limited to:

- Solid radioactive waste shipped to a licenced off-site facility for incineration, decontamination, volume minimization, and/or storage
- Radioactive liquid chemicals are likely to be incinerated or solidified and stored at an OPG licensed facility
- Non radiological solid conventional waste will be monitored to confirm it is below all regulatory limits, and then shipped to public landfill or recycled
- Non radiological chemicals/liquid industrial waste will be monitored to confirm below all regulatory limits, and shipped to a hazardous waste receiving company for incineration or disposal in hazardous landfill

4.5.17 Applicable OPG Documents

The OPG governance documents for the Physical Design SCA, which supports the licensing basis, are included in Table 4.5-2 below:

Table 4.5-2: Management System Document for the Physical Design SCA

Document	Title
N-PROG-MP-0009	Design Management
N-STD-MP-0009	Contractor/Owner Engineering Interface and Oversight

4.6 Fitness for Service

The Fitness for Service SCA covers the activities that impact the physical condition of SSCs to ensure that they remain effective over time. This area includes programs that ensure all equipment is available to perform its intended design function when called upon to do so.

The main objective of the fitness for service operational program is to ensure that SSCs identified as important to safety are able to meet their intended design and safety functions when called upon to do so. This is accomplished by implementation of programs that establish SSC requirements and practices in the following areas:

- Reliability;
- Maintenance;
- Aging Management;
- Chemistry Control; and
- Periodic and In-Service Inspection.

For existing reactor facilities, OPG has effective and robust programs in place. These programs will be adapted to meet the requirements of the BWRX-300 design as part of the staged approach in developing and implementing operational programs to satisfy regulatory requirements.

The details for the programs covering these areas will be addressed as part of a future operating licence application.

4.7 Radiation Protection

During the construction phase for the DNNP, workers and the public will not be at risk of receiving radioactive doses exceeding public dose limits as a result of activities to be conducted as part of this Application. Work done with construction-related tools containing radioactive nuclear substances will be performed and controlled under the authority of separate CNSC Nuclear Substances and Radiation Devices licence(s).

The DNNP site is located within proximity of the DWMF and DNGS, and thus there could be very low-level exposure to ionizing radiation above background levels. Any resulting exposure to workers will be a small fraction of the regulatory limits for members of the public and will be governed by the DNGS RP program and controls.

The overriding objective of the future Radiation Protection (RP) Program for the DNNP BWRX-300 will be the control of occupational and public exposure to radiation during operations. For the purposes of controlling doses to workers, this program has the following implementing objectives:

- Keeping individual doses below regulatory limits.
- Avoiding unplanned exposures.
- Keeping individual risk from lifetime radiation exposure to an acceptable level.
- Keeping collective doses ALARA, social and economic factors taken into account.

In terms of protecting the public, the RP Program will prevent the uncontrolled release of contamination or radioactive materials from the site by controls and monitoring of people and materials.

The RP Program will include a set of action levels (ALs) to provide an alert before a more significant event occurs.

The future radiation protection program for the DNNP BWRX-300 will cover the following:

- Organization and administration for radiation protection
- Radiation protection training and qualification
- Classification of radiation zones and local rules
- Radiation exposure and dose control
- Worker dose
- Radiation protection equipment and instrumentation
- Contamination control
- Planning for unusual situations
- Radiation protection program oversight
- Dose to the public
- Management control over work practices

Sections 4.5.1, 4.5.3 and 4.5.4 of this Application document provide information on how radiation protection is considered in physical design. Further details are provided in PSAR Chapter 12 [R-6].

The following sections provide an overview of some of the key elements that are taken into account in the design and future operation of the DNNP BWRX-300.

Application of ALARA

The radiation protection program implements a series of standards and procedures for the conduct of activities within a nuclear station intended to keep radiation exposure to workers ALARA.

The ALARA strategy identifies initiatives, actions and programs that support achieving these objectives. The strategy applies whether the unit is operating, in outage or in safe storage. Equally, the strategy applies to all staff, contractors and visitors. The strategy is regularly updated to reflect the results of benchmarking, corrective action plans and industry best practices.

Collective dose performance targets are established taking into account planned maintenance outage scope, past performance, and anticipated dose savings from planned initiatives and application of ALARA techniques. As work is planned in more detail, collective dose projections are reviewed, and actions are taken to ensure dose is ALARA. Actual performance against targets is reviewed and corrective actions are taken where expectations are not met.

The application of ALARA begins with reactor facility design. ALARA in design and construction policies are described in PSAR Chapter 12, Section 12.1 [R-6], while radiation protection design features are described in PSAR Section 12.3. A key consideration is shielding design, which is addressed in PSAR Section 12.4. The minimization of radiation exposure is addressed by PSAR Section 12.6. Design considerations to reduce/limit radiation accumulation on pipe surfaces and in radioactive drains, sumps and resin systems is described in PSAR Section 12.3.

4.8 Conventional Health and Safety

The Conventional Health and Safety SCA ensures implementation and oversight of a program to manage workplace health and safety, the non-radiological hazards within the workplace, and to protect personnel by application of a rigorous and effective safety management system.

4.8.1 General considerations

OPG has a management system which is regularly maintained and continually improved upon as captured in OPG-PROG-0005 [R-60]. The Health and Safety Managed System (HSMS) implements the requirements of OPG-POL-0001, *Employee Health and Safety Policy* [R-94].

OPG and contractors will manage the health and safety of all workers associated with the DNNP as per the *DNNP Health and Safety Plan* [R-69], which establishes the framework for the management of worker health and safety in accordance with defined health and safety values, goals, objectives, and expectations of OPG.

OPG has an existing procedure for contractor safety management that establishes the requirements for managing contracted work to ensure the work is performed safely. This procedure applies to all OPG workers and contract workers performing work at OPG workplaces. The foundation of OPG's HSMS is the OPG-POL-0001, which describes the approach and commitments to conventional health and safety for the organization, as well as the requirements and accountabilities of all employees and outlines expectations for contractors.

Within the policy it states that OPG is committed to the prevention of workplace injuries and ill health, and to continuous improvement of its employee health and safety performance. Additionally, the policy requires that OPG meet or exceed all applicable health and safety legislative requirements as well as other associated health and safety standards to which OPG subscribes. OPG requires that its contractors maintain a level of safety equivalent to that of OPG employees. OPG is committed to preventing workplace injuries and to continuously improve employee health and safety performance.

The HSMS is structured in accordance with ISO 45001, *Occupational Health and Safety Management Systems* [R-28]. The Program governs the design and implements the requirements and expectations of OPG-POL-0001 [R-94].

OPG's HSMS establishes the process requirements that are implemented and maintained to ensure that health and safety hazards and risks to workers are being mitigated. The HSMS scope includes:

- Occupational conditions and factors that could affect the health and safety of workers, in all workplaces or from work-related activities under the control of OPG.
- Non-occupational health-related conditions and factors that could affect the health of OPG workers where it impacts achievement of OPG's business objectives.
- Contractor health and safety.

HSMS oversight includes an audit and assessment program to ensure that all requirements of the OPG-POL-0001 and the HSMS are being properly implemented and maintained throughout the organization.

Within the HSMS, compliance obligations for both federal and provincial legislation and municipal bylaws are captured and evaluated. The identified legislative requirements are a key input into the development, implementation and enforcement of operational level procedures and guidance documents.

These operational procedures capture the legislative requirements and outline how they will be demonstrated and complied with in the field, while also establishing roles and responsibilities of managers, supervisors, and workers and the expectations as it relates to application of the procedure and requirements for health and safety related training.

OPG also has in place procedures for the classification, reporting, and recording of worker and contractor safety incidents and regulatory events, as well as the investigation of safety incidents and the implementation of corrective actions. These procedures apply to all safety incidents affecting OPG workers and contractors as a result of OPG operations and facilities.

The HSMS monitors, measures, analyses, and evaluates performance against objectives established in business planning through regular monthly, quarterly, and annual reporting requirements. These reports capture OPG employee safety incidents, contractor safety incidents, regulatory events and non-occupational worker sick leave.

As outlined in OPG-PROG-0005 [R-60], OPG continually improves the suitability, adequacy and effectiveness of the HSMS by enhancing health and safety performance through periodic inspections/audits/assessments, revising training material, improving communications, implementation of corrective actions, and implementation of annual Management Review outputs.

OPG is committed to involving its worker representative groups to meet its health and safety objectives. Worker representative involvement is built into elements of the HSMS throughout the organization:

- Tripartite Advisory Committee consisting of senior executives of OPG and the worker representative groups who meet regularly to review health and safety issues of interest throughout OPG, including non-occupational health issues, and recommend strategies.
- Joint Health and Safety Working Committee and its sub-committees, Corporate Safety Rule Advisory Group and Corporate Code Advisory Group, where senior staff from Corporate Environment, Health and Safety, the worker representative groups meet regularly to discuss health and safety issues, including non-occupational health issues, and recommend strategies to address these issues.
- Worker involvement occurs at the business level at tripartite Joint Health and Safety Committees for each workplace as per requirements in Ontario's Occupational Health and Safety Act (OHSA) and agreements with the unions, where regular workplace inspections are performed, and local issues are discussed and dispositioned with corrective actions.
- Locally, workers participate in identifying hazards and safe work planning through, for example, safety meetings, pre-job briefs, two-minute job-site drills and tailboard meetings.

Furthermore, the HSMS interfaces with several other OPG programs to strengthen enterprise resiliency and collaboration.

4.8.2 Applicable OPG Documents

The OPG governance documents for the Conventional Health and Safety SCA, which supports the licensing basis, are included in Table 4.8-1 below:

Table 4.8-1: Management System Document for the Conventional Health and Safety SCA

Document	Title
OPG-PROG-0005	Environment Health and Safety Managed Systems

4.9 Environmental Protection

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

4.9.1 General Considerations

OPG's OPG-POL-0021, *Environmental Policy* [R-95] provides direction related to OPG's environmental performance and environmental management and requires an ISO 14001-certified [R-96] Environmental Management System (EMS) to be in place. OPG's EMS establishes a robust environmental protection program that meets or exceeds all applicable regulatory requirements (further discussed in Section 4.9.2).

As part of its environmental protection program, OPG maintains an effluent monitoring program governed by N-STD-OP-0031, *Monitoring of Nuclear and Hazardous Substances in Effluents* [R-97], an environmental monitoring program for the entire Darlington site governed by N-PROC-OP-0025, *Management of the Environmental Monitoring Programs* [R-98], and conducts routine environmental risk assessments that meet the requirements of CSA N288.6-12, *Environmental Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills* [R-10].

The DNNP will be executed in a manner which complies with the requirements of OPG-POL-0021 [R-95] and EMS. In addition to OPG's existing environmental protection program, the DNNP will also have in place project-specific environmental protection measures that address potential environmental effects associated with DNNP construction and commissioning activities as identified by the EIS [R-4 and R-29]. This comprises the NK054-PLAN-07730-00022, *DNNP - Environmental Management and Protection Plan (EMPP)* [R-80] as well as the NK054-PLAN-007730-00014, *Environmental Monitoring and Environmental Assessment Follow-up (EMEAF) for DNNP* [R-128] and its associated monitoring plan/methodology reports (collectively referred to as the DNNP EA follow-up monitoring program). The DNNP EMPP and the EA follow-up monitoring program were initially developed for site preparation. Prior to start of construction, the DNNP

EMPP and the environmental monitoring plan/methodology reports for the EA follow-up program will be updated to address construction activities, as required.

During the construction phase of DNNP, management and monitoring of releases from DNNP construction activities are addressed by the DNNP EMPP and EA follow-up monitoring program as well as compliance with the relevant environmental permits and ECAs. The NK38-PLAN-03480-10001, *Darlington Effluent Monitoring Plan* program [R-102] will be updated prior to DNNP commissioning to encompass the relevant releases from the DNNP facility. The DNNP will not contribute to radiological releases during site preparation, construction, and fuel-out commissioning. OPG will update existing radiological environmental programs for the DN site to encompass the DNNP and implement the measures to address radiological releases one year prior to fuel-in commissioning as part of DNNP commitment [R-2].

4.9.2 Environmental Management System

OPG-PROG-0005 [R-60] documents OPG's ISO 14001 [R-96] certified EMS and provides the structure and processes to implement OPG-POL-0021 [R-95]. OPG's EMS ensures environmental risks associated with OPG's activities are assessed, and that activities are conducted in a way that adverse environmental effects are prevented or mitigated.

DNNP activities will be executed in a manner that conforms to the requirements of OPG-POL-0021 and EMS. OPG will provide ongoing oversight through its EMS. The other contract parties will have defined and implemented their own EMS compliant with applicable current standards or implement an alternative management system as agreed to by the contract partners.

During the construction phase, management and monitoring of environmental impacts and emissions will be addressed via the implementation of the *DNNP EMPP* [R-80] and DNNP EA follow-up monitoring program respectively. Both the DNNP EMPP and the DNNP EA follow-up monitoring program will be implemented through the existing OPG EMS.

Within OPG's EMS framework, the DNNP EA follow-up monitoring program will be implemented as project-specific supplementary studies under the existing NK38-

MAN-03443-10002, *Darlington Environmental Monitoring Program* [R-129]. The DNNP EA follow-up monitoring program is further discussed in in Section 4.9.3 of this Application document.

DNNP Environmental Management and Protection Plan

Control measures that eliminate, manage, reduce, or mitigate risks associated with DNNP construction are implemented via the *DNNP EMPP* [R-80]. The DNNP EMPP requires the constructor and all sub-contractors to meet the requirements of the DNNP EMPP and requirements or conditions stipulated in respective federal, provincial and municipal permits, approvals or authorizations. OPG will ensure the implementation of the DNNP EMPP throughout construction and commissioning.

The performance of the control and mitigation measures is verified by DNNP EA follow-up monitoring program. Adaptive management will be applied throughout the construction phase of the DNNP. The adaptive management process is described in the DNNP EMEAF [R-128] and allows continuous improvement of environmental management practices by monitoring their outcomes and applying the knowledge that has been gained to subsequent actions.

OPG's approach for the management of spills, excess soils, waste and hazardous substances, land assessment and remediation management, and management of impacts to fish are addressed by the DNNP EMPP. The management of soils, waste and hazardous substances are discussed in Section 4.11 of this Application document. The management of spills, land assessment and remediation management are summarized below.

Spills Management

The *DNNP EMPP* [R-80] requires the constructor to develop and implement spill management measures to address spill prevention, response mechanisms, and reporting requirements. The constructor will establish a spill prevention and contingency plan for DNNP construction activities to demonstrate their project specific commitment to spill prevention, preparedness, response, reporting, and clean-up in accordance with current standards and regulations.

Land Assessment and Remediation Management

OPG has an existing process in place for land assessment and remediation management at its nuclear generating stations. The procedure outlines directions and accountabilities for identifying, assessing and managing contaminated lands within OPG as well as for establishing and managing OPG's groundwater monitoring program. The contaminated lands management program ensures suspected or discovered contamination resulting from past or on-going operations at OPG properties is assessed, monitored, and/or remediated such that adverse effects are mitigated.

Management of Effects on Fish

Potential effects to fish populations were assessed as part of the DNNP EIS [R-4 and R-29] and OPG has made a number of commitments to mitigate and monitor for adverse effects of the DNNP on fish.

In-water work is required for shoreline protection and the construction of the intake and discharge structures. Prior to in-water construction activities, a Department of Fisheries and Oceans Canada (DFO) Fisheries Authorization will be obtained to address any potential impacts to fish and fish habitat during in-water construction. The future application for a DFO authorization will outline avoidance and mitigation measures, as well as compensatory measures where required.

The DNNP construction activities will be conducted in accordance with the conditions outlined in the DFO authorization obtained for this phase of the project, particularly when undertaking activity that may result in the death of fish or the harmful alteration, disruption or destruction of fish habitat. Monitoring of adverse effects on fish is discussed in Section 4.9.3.

Licensed Release Limits and Action Levels and Monitoring of Radioactivity

Given that DNNP will not produce any radiological releases during construction phase activities, DNNP-specific radiological release limits and ALs will not apply until the operations phase. Such limits and ALs will be developed for the DNNP prior to the operations phase, in accordance with the applicable processes to establish such values and in accordance with the applicable standards and regulatory documents in effect at that time.

OPG's existing Environmental Monitoring Program (EMP) for the Darlington site [R-129] will continue monitoring throughout the execution of DNNP construction activities. Given the future operation of DNNP will contribute radiological emissions in addition to those generated by the existing Darlington site facilities, OPG will review and revise, as appropriate, the existing EMP for the Darlington site to encompass radiological releases from the DNNP. During DNNP operations, licensed release limits will be reviewed at the minimum frequency required by the DNNP's Licence Conditions Handbook.

4.9.3 Assessment and Monitoring

OPG has an established and effective environmental protection program for the Darlington site that meets or exceeds all applicable regulatory requirements. This includes a variety of monitoring programs (*Darlington Effluent Monitoring* program [R-102], EMP [R-129], and the DNNP EA follow-up monitoring program), each of which has its own scope and objectives which include monitoring to meet licence and regulatory requirements, to address issues of relevance to the EA, as well as overall monitoring for the operational facilities on site. Operational monitoring is not applicable for the DNNP during its construction phase; however, it is necessary to confirm performance goals for the existing facilities are being achieved.

The objectives of these monitoring programs are complementary in that collectively, they will provide data that will be useful in planning and implementing the long-term environmental management requirements of the EMS. To the extent practical, the environmental monitoring activities will be performed on a "one-site" basis such that data can be shared to meet the collective requirements of all licensed activities on the Darlington Site including the DNNP, DNGS, the TRF and the DWMF. This monitoring will consider the cumulative effects of all activities/operations within Darlington site on the local environment.

The DNNP's EA follow-up monitoring program and the Darlington EMP [R-129] are discussed below, and effluent monitoring is discussed in Section 4.9.4 of this Application document.

DNNP EA Follow-Up Monitoring Program and Environmental Monitoring

Environmental monitoring during DNNP construction is established through the DNNP EA follow-up monitoring program which will be undertaken as a supplementary study under the Darlington EMP. Through the EA and JRP process, OPG made commitments to design and implement EA follow-up programs to verify the predictions made in the EIS, confirm the effectiveness of mitigation measures, and provide assurance that the applicable guidelines or regulatory criteria are being met through each phase of the project. The EA follow-up monitoring program addresses the pathways, contaminants and parameters identified to be relevant to the protection of the environment and health and safety of persons as identified through the EA and JRP process and aligns with the requirements of REGDOC-2.9.1, *Environmental Principles, Assessments and Protection Measures* [R-99]. Its scope was further refined through a consultative process involving stakeholders and Indigenous communities. In addition to addressing the EA follow-up monitoring requirements for the DNNP, some of the monitoring activities may also fulfill other requirements, such as monitoring required by the CNSC licences, applicable federal, provincial or municipal laws or permits and other monitoring as part of due diligence by OPG for the DNNP and overall Darlington site.

OPG's EA follow-up monitoring plan [R-128] documents the overall objectives of the EA-follow-up monitoring program, the implementation framework, the environmental monitoring components, and discusses high-level scoping for the program. The environmental components (and sub-components) are consistent with those identified in the EIS and DNNP commitments which include: the atmospheric environment, surface water environment, aquatic environment, terrestrial environment, geological and hydrogeological environment, land use, traffic and transportation environment, and health of human and non-human biota.

Radiation and radioactivity was not identified as an environmental component requiring separate monitoring as part of the DNNP EA follow-up monitoring program. OPG's commitments related to Radiation and Radioactivity are addressed by DNNP [R-2]. The Darlington EMP [R-129] will continue to monitor radiation and radioactivity at the DN site during construction of the DNNP and OPG will implement any necessary changes to the Darlington EMP to encompass radiological environmental monitoring for the DNNP prior to DNNP's operational phase.

Detailed scopes of work for each environmental component of the EA follow-up monitoring program are documented within the respective monitoring plan/methodology report. Each monitoring plan/methodology report provides the detailed description of the objectives and rationale for the monitoring activities. Where relevant, the document identifies the environmental media to be sampled, the parameters to be monitored, assessment criteria, sampling and analytical methods, sampling and analytical frequency, sampling locations, requirements for the calibration of equipment, and analytical detection limits.

OPG prepared and submitted the EA follow-up monitoring plan [R-128] and the associated monitoring plan/methodology reports to the CNSC prior to the commencement of licensed site preparation activities [R-130] [R-131] [R-132] [R-133] [R-134]. OPG began implementing relevant EA follow-up monitoring activities prior to the start of licensed site preparation activities and will continue to maintain the program throughout the construction and operations phases of the DNNP in accordance with the EA Follow-Up Monitoring Plan. Revision 1 of the EA follow-up monitoring plan [R-128] is being submitted together with this Application. The scope and nature of the EA follow-up monitoring activities will be reviewed and adjusted as part of the adaptive management process to incorporate subsequent phases of the project (i.e. construction), changes to site conditions, and to address monitoring results as needed. OPG's adaptive management process for the EA follow-up monitoring program is described in the Plan.

All EA follow-up monitoring activities will be subject to the requirements of N-PROC-OP-0025 [R-98], given DNNP EA follow-up monitoring activities would be conducted as supplementary studies under the Darlington EMP [R-129]. N-PROC-OP-0025 is aligned with CSA N288.4, *Environmental monitoring programs at class I nuclear facilities and uranium mines and mills* [R-100]. Groundwater monitoring will be compliant with CSA N288.7, *Groundwater protection programs at class I nuclear facilities and uranium mines and mills* [R-14].

Prior to the operation of DNNP, OPG will review and update the existing EMP, as needed, to encompass the DNNP. As part of the update process, OPG will re-examine the objectives of each EA follow-up monitoring activity. Once the objective of each EA follow-up monitoring activity is achieved, a determination will be made, based on the results of each monitoring activity, to either close off the EA follow-up monitoring activity or integrate the monitoring into the routine EMP.

Environmental Risk Assessment (ERA)

As part of the EA process, OPG completed an Ecological Risk Assessment (EcoRA) and assessment of effects on human health for the DNNP which considered the physical, mental and social well-being of workers and members of the public. The assessments concluded that no residual adverse effects on human health or non-human biota are expected as a result of DNNP activities (including construction, operation, and decommissioning). This conclusion was confirmed by OPG's comprehensive review of the EIS following the selection of the BWRX-300 reactor technology [R-4]. The DNNP EA follow-up monitoring program is risk-informed and has been designed to address the potential environmental effects associated with the DNNP.

The D-REP-07701-00001, *2020 Environmental Risk Assessment for the DNNP Site* [R-41] forms the basis for the EMP and the effluent monitoring program for the site. The ERA consists of a Human Health Risk Assessment and an EcoRA for both radiological and non-radiological parameters as well as physical stressors. OPG routinely updates the ERA to reflect current conditions across the site in accordance with REGDOC-3.1.1, *Reporting Requirements for Nuclear Power Plants* [R-72]. A CSA N288.6-compliant [R-10] ERA was produced for the Darlington site most recently in 2020, which concluded that the site (which encompasses the DNGS, TRF, and DWMF) is being operated in a manner that is protective of human and ecological receptors residing in the surrounding area.

Monitoring of Effects on Fish

OPG has commitments to ensure impacts to fish are minimized. Monitoring of effects from DNNP construction and operations (e.g., including thermal effects and impingement and entrainment), will be addressed by the DNNP EA follow-up monitoring program. Impingement and entrainment monitoring program design will consider guidance from CSA N288.9, *Guideline for design of fish impingement and entrainment programs at nuclear facilities* [R-101] and will be carried out in accordance with the conditions outlined in a future DFO Fisheries Authorization.

4.9.4 Effluent and Emissions Control

Existing OPG governance for monitoring nuclear and hazardous substances in airborne and waterborne effluents (N-STD-OP-0031) [R-97] applies to all of OPG's nuclear facilities and governs OPG's *Darlington Effluent Monitoring* program [R-102]. N-STD-OP-0031 complies with CSA N288.5, *Effluent monitoring programs at Class I nuclear facilities and uranium mines and mills* [R-103] and CSA N288.8, *Establishing and implementing action levels for releases to the environment from nuclear facilities* [R-104].

N-STD-OP-0031 will be applied through all phases of the DNNP including for construction activities.

During the construction phase of DNNP (prior to fuel-out commissioning), effluents and emissions associated with the project would be limited to non-radiological discharges associated with storm water runoff, dewatering activities, blasting, and airborne emissions from construction equipment. Management and monitoring of such releases will be addressed by the *DNNP EMPP* [R-80] and EA follow-up monitoring program as well as the relevant ECA.

Prior to the start of construction, OPG will update relevant ECAs that currently exist for the Darlington site to establish compliance limits for non-radiological hazards relevant to DNNP construction. The updates will be developed and implemented in accordance with the applicable regulations. Compliance with the ECA limits is ensured through the implementation of the DNNP EMPP measures and completion of any necessary monitoring as per the requirements of the ECAs. The overall environmental impacts of the DNNP construction activities are confirmed via the DNNP EA follow up monitoring program.

OPG's existing DN EMP and Darlington effluent monitoring program will continue to be implemented across the site through all phases of the DNNP. The existing effluent monitoring program includes monitoring of routine releases of nuclear and hazardous substances from site operating facilities. The monitoring and control measures OPG has established for the DNGS ensure that effluent discharges from the plant comply with all regulatory requirements. OPG routinely conducts ERAs for the Darlington site evaluating the risk to relevant human and ecological receptors based on the actual activities that occurred on the DN site. Together, the Darlington effluent monitoring program, environmental monitoring program and the routine ERA verify that releases to air, surface water, groundwater and soils from normal operation and waste management activities meet all regulatory, federal and provincial guidelines.

The DNNP may release non-radiological process effluents during fuel-out commissioning. OPG will update the *Darlington Effluent Monitoring* program [R-102] for the site to encompass releases from DNNP in two steps: first to encompass DNNP's non-radiological effluents prior to fuel-out commissioning and then to encompass its radiological effluents prior to facility operations. A detailed assessment of predicted effluent releases from the DNNP was completed as part of the comprehensive review of the EIS [R-4].

During the operational phase of the DNNP, OPG will periodically monitor data on cooling water discharge temperature and plume characteristics. The data will be used to inform the interpretation of effects on fish habitat, and the susceptibility of Valued Ecosystem Component (VEC) species to verify the EIS conclusions. Adaptive management will be implemented to address changes to the environment associated with the aquatic ecosystem over time.

<u>Sampling and Analytical Methods and Calibration of Equipment</u>

The *Darlington Effluent Monitoring Plan* [R-102] describes procedures for sample collection, preservation and compositing, representative sampling, and use of accredited laboratories as required by CSA N288.5 [R-103]. Sampling systems are subject to QA/QC testing and regular maintenance programs which ensure samplers and instrumentation are calibrated and tested in accordance with OPG's N-STD-OP-0031 [R-97]. Sampling methods and analytical procedures will be in accordance with the requirements of CSA N288.5 as described in N-STD-OP-0031.

Training and QA/QC

OPG will ensure all specialist staff and contractors supporting the implementation of the effluent monitoring program have the appropriate training and required certifications per the requirements of CSA N288.5 and OPG's governance, as summarized in N-STD-OP-0031 [R-97].

OPG's overarching standard for monitoring nuclear and hazardous substances in effluents, N-STD-OP-0031, requires all aspects of the effluent monitoring program to have appropriate QA and QC in accordance with CSA N288.5. This QA program applies both to OPG employees and contractors involved in program implementation and requires the establishment of a QA program to verify that the effluent monitoring program is adequate and accurate and identify any deficiencies requiring corrective actions.

Audit and Review Process

Core elements of the effluent monitoring program include sampling and analytical procedures, data analysis, interpretation, QA/QC and reporting, review and auditing. In accordance with the requirements of CSA N288.5 [R-103], audits are completed at least once in five years or more frequently if conditions change. The scope of any individual audit may be limited to certain aspects of the program. The *Darlington Effluent Monitoring Plan* [R-102] discusses the chemistry self assessment program, environment effluent monitoring program self assessment and nuclear oversight audits.

Records of Releases and Information to the Authorities and the Public

Although the DNNP will not contribute to radiological releases prior to its operations phase, DNNP will adhere to OPG's existing processes for records and reporting.

Designed Control Measures

The design of the DNNP is being optimized to minimize or eliminate the release of nuclear and hazardous substances incorporating Best Available Technologies Economically Achievable and ALARA principles into the design process. Plant features which minimize environmental impact and ensure releases are monitored and controlled are discussed in detail in PSAR Chapter 20 [R-6].

4.9.5 Applicable OPG Documents

The OPG governance documents for the Safety Analysis SCA, which support the licensing basis, are included in Table 4.9-1 below:

Table 4.9-1: Management System Documents for the Environmental Protection SCA

Document	Title
OPG-POL-0021	Environmental Policy
OPG-PROG-0005	Environment Health and Safety Managed Systems
N-STD-OP-0031	Monitoring of Nuclear and Hazardous Substances in Effluents
N-PROC-OP-0025	Management of the Environmental Monitoring Programs

4.10 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. This area also includes any results of participation in exercises.

4.10.1 General Consideration

OPG has an effective emergency management and fire protection program that meets or exceeds all applicable regulatory requirements and related objectives. Emergency management measures and fire protection response capabilities are in place at OPG to prevent and mitigate the effects of nuclear and hazardous substances releases, both on-site and off-site, and fire hazards in order to protect workers, the public and the environment.

The nuclear Emergency Management program is documented in N-PROG-RA-0001, *Consolidated Nuclear Emergency Plan (CNEP)* [R-107].

This plan describes concepts, structures, roles, and processes to implement and maintain an effective OPG response in the unlikely event of a nuclear emergency that could endanger on-site staff, the public, or the environment. The CNEP provides a framework for interaction with external authorities and defines OPG commitments under the Provincial Nuclear Emergency Response Plan (PNERP). The plan was predominantly conceived to deal with releases of radioactive materials from fixed facilities, but infrastructures defined within may be used in planning and response to virtually all emergencies. Given the proximity of the DNGS and the DWMF to the area where DNNP construction activities will occur, OPG has taken into consideration potential risks that may arise in the event of an accident or malfunctions at these facilities.

A MOU has been established with the Province of Ontario to revise the PNERP to include DNNP and issue a revised Darlington Implementing Plan or separate Implementing Plan for DNNP. OPG's existing MOU with Durham Region references the process if the PNERP is amended in a way that impacts the Durham Nuclear Program.

The NK054-PLAN-01210-00002, *DNNP Nuclear Emergency Preparedness Plan* [R-20] describes the protocols to be implemented on the DNNP site in the unlikely event of a nuclear emergency from the DNGS or other hazards that may impact the entire DN site requiring activation of the Emergency Response Organization . The plan documents the concepts, roles and resources required to implement and maintain emergency response on the DNNP site to protect employees, visitors and contractors. The document describes the different phases of the project—Site Preparation, Construction, Commissioning, and Operations—and how the program will transition from construction site to commissioning once fuel has been placed into the reactor.

During construction and fuel-out commissioning, there is no potential for a radiological emergency from the DNNP site. The emergency preparedness and emergency management program for DNNP will adhere to the requirements of REGDOC-2.3.1 [R-54].

The *DNNP Health and Safety Plan* [R-69] establishes the framework for the management of worker H&S and includes the emergency response and fire protection controls. OPG will use this plan for the construction phase to guide the contractors involved in the construction activities, and to prepare their emergency response and fire protection plans. See section 4.8 of this Application that describes the conventional health and safety program.

OPG construction oversight will ensure that fire protection controls implemented by contractors are fully compliant with required regulations such as Ontario OHSA, National Building Code of Canada, and National Fire Code of Canada. The fire protection controls and response will be detailed in the constructor's Site-Specific Safety Plan (SSSP), which will be accepted by OPG. Periodic surveillance by OPG Construction Oversight will provide checks of compliance to the SSSP.

CSA N293, *Fire protection for nuclear power plants (application to small modular reactor)* [R-88] requirements for a fire protection program largely overlap the conventional fire protection, especially in the context of construction and commissioning phases. Therefore, the fire protection controls and response detailed in the constructor's SSSP will serve as the preliminary fire protection program. This will transition to a nuclear fire protection program that will address the needs of the BWRX-300 technology that will be compliant with CSA N293. This program will be in effect at the point of operations turnover.

4.10.2 Emergency Preparedness and Response

The *DNNP Nuclear Emergency Preparedness Plan* [R-20] receives its authority from the N-PROG-RA-0001 [R-107] and provides a written basis to document the concepts, roles and resources required to implement and maintain emergency response on the DNNP site to protect employees, visitors and contractors in the unlikely event of a nuclear emergency or event impacting the DN site. Section 3.0 of this Application describes the DNNP site-specific natural and human-induced external hazards. The document also describes the different phases of the project and how the program will transition from construction site to commissioning once fuel has been placed into the reactor.

As part of the OPG's Emergency Preparedness drills and exercises procedure, emergency and site evacuation drills may be held at the DNNP site or require all Darlington Controlled Area personnel to participate. In 2021, a number of DNNP contract partners participated in the annual DNGS full site assembly and accounting drill with early dismissal [R-110]. The DNNP contract partners were conducting early site preparation geotechnical field work on site the day of the drill. This allowed the opportunity for OPG and contract partners to effectively test the unique nuclear assembly and accounting protocols as identified in the contract partners SSSP including station emergency tone notifications and follow-up.

Since the PRSL application renewal, OPG has made available to off-site planning authorities a revised Darlington Site Evacuation Time Estimate (ETE) using the 2016 National Census Data with per decade population projections out to 2088, as well as current and forecasted infrastructure.

The estimate provides off-site emergency planners with projections on how long it may take for sectors of the Detailed Planning Zone to evacuate, if required. Variables such as time of day, day of week, road restrictions, special event assemblies and weather were assessed as to how those factors may impact the evacuation duration. OPG will issue an updated Darlington Site ETE based on 2021 national census data and will subsequently share with external emergency planning stakeholders.

The proximity to the reactor facility of airport, railways, roads and emergency services is described in PSAR Chapter 2, Section 2.3 [R-6].

PSAR Chapter 19 provides further information on the emergency preparedness program for the BWRX-300.

4.10.3 Fire Protection Program

During construction, prior to turnover to Operations, fire protection controls and response, which comprises the preliminary fire protection program, is primarily the responsibility of the constructor as detailed in their SSSP. This plan is accepted by OPG and periodic surveillance by OPG construction oversight will provide checks of compliance to the SSSP as described under Section 4.8 of this Application document.

Construction and fuel-out commissioning are conventional in nature and the fire protection program implemented by the constructor will be fully compliant with the Ontario OHSA. OPG oversight will ensure that the constructor has policies, procedures and programs in place in compliance to applicable codes, standards and regulations as described under Section 4.8 of this Application document. This will serve as the preliminary fire protection program for the facility for the construction phase.

As the project advances through commissioning, the constructor is responsible for advancing the preliminary fire protection program with OPG providing continuing oversight and accountability. Once the handover to operations occurs, this preliminary fire protection program will transition to a nuclear N-PROG-RA-0012, *Fire Protection* [R-108] program that is applicable for the DNNP facility.

Fire Response

Fire response in all phases of construction will primarily be with Clarington Emergency Fire Services (CEFS). The DNNP site is within the DNGS controlled area for which a MOU is in place with the CEFS to respond in case of emergencies. OPG oversight will ensure that fire response strategy stated in the constructor's fire protection program complies with the Ontario OHSA and ensures effectively and timely communications with CEFS.

Fire Protection Assessments

OPG will review the fire protection assessments of the physical design against the requirements of CSA N293 to ensure that the physical design of the completed facility meets the fire safety goals, objectives and criteria as defined by CSA N293 [R-88].

4.10.4 Applicable OPG Documents

The OPG governance documents for the Emergency Management and Fire Protection SCA, which supports the licensing basis, are included in Table 4.10-1 below:

Table 4.10-1: Management System Document for the Emergency Management and Fire Protection SCA

Document	Title
N-PROG-RA-0001	Consolidated Nuclear Emergency Plan
NK054-PLAN-01210-00002	DNNP Nuclear Emergency Preparedness Plan

4.11 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.

Waste management includes both nuclear and hazardous substances that are used or produced in the course of carrying on a licensed activity and that may pose a risk to the environment or the health and safety of persons. This area also covers the planning for decommissioning.

4.11.1 General Considerations

Similar to the DNNP PRSL [R-1], the activities to be licensed under the DNNP LTC will not involve the generation of radioactive wastes. The handling of radioactive wastes is not part of this Application. However information is provided for how these wastes will be managed through the lifecycle of the BWRX-300 NGS to be located at the DNNP site.

For the PRSL renewal application [R-16], a review of the licence basis materials was conducted in the areas of nuclear waste management. The review did not identify any changes with respect to baseline information or codes, standards and practices.

OPG has a mature and effective waste management program that meets all applicable regulatory requirements and related objectives. These programs are fully developed, implemented and audited to control and minimize the volume of nuclear and hazardous wastes generated by licensed activities.

OPG's Standard OPG-STD-0156, *Management of Waste and Other Environmentally Regulated Materials* [R-111], ensures that low and intermediate level radioactive waste and non-radioactive hazardous substances and waste are handled, processed, shipped and stored in accordance with the applicable federal, provincial and municipal regulations. Waste generated is minimized through application of the 3Rs – Reduce, Reuse, Recycle.

For radioactive waste generated during future operations and decommissioning, upon removal from the generating station, OPG will be directly responsible for the safe handling, movement, processing, storage and monitoring of nuclear fuel and all associated hazards with these processes. Organizational responsibilities, interfaces, controls, and key program elements will be described within the program for the management of nuclear waste.

OPG will submit the supporting document Radioactive Waste Management Plan which will provide further details on the radioactive waste management plan for the future operation of the DNNP BWRX-300 [R-32] and also address an existing DNNP commitment [R-2].

Waste Minimization, Segregation and Characterization

Section 4.5.16 of this Application document discusses how the BWRX-300 design helps in waste minimization and segregation. OPG has a robust procedure to minimize the waste, segregate and characterize the waste generated during operation of its nuclear facilities. This procedure will be amended or replaced as necessary to include the waste generated by the BWRX-300 plant design.

Waste generated from BWRX-300 as a result of daily operations and maintenance activities and during planned and unplanned outages will be characterized as either radiological or conventional depending on the radiological zone of its origin and from radiological surveys and analysis, in order to ensure that waste is safely managed.

4.11.2 Soils, Wastes and Hazardous Substances

Hazardous substances that may be present and/or hazardous wastes generated as a result of construction activities will be limited to those employed during standard construction processes. These would include chemicals, fuel, lubricants and compressed gases used during operation and maintenance of construction equipment, as well as solvents and cleaners used to clean the equipment. Additional substances on-site may consist of paint, aerosol cans, oil, batteries and electrical components used in the construction of the facility.

DNNP construction activities may generate surplus soil that will require management or proper placement and/or disposal. Surplus soils will be handled in accordance with the requirements of the *DNNP EMPP* [R-80].

During construction, non-nuclear hazardous substances and waste will be managed through site-specific environmental protection plans/procedures as required in the Commitments Report [R-2].

Radioactive Wastes

The activities to be licensed under the DNNP LTC will not involve the handling of radioactive materials and will not generate any radioactive wastes.

Upon the station becoming operational, low level radioactive waste will be collected from designated areas throughout the facility. Staff will separate the solid waste into conventional and radiological and hazardous waste streams and prepare and stage for shipment and disposal. Low and intermediate level waste will be transported to a licensed waste management facility for interim storage followed by long-term management/disposal.

Further information on the measures proposed for the safe management of radioactive waste of all types that will be generated by DNNP BWRX-300 are in Section 4.5.16 of this document and, PSAR Chapter 11 and 20 [R-6]. The PSAR also describes how these measures meet the relevant safety requirements including the measures taken for the safe management and disposal of this waste.

Additional details will be provided in the Radioactive Waste Management Plan [R-32].

Conventional Solid Waste

Conventional (non-radioactive, non-hazardous) solid waste is minimized to reduce the impact on the environment. This reduction includes implementation of the 3Rs (Reuse, Reduce and Recycle). Potential recyclable material collected and processed includes wood, cans, cardboard, paper, paper towels, newspaper, plastic, asphalt, concrete, compost, metal and glass. Waste that cannot be recycled is sent to landfill.

Non-hazardous waste treatment during construction will be based on sustainable materials management, a systemic approach to productively use and reuse materials over their life cycles. This method is based on the waste hierarchy, made up of five steps: reducing waste at the source, reuse of materials, recycling, energy recovery, and landfilling. Dumpsters for segregated waste collection available during construction are included in the site services plan and will be managed by a separate contract with a local third-party waste management provider.

4.11.3 Waste Minimization

OPG is committed to minimizing the amount of low-level radioactive waste requiring long term disposal. OPG is focused on reducing its environmental footprint, with a renewed focus on waste minimization, volume-reduction, and processing to divert material from interim storage and permanent disposal. OPG will deploy the same principles under a future operating licence for DNNP. Waste minimization is further discussed in Section 4.5.16 of this document and PSAR Chapter 11 and 21 [R-6]

4.11.4 Decommissioning Practices

OPG is responsible for planning, executing and funding all the phases of decommissioning the DNNP. Any DNNP decommissioning work will be conducted in accordance with the management system requirements (see Section 4.1) and in compliance with OPG's W-PROG-WM-0003, *Decommissioning Program* [R-112] and regulatory requirements. This ensures that when retiring the DNNP facility permanently from service and rendering it to a predetermined end-state condition, actions are taken in the interest of health, safety, environment, security, quality and economics.

Preliminary Decommissioning Plans (PDPs) will be prepared and submitted it to the CNSC for acceptance for each stage of the DNNP lifecycle [R-32]. In November 2021, OPG provided an updated site preparation PDP [R-33], which was accepted by the CNSC.

Each PDP defines the areas to be decommissioned and the sequence of the principal decommissioning work. The PDPs also demonstrates that decommissioning is feasible with existing technology and provide the basis for estimating the cost of decommissioning. The PDPs document the selected decommissioning strategy for that stage: main decontamination, dismantling and/or clean-up activities; end-state objectives; an overview of the principal hazards and protection strategies; a waste management strategy; a cost estimate; and financial guarantee arrangements.

PDPs are reviewed, and as necessary, updated to reflect new information and submitted to the CNSC for acceptance every five years or as requested by the CNSC.

A PDP is progressively updated, where needed, to reflect the appropriate level of detail required for the licensed activities. Prior to the decommissioning stage, a detailed decommissioning plan will be developed. The document will refine and add details to the PDP.

For the LTC, the site preparation PDP will be superseded by a PDP for the decommissioning of the site and an updated Financial Guarantee, in the event the project would be cancelled during or after completion of construction (prior to operations). The PDP for the construction phase will be prepared to the requirements of REGDOC-2.11.2, *Decommissioning* [R-113] and will describe the end state of the facility after decommissioning, for CNSC acceptance. Considerations for construction from a decommissioning perspective and Financial Guarantees associated with decommissioning are discussed in Section 4.5.3 of this Application. PDPs for both the as-built facility and end-of-life will be submitted as part of the LTC Application supporting materials [R-32].

4.11.5 Applicable OPG Documents

The OPG governance documents for the Waste Management SCA, which supports the licensing basis, are included in Table 4.11-1 below:

Table 4.11-1: Management System Documents for the Waste Management SCA

Document	Title
OPG-PROG-0005	Environment Health and Safety Managed Systems
OPG-STD-0156	Management of Waste and Other Environmentally Regulated Materials
W-PROG-WM-0003	Decommissioning Program

4.12 Security

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

The objective of the OPG Security Program is to establish a state of security readiness to ensure safe and secure operation of OPG stations and facilities.

4.12.1 General Considerations

OPG will implement security measures appropriate for each phase of DNNP based on the requirements of the Nuclear Security Regulations and associated regulatory documents as well as any additional measures identified through its threat and risk assessment process. Security measures relating to the DNNP site will be established in accordance with N-PROG-RA-0011, *Nuclear Security* [R-114].
The OPG Security Program supports OPG's need to manage residual risk to the public created by the operation of its facilities, protect assets and respond to emergencies that impact operations and the public. Key elements of this program include response to threats and maintaining compliance with legislative requirements, while minimizing the adverse impact on legitimate staff and plant operations. The security program implemented for DNNP will be revised as required to address risk and regulatory requirements associated with the project. Revisions will occur in a phased approach reflecting the stages of the project life cycle currently in site preparation, through to construction, and operation and finally to decommissioning.

The regulatory documents REGDOC-1.1.1 [R-18], REGDOC-2.12.2 Volume 1 [R-38], REGDOC-2.12.1 Volume II [R-39], and REGDOC-2.12.2, *Security: Site Access Security Clearance* [R-116] were reviewed in the context of OPG's Nuclear Security program as it relates to the phases of DNNP. The review concluded that OPG's Management System relating to security, implemented through the Nuclear Security program document and associated instructions and guides, meets the requirements of the applicable regulatory documents.

<u>Threat Risk Assessment</u>

Construction NK054-REP-00531-10000, *Site Threat and Risk Assessment (TRA) Report* [R-109] has been completed for the DNNP site. The assessment will be updated as the project proceeds to reflect any changes in risks and necessary security measures.

Facilities and Equipment

The proposed DNNP facility will utilize security by design principles, creating engineered barriers against malevolent acts. The protected area will include detection and assessment and access control devices that meet or exceed the requirements of REGDOC-2.12.1 Volume II [R-39] and REGDOC-2.12.3, *Security of Nuclear Substances: Sealed Sources and Category I, II and III Nuclear Material* [R-117]. Additional access control measures will be incorporated for any defined vital areas and areas containing sealed sources as appropriate. The specifics that detail the aforementioned approach are contained in the Darlington BWRX-300 Security Assessment [R-77].

Response Arrangements

OPG maintains an on-site nuclear response force, trained and equipped in accordance with REGDOC-2.12.1 Volume I [R-38] at the DN site. OPG has MOUs with the Durham Regional Police Service for additional off-site response. OPG continues to regularly review and revise as needed the tactical plans and response agreements with Durham Regional Police Service as appropriate. The detailed tactical deployment plan for DNNP will be developed in support of future operations.

Security Practices

OPG has a mature and robust security program, N-PROG-RA-0011 [R-114], which documents high level security program requirements and outlines the implementing documentation. The program and supporting procedures include measures to protect prescribed information and a clearance process for personnel requiring site access or access to prescribed information.

Security Training and Qualification

OPG maintains a complement of fully trained and qualified Nuclear Security Officers. OPG has a comprehensive security training program, developed and delivered by a dedicated in-house training group. OPG's training program meets or exceeds all regulatory requirements. Updates to the program will be determined during the detailed design phase of construction and developed to support future operations and the operating licence.

4.12.2 Cyber Security

OPG has an established cyber security program, OPG-PROG-0042, *Cyber Security* [R-118] that complies with CSA N290.7-14, *Cyber security for nuclear power plants and small reactor facilities* [R-119]. OPG's nuclear cyber security program incorporates continuous improvement efforts to ensure the enhancement of OPG's cyber security posture (i.e., applicable cyber assets are protected from cyber attacks).

Continuous improvement efforts are based on applicable CSA standards (and their updates), OPEX, benchmarking, and industry best practices.

OPG's existing cyber security program and procedure include the following elements:

- Defensive strategy and security architecture
- Policies and procedures
- Asset identification and classification
- Roles and responsibilities
- Security controls

OPG has a dedicated nuclear cyber security training, as well as individual qualifications throughout the other lines of business within nuclear.

Configuration management and lifecycle management are ensured through change control processes that are allowed for cyber essential assets (CEAs), which have cyber security portions embedded to ensure CEAs are in compliance with CSA N290.7-14 [R-119].

Nuclear cyber security is embedded within the applicable programs throughout OPG. This includes dedicated single points of contact within lines of business, embedded steps within their procedures and training, and OPG wide cyber security interface meetings.

OPG's cyber security incident response guide provides the necessary guidance for responding to cyber security incidents. Reporting is provided through the cyber security program and procedure while recovery plans are covered within the individual CEA controls where required.

4.12.3 Applicable OPG Documents

The OPG governance documents for the Security SCA, which supports the licensing basis, are included in Table 4.12-1 below:

Table 4.12-1: Management System Document for the Security SCA

Document	Title
N-PROG-RA-0011	Nuclear Security
OPG-PROG-0042	Cyber Security

4.13 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 IAEA INFCIRC/140, *Treaty on the Non-Proliferation of Nuclear Weapons* [R-120].

4.13.1 General Considerations

OPG has an effective N-PROG-RA-0015, *Safeguards and Nuclear Material Accountancy* program [R-121], which meets the requirements outlined in REGDOC-2.13.1, *Safeguards and Nuclear Material Accountancy* [R-122]. This is achieved through:

- strict inventory tracking of nuclear materials by use of a nuclear material accountancy software;
- training of staff involved with the maintenance, inspection, verification, or movement of nuclear materials on the requirements of reporting and tracking information in software or by physical means;
- site specific operating procedures that provide instructions on meeting the established Safeguards agreement;
- strict access control to areas that contain nuclear materials; and
- responsibilities of facility specific Safeguards Officers to perform day-to-day safeguard related duties.

OPG has programs in place at existing operating nuclear facilities to facilitate Canada's compliance with all applicable safeguards' agreements. In support of future operations and a LTO, OPG will put in place similar measures at DNNP.

Additionally, OPG will meet the requirements such as declarations pursuant to the Additional Protocol on future plans and providing access and assistance to IAEA inspectors for complementary access.

Measures to control access to prescribed equipment, as well as prevent loss or illegal use, possession, or removal of prescribed equipment, already exist, or will be developed upon identification that any such equipment will be used during the licensed activities.

As described in Section 2.2.1 of this application, OPG has a number of Import or Export licences related to DNNP that will continue to be managed separately from the current PRSL and the proposed construction licence.

The records required by safeguard agreements will be kept and disclosed as appropriate to the CNSC and IAEA inspectors.

Design Information Questionnaire

OPG has submitted a *Design Information Questionnaire (DIQ)* [R-82] for the facility to the CNSC as part of the supporting documents for this Application. The DIQ provides the IAEA/CNSC with information pertaining to the facility's design, operation, locations of nuclear material inventory and nuclear material flow points. This ensures that the IAEA has the required information to establish safeguards measures and responsibilities to be met as per the facility attachment. The DIQ will be updated further as construction progresses to notify the CNSC and IAEA of changes that may impact the proposed safeguards measures.

The DIQ ensures that general information describing the facility, design and operation, nuclear material descriptions, processing and flow of nuclear materials, safeguard measures, and accounting and reporting of nuclear materials are accurate, up-to-date, and available to support a Design Information Verification inspection during future operations. Design Information Verifications are supported by Safeguards Officers and impacted work groups through planned or unplanned inspections.

Safeguards Equipment

Safeguards equipment for containment and surveillance is developed by the IAEA as a result of the DIQ and proposed safeguards approach. OPG will provide the IAEA with any requested assistance to facilitate access for the installation of safeguards equipment, including remote monitoring and surveillance systems.

OPG maintains a strong program and working culture based on IAEA support and respect for safeguards equipment and seals. Individuals that work on, around, or near IAEA equipment will be trained to identify and not manipulate any IAEA equipment or seals. Future operating procedures will be developed outlining required responses and notifications in the event that services are lost or damage occurs to IAEA equipment.

<u>Access and Assistance to IAEA</u>

OPG's safeguards implementing standard gives guidance on the reporting requirements and access by IAEA inspectors into the facilities.

As required by REGDOC-2.13.1 [R-122] OPG provides access to the IAEA following notification and identified scope of activities. Where required, OPG provides provisions of ladders, scaffolding, lifting, or training. OPG provides prompt access for all reasonable requests on short notice notifications for access to equipment.

<u>Nuclear Accountancy and Control</u>

As per REGDOC-2.13.1, OPG will be required to submit DNNP documents supporting Canada's Safeguards and Nuclear Material Accountancy obligations. Safeguards related information is maintained on a Nuclear Material Accountancy software, which provides near real-time inventory of:

- fresh and irradiated fuels;
- fuel that is in the reactor core;
- fuel that is located anywhere else within the facility to support operations; and
- other relevant non-fuel nuclear material required to support operations.

As previously noted, the proposed licenced activities for this Application do not include the receipt or handling of nuclear fuel. OPG will ensure that nuclear accountancy and control measures will be in place in support of a future application to receive and handle nuclear fuel at the DNNP.

4.13.2 Applicable OPG Documents

The OPG governance documents for the Safeguards and Non-Proliferation SCA, which supports the licensing basis, are included in Table 4.13-1 below:

Table 4.13-1: Management System Document for the Safeguards and Non-Proliferation SCA

Document	Title
N-PROG-RA-0015	Safeguards and Nuclear Material Accountancy

4.14 Packaging and Transport

The Packaging and Transport SCA covers programs for the safe packaging and transport of nuclear substances to and from the facility.

There will be no nuclear substances or controlled nuclear components encompassed by the requested construction licence. While construction related tools containing radioactive nuclear substances may be used, these will be under the authority of separate CNSC nuclear substance and device licences.

OPG has a mature and effective packaging and transport program that meets or exceeds all applicable regulatory requirements and related objectives. The objective of the program is to ensure the safe packaging and transportation of nuclear substances and radiation devices to and from OPG's nuclear facilities such that the risk to the public, workers, and the environment is low.

The program addresses package engineering, qualification and certification, operations and maintenance (including periodic testing and inspection), records, staff training, preliminary event notification, and transportation emergency response.

Detailed information pertaining to radioactive transportation operations specific to the DNNP is to be addressed in later licensing stages. The information provided will effectively facilitate commissioning before fuel load and allow preparation for the transition to fuel-in commissioning and operation upon a future LTO.

5.0 Other Regulatory Areas

5.1 Reporting Requirements

OPG will provide scheduled and event reporting in accordance with requirements of REGDOC-3.1.1 [R-72], as they apply to the licensed activities, and in accordance with reporting requirements specified in the Licence and Licence Conditions Handbook. Since REGDOC-3.1.1 is primarily intended for operating NGSs, OPG will work with the CNSC to identify those requirements that remain applicable during the construction and fuel-out commissioning phases.

5.2 Environmental Assessment

In 2006, the Government of Ontario directed OPG to initiate the federal approvals process for new nuclear generation at an existing OPG site. A public hearing, conducted by a JRP of the Canadian Environmental Assessment Agency and CNSC was held from March to April 2011 to consider the DNNP EIS [R-29] and original PRSL application [R-9]. The DNNP EA was assessed for a plant capacity of up to 4800 MWe using a Plant Parameter Envelope (PPE) [R-64] approach that encompassed several reactor technologies being considered.

In May 2012, the Government of Canada accepted the JRP recommendations for the DNNP, which concluded that the DNNP would not likely cause significant adverse environmental effects provided that OPG implements the mitigation measures proposed and commitments made during the hearing as well as the JRP's recommendations.

As part of the EA process, OPG prepared the DNNP EIS which identified VECs including pathways for transfer of potential effects and evaluated the potential impacts of the project on the environment.

Since the nuclear technology had not been selected at the time of the EA, the EIS was completed using a bounding framework which encompassed a range of reactor designs under consideration at that time.

OPG selected the BWRX-300 reactor technology for the DNNP in December 2021 and subsequently conducted a comprehensive EIS review [R-4] to ensure that the results of the EIS remain valid using the chosen SMR technology.

The comprehensive EIS review found that the BWRX-300 is expected to involve works and activities that are essentially the same as those evaluated in the original EIS. As compared to the reactors considered in the EIS, the BWRX-300 reactors are smaller in physical size and electrical power. As a result, the effects of the BWRX-300 deployment on the environment are generally less than those examined in the EIS.

Overall, environmental effects (including effects from accidents, malfunctions and malevolent acts, effects of the environment on the Project, and cumulative effects) from the BWRX-300 are expected to be less than those assessed in the EIS. Therefore, the determinations regarding the significance of residual effects made in the EIS remain valid.

OPG recognizes that while the assessment of environmental impacts from DNNP has been satisfied from the Western/scientific knowledge perspective, it may not fully address the impact of the DNNP on Indigenous inherent and treaty rights as they are understood today. OPG endeavors to continue to work with Indigenous Nations and communities to appropriately identify the rights impacted by the Project and to achieve feasible mitigation measures and/or accommodations.

5.3 Public Information and Disclosure Program

OPG's principles and processes for external communications are governed by the nuclear standard N-STD-AS-0013, *Nuclear Public Information and Disclosure* [R-105] and this standard will apply to the DNNP.

This document guides OPG's external community stakeholder activities, public response requirements of issues or significant events and OPG's standards to respond to the public.

OPG's nuclear public information disclosure protocol is posted to our public website: www.opg.com.

For OPG's Darlington site, the community relations program proactively provides information to the public on the existing Darlington station operations and the status of key projects, including the DNNP.

Communications and outreach activities in support of the DNNP are integrated into the framework of the existing public information program for the DNGS. The information program has been in existence for many years and meets or exceeds all regulatory and OPG corporate requirements, specifically:

- CNSC REGDOC-3.2.1, *Public Information and Disclosure* [R-106]
- CNSC REGDOC-3.2.2, *Indigenous Engagement* [R-123]

OPG has continued to inform the public and stakeholders about the status of DNNP as part of the existing DN public information program, through various methods and forums, and OPG will continue to do so throughout the construction period under the requested LTC Application.

Objectives of OPG's Public Information and Disclosure Program

- Share information and communicate plans and activities to Indigenous communities, stakeholders, and the public;
- Proactively identify key information and activities from the DNNP that may be of interest or impact the community including from an environmental (including effects and mitigation), health and safety perspective;
- Inform and educate stakeholders on the characteristics of the SMR project, technology, and licensed activities;
- Seek informed views and perspectives;
- Promptly respond to issues raised by the community, stakeholders, and the public; and,
- Incorporate topics of interest into future DNNP communication materials and activities.

<u>Target Audiences</u>

The primary focus area for the engagement activities, in addition to the public at large, includes two municipalities proximate to the DN site including the host community (the Municipality of Clarington) and adjacent communities within 10 km of the project (the City of Oshawa). The 10 km radius is consistent with the DNGS Detailed Planning Zone for nuclear emergency planning purposes, an area where residents are most familiar with nuclear plant operations and regularly receive information station and operational updates.

OPG seeks to ensure the public and stakeholders with a potential interest in the DNNP are provided with relevant information and have the opportunity to share their views and perspectives. Information will be communicated in a number of ways based on target audience identification, their interests, and preferred means of communication.

DNNP Stakeholders and target audiences may include but are not limited to:

- Indigenous Nations and communities;
- Residents in the vicinity of the DNGS and the public;
- Elected officials in the host community and adjacent areas;
- Key community stakeholders and leaders;
- Established community committees such as the Darlington Community Advisory Committee and the Durham Nuclear Health Committee;
- Local businesses and business organizations, such as boards of trade and chambers of commerce;
- Private/public community organizations and special interest groups;
- Non-Governmental Organizations;
- Nuclear industry associations/organizations and regulatory bodies;
- Media;
- Federal, provincial, regional, and municipal agencies and officials with a regulatory role or project interest;
- OPG employees and retirees;
- Intergovernmental agencies; and,
- Broader community of interest (Regional, National, International), including those interested in SMR technology, energy, climate change, and/or environmental issues.

Communication Methods

Communications methods are the approaches and activities used to distribute information, and to solicit feedback and input during the DNNP. The methods to be employed during the construction licensing phases of the project will be specific to the issues and matters that arise, however, will include:

Notification Advertisements and Letters

Public notifications will be prepared and distributed to announce the submission of the LTC Application as well as commencement of site preparation and construction activities, via a press release (as required), stakeholder letter(s), web communications, the DN community newsletter (Darlington Neighbours) and advertisements in local print media (as required).

Website

The OPG website for the DNNP will be updated. The web site serves as a vehicle to provide access to information, as well as a mechanism to receive input from interested persons as an enhancement of the public consultation program. Information such as project scope, schedule, descriptions, process steps, events, and contacts will be maintained.

Toll Free Information Line

A 1-800 information line will continue to be maintained. Messages will be checked and responded to on weekdays and any required follow-up will be completed in a timely manner.

Media Relations

Ongoing liaison with respect to site preparation and construction activities will be initiated and maintained by OPG with reporters and news editors for both electronic and print media.

OPG Employee Consultation Activities

The employee communication program will include articles written in OPG wide and Darlington Station specific employee vehicles. Staff presentations and information sessions will continue to be held. A specific intranet site will continue to be maintained to facilitate communication with employees.

Key Stakeholder Briefings and Interviews

Interviews and briefings will continue to be conducted to present information and provide an opportunity to have questions and comments addressed. Regular updates will be presented to municipal representatives, established community committees including the Darlington Community Advisory Committee and Darlington Nuclear Health Committee and other key stakeholders on a frequency commensurate with key project activities and milestones. Feedback from these meetings will be recorded for response and issue management.

Workshops

Key stakeholders with a high level of interest in the project activities may be invited to participate in workshops that will involve meaningful discussions and provide substantive input to phases of the project.

Information Sessions

Information sessions (in person or virtual) advertised broadly and open to any participants will provide an opportunity to learn about the DNNP and the licensing phases/activities and provided comments and/or have questions answered.

Public Comments Tracking

A public comment tracking system is maintained to record and monitor comments received by the public and stakeholders involved in or affected by the DNNP. A public comment tracking system helps ensure that any issues and concerns held by the public or stakeholders are identified and responded to, to the extent possible.

Information Centre

DN maintains a fully staffed Public Information Centre where members of the public can receive information on DNNP as well as on current operations, and where staff can respond to questions.

Social Media

OPG maintains a presence on social media (Facebook, Twitter, and Instagram) and shares information through these media.

Community Outreach

Outreach activities to interested groups and communities may include:

- **Presentations and site bus tours** of the Darlington site (including the DNNP lands) to community groups, key stakeholders, industry partners and the general public.
- **Quarterly Neighbours Newsletter** for the DNGS, which is distributed to about 120,000 residents and businesses within ten kilometers of Darlington and posted online. Applicable updates and/or inserts on DNNP and licensing activities are included in these newsletters.
- **The Information Centre** is available to community groups to host events related and unrelated to the industry. Its meeting room and event space were built to help build greater ties to the community, and DNNP materials are available at the Information Centre to provide information and generate discussion.
- **OPG's annual public open house**, which is widely advertised with a focus on the nearby community, provides information on DNNP. Staff from OPG and various industry partners are present to answer questions and provide information on DNNP, nuclear power and existing station operations (see Figure 5.3-1).

Figure 5.3-1: OPG Annual Public Open House

Community Committees

OPG works with established local community committees on matters of interest and concern related to our operations and projects. Updates on the status of DNNP and licensing activities are provided to the committees.

- The Darlington Community Advisory Committee meets regularly to exchange information with community leaders and local residents, who in turn provide advice to senior OPG staff on issues of environmental, economic and public concern.
- OPG has representatives on the Durham Nuclear Health Committee and OPG staff make regular presentations on a variety of environmental, community outreach and operational issues. The committee is chaired by the Durham Region Medical Officer of Health.

OPG meets often with stakeholder groups, elected officials and municipal representatives, as well as with stakeholder groups that have an interest in nuclear and/or SMR technology, safety energy, climate change, and/or environmental issues.

OPG recognizes that members of the local community, stakeholders and the general public have legitimate interests in licensing activities related to DNNP. Through all phases of the DNNP, OPG has continued to use the existing public information program (including activities and forums described above) to communicate relevant information and will continue to do so throughout the licensing activity period, including promotion of opportunities for public involvement in the process.

In preparation for this Application, OPG established communications and engagement activities with the specific objectives of ensuring stakeholders, the public and Indigenous Nations and communities were:

- aware of OPG's intention to apply for the licence;
- provided with a forum to discuss key topics of interest related to the Application; and,
- made aware of opportunities for participation in the licensing process.

The activities to be undertaken will meet the requirements outlined in CNSC REGDOC-3.2.1 [R-106].

OPG will also continue to meet the commitments made as part of the initial PRSL [R-1] with respect to communications and consultation.

Further details on public and stakeholder engagement plan and activities in support of OPG's DNNP are provided in the *DNNP Stakeholder Engagement Plan* [R-68].

5.4 Indigenous Engagement

OPG recognises that engagement begins with relationship-building and establishment of trust, and is committed to respect, openness and transparency in building these relationships. OPG's OPG-POL-0027, *Indigenous Relations Policy* [R-124], provides a framework for engaging with Indigenous Nations and communities and providing support for programs and initiatives.

Indigenous Nations and communities that have been identified as having rights or interests in the DNNP, include:

- Members of the Williams Treaties First Nations:
    - Beausoleil
    - Rama
    - Georgina Island
    - Scugog Island
    - Hiawatha
    - Curve Lake
    - Alderville;
- Huron Wendat;
- Kawartha Nishnawbe;
- Mohawks of the Bay of Quinte;
- Métis Nation of Ontario, Region 8; and,
- Saugeen Ojibway Nation.

Regular project updates are provided to Indigenous Nations and community representatives and OPG has hosted a number of tours of the DNNP site and our current DNGS operations and DWMF.

At this stage of the project, the formal engagement has included: Information sharing through meetings and presentations, requests for feedback and perspectives from communities with an emphasis on working together to recognize and minimize environmental impacts, requests for review and input to draft and early revision documents, seeking opportunities to incorporate Indigenous Knowledge, invitation to input to and observe archaeological activities on the DNNP site, and discussion of economic benefit opportunities.

Through the establishment of a formal agreement supporting ongoing engagement, OPG meets monthly with members of the Williams Treaties First Nations to provide updates, receive feedback and answer questions. The agreement stipulates capacity funding to support the communities' participation. OPG works with the Indigenous Nations and communities to ensure meaningful engagement and communications. Further information is detailed in the NK054-

REP-07421.3-00002, *DNNP Indigenous Engagement Report April 2020 – August 2022* [R-125].

Despite the DNNP project being paused for about six years after the PRSL was first issued in 2012, OPG has continued to meet regularly with the Indigenous Nations and communities to provide details of OPG's nuclear operations, projects (including DNNP where appropriate), environmental performance, and to discuss interests and identify concerns. With the restart of the DNNP Project about 2019, more focus has been added in this engagement, specific to DNNP.
OPG also maintains a public website (www.opg.com) which disseminates relevant information, reporting and notices regarding company activities, including for DNNP to the general public as well as Indigenous Nations and communities.

REGDOC-3.2.2 [R-123] sets out requirements and guidance for licensees on Indigenous engagement. OPG conducted a review of REGDOC-3.2.2 to ensure compliance between the current regulatory document and OPG's relevant management system documents.

As recommended in REGDOC-3.2.2 and as required by DNNP site preparation commitment [R-2], the NK054-PLAN-01210-00028, *DNNP Indigenous Engagement Plan* [R-126] was prepared and submitted to the CNSC in September 2021, to document the engagement scope and activities on the DNNP that OPG has and will conduct throughout the site preparation and construction phases of the project with identified Indigenous Nations and communities.

Additionally, OPG submitted an *Indigenous Engagement Report* [R-125] for DNNP to the CNSC. The report provides a list of Indigenous Nations and communities identified for engagement, a summary of the Indigenous engagement activities conducted to date, a description of planned Indigenous engagement activities, and the proposed schedule for interim reporting to the CNSC.

Planned engagement in 2022 and 2023 and 2024 ahead of the LTC hearing will be focused on the identified Indigenous Nations and communities noted above and will include continued information sharing with an emphasis on working together to incorporate Indigenous Knowledge into the project as well as other topics of interest raised in previous engagement sessions, for example:

- Employment and potential commercial opportunities
- Minimization of environmental impacts (terrestrial & aquatic)
- Air and water emissions
- Land Use
- Nuclear waste management

Further details on the indigenous engagement plan and activities for the DNNP are provided in the *DNNP Indigenous Engagement Report* [R-125].

5.5 Cost Recovery and Financial Guarantees

OPG recognizes and will meet its financial obligations under the Cost Recovery Fees Regulations, to pay applicable licensing fees for the proposed licence.

The NSCA [R-37] and its Regulations require that applicants make adequate provisions for the decommissioning of facilities licensed by CNSC. CNSC REGDOC-3.3.1, *Financial Guarantees for Decommissioning of Nuclear Facilities and Termination of Licensed Activities* [R-127] provides guidance regarding the establishment and maintenance of measures to fund the decommissioning of nuclear facilities.

A financial guarantee is currently in place for the PRSL. A financial guarantee for the construction phase will be submitted as a supplemental document to the LTC Application [R-32].

5.6 DNNP Commitments

OPG made a number of commitments during the EA and licensing process to obtain the PRSL. Following the issuance of the licence [R-8], these commitments were captured in the DNNP Commitments Report [R-2]. These commitments are tracked, reviewed and regularly reported to CNSC.

To facilitate their implementation, OPG commitments associated with DNNP have been organized into three groups to align with the applicable project phase (i.e., site preparation, construction, and operation). At this time, all commitments that are applicable to phase 1 of site preparation (early works) have been addressed and submitted to the CNSC for acceptance and closure, where applicable.

The commitments applicable to the construction phase and their status are listed in Table 5.6-1 below.

Table 5.6-1: DNNP Commitments for Construction Phase

Commitment #	Deliverable Title	Deliverable Elements	Submission Timeline
D-P-9.3	Site Geotechnical Investigation	Geotechnical Investigation Results Report	Linked to the LTC Application
D-P-9.4	Site Seismic Hazards Studies and Investigation	Seismic Investigation Results Report	Linked to the LTC Application
D-P-12.1(a)	EIS Review	EIS Comprehensive Review Report	Prior to LTC Application
D-C-1.2	EPC CCW Design	CCW Design	Prior to construction of CCW
D-C-2.1	Non-Radiological Effluents Management Program / Design Documentation	Non-Radiological Effluents Management Program / Design Documentation	Prior to fuel-out commissioning
D-C-3.1	Preliminary Safety Analysis and Design	PPE Review Report	Prior to LTC Application
		Preliminary Safety Analysis Report	Linked to the LTC Application
D-C-4.1	Radiological Effluents Management Program / Design Documentation	Radiological Effluents Management	Prior to construction of the reactor building/nuclear island

Commitment #	Deliverable Title	Deliverable Elements	Submission Timeline
		Program / Design Documentation	
D-C-5.1	Radiological Air Emissions Management Program / Design Documentation	Radiological Air Emissions Management Program / Design Documentation	Prior to construction of the reactor building/nuclear island
D-C-5.2	Non-Radiological Air Emissions Management Program / Design Documentation	Non-Radiological Air Emissions Management Program / Design Documentation	Prior to fuel-out commissioning
D-C-6.1	Radiological Environmental Monitoring Program	Radiological Environmental Monitoring Program	Prior to fuel-in commissioning
D-C-7.1	Contingency Plan for Flooding and Other Extreme Weather Hazards	Contingency Plan for Flooding and Other Extreme Weather Hazards	Prior to construction of the reactor building/nuclear island
D-C-8.1	Meteorological Monitoring Station	Meteorological Monitoring Station Strategy	Prior to construction of meteorological monitoring station
D-C-9.1	Radioactive Waste Management Plan	Radioactive Waste Management Plan	Linked to the LTC Application

6.0 Overall Conclusion

OPG is requesting a ten (10) year licence to construct a Class 1A Nuclear Facility at OPG's DN site and conduct the authorized activities listed in Section 2 of this Application.

Through this Application and its supporting documents, OPG has demonstrated that:

(a) It is qualified to carry on the activity to be licensed; and
(b) will, in carrying on that activity, make adequate provision for the protection of the environment, the health and safety persons and the maintenance of national security and measures required to implement international obligations to which Canada has agreed.

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8.0 Glossary

ABWR	Advanced Boiling Water Reactor
AL	Action Level
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
BDBA	Beyond Design Basis Accident
BIS	Boron Injection System
BL-DSA	Baseline – Deterministic Safety Analysis
BLEVE	Boiling Liquid Expanding Vapour Explosion
BP	Balance of Plant
BWR	Boiling Water Reactor
CB	Control Building
CCF	Common Cause Failure
CCS	Containment Cooling System
CCW	Condenser Cooling Water
CFD	Condensate Filters and Demineralizer System
CFHS	Condensate and Feedwater Heating System
CN-DSA	Conservative – Deterministic Safety Analysis
CNO	Chief Nuclear Officer
CNSC	Canadian Nuclear Safety Commission
CRD	Control Rod Drive
CRDH	Control Rod Drive Hydraulic
CST	Condensate Storage Tank
CWE	Chilled Water Equipment
CWS	Circulating Water System
DA	Design Authority
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DFO	Department of Fisheries and Oceans

DiD	Defence-in-Depth
DIQ	Design Information Questionnaire
DL	Defence Line
DN	Darlington Nuclear
DNGS	Darlington Nuclear Generating Station
DNNP	Darlington New Nuclear Project
DPS	Diverse Protection System
DSA	Deterministic Safety Analysis
DWMF	Darlington Waste Management Facility
EA	Environmental Assessment
ECA	Environmental Compliance Approval
EIS	Environmental Impact Statement
EMEAF	Environmental Monitoring and Environmental Assessment Follow-up
EMI	Electromagnetic Interference
EMP	Environmental Monitoring Program
EMPP	Environmental Management and Protection Plan
EMS	Environmental Management System
EOP	Emergency Operating Procedure
EPZ	Emergency Planning Zone
EQ	Equipment Qualification
ERA	Environmental Risk Assessment
ERF	Emergency Response Facilities
ESA	Endangered Species Act
ESBWR	Economic Simplified Boiling Water Reactor
ETE	Evacuation Time Estimate
EX-DSA	Extended – Deterministic Safety Analysis
FMCRD	Fine Motor Control Rod Drive System
FPC	Fuel Pool Cooling and Cleanup
FPS	Fire Protection System
FSF	Fundamental Safety Function
GEH	General Electric Hitachi

GHG	Greenhouse Gas
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
HFE	Human Factors Engineering
HFEPP	Human Factors Engineering Program Plan
HP	High Pressure
HSI	Human-System Interface
HSMS	Health and Safety Managed System
HVAC	Heating Ventilation and Air Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICS	Isolation Condenser System
INPO	Institute of Nuclear Power Operation
JRP	Joint Review Panel
LOCA	Loss of Coolant Accident
LOPP	Loss of Preferred Power
LP	Low Pressure
LTC	Licence to Construct
LTO	Licence to Operate
LWMS	Liquid Waste Management System
MCR	Main Control Room
MECP	Ministry of Environment, Conservation and Parks
MOU	Memorandum of Understanding
MWe	Megawatt Electric
MWS	Makeup Water System
NBS	Nuclear Boiler System
NGS	Nuclear Generating Station
NHS	Normal Heat Sink
NMS	Nuclear Management System
NSCA	Nuclear Safety and Control Act

OHSA	Occupational Health and Safety Act
OLC	Operating Limits and Conditions
OPEX	Operating Experience
OPG	Ontario Power Generation Inc.
PCCS	Passive Containment Cooling System
PCS	Primary Containment System
PCV	Primary Containment Vessel
PCW	Plant Cooling Water
PDP	Preliminary Decommissioning Plan
PHC	Petroleum Hydrocarbon
PIE	Postulated Initiating Event
PNERP	Provincial Nuclear Emergency Response Plan
PNGS	Pickering Nuclear Generating Station
PPE	Plant Parameter Envelope
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring
PRSL	Nuclear Power Reactor Site Preparation Licence
PSA	Probabilistic Safety Assessment
PLSA	Plant Service Area
PSAR	Preliminary Safety Analysis Report
QA	Quality Assurance
QC	Quality Control
R&D	Research and Development
RB	Reactor Building
RC&IS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RP	Radiation Protection
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
RWCU	Reactor Water Cleanup
SAF	Safety Assessment Framework

SAT	Systematic Approach to Training
SC	Safety Class
SCCV	Steel-plate Composite Containment Vessel
SCR	Secondary Control Room
SDC	Shutdown Cooling
SDD	System Design Description
SFP	Spent Fuel Pool
SMC	Site Management Centre
SMR	Small Modular Reactor
SSC	Structures, Systems and Component
SSSP	Site-Specific Safety Plan
SWMS	Solid Waste Management System
TB	Turbine Building
TRF	Tritium Removal Facility
USNRC	United States Nuclear Regulatory Commission
VCE	Vapour Cloud Explosion
VEC	Valued Ecosystem Component
WANO	World Association of Nuclear Operators

Appendix A Licence Application Matrix – Application Legislation

Legislation	Clause(s)	Application Cross-Reference
Nuclear Safety and Control Act	24(4)	4.1-4.14 5.0
	26(a),(e)	4.1-4.14 5.0
Legislation	Clause(s)	Application Cross-Reference
General Nuclear Safety and Control Regulations	3(1)(a)	1.2
	3(1)(b)	2.1
	3(1)(c)	4.9 4.11
	3(1)(d)	2.1 4.4 4.5 4.10 4.11 4.12
	3(1)(e)	4.4 4.5 4.7 4.9 4.11 4.12 4.14
	3(1)(f)	4.7
	3(1)(g)	4.5 4.12 4.13
	3(1)(h)	4.5 4.12

		4.13
	3(1)(i)	3.0 4.4 4.5 4.7 4.9 4.10 4.11 4.12
	3(1)(j)	4.5 4.11
	3(1)(k)	1.2 4.1 4.2 4.3
	3(1)(l)	5.5
	3(1)(m)	5.0
	3(2)	4.13
	10(b)	4.13
	12(1)(a)	4.1 4.2 4.7 4.10 4.12
	12(1)(b)	4.1-4.14
	12(1)(c)	4.3 4.4 4.5 4.7 4.8 4.9 4.10 4.11

		4.12
	12(1)(d)	4.7 4.9 4.10 4.12
	12(1)(e)	4.2 4.3 4.7 4.10
	12(1)(f)	4.3 4.4 4.5 4.7 4.9 4.10
	12(1)(g)	4.10 4.12
	12(1)(h)	4.10 4.12
	12(1)(i)	4.13
	12(1)(j)	4.2 4.12
	15	1.2 4.1
	15(a)	1.2
	15(b)	1.2
	15(c)	1.2
	17(a)	4.2 4.3 4.7 4.8 4.9
	17(b)	4.2 4.3 4.7 4.8

		4.9
	17(c)	4.1 4.2 4.3 4.7 4.8 4.9 4.12
	17(d)	4.2 4.3 4.7 4.8
	17(e)	4.1 4.2 4.3 4.7 4.8 4.9 4.12
	20(a)	4.14
	20(d)	4.13
	21	4.12
	21(1)(a)	4.13
	21(1)(b)	4.13
	22	4.12
	23	4.12
	23(2)	4.12 4.13
	27	4.1
	28	4.1 4.12
	28(1)	4.1
	29	4.3 4.7 4.12

		5.1
	30	4.3 4.12 4.13 5.1
	31	5.1
	32	4.3 5.1
Legislation	Clause(s)	Application Cross-Reference
Canadian Nuclear Safety Commission Cost Recover Fees Regulations	All	5.5
Legislation	Clause(s)	Application Cross-Reference
Class I Nuclear Facilities Regulation	3(a)	2.1 4.5 4.10 4.12
	3(b)	2.1 4.4 4.5 4.12
	3(c)	1.2
	3(d)	4.1
	3(e)	2.1 4.8 4.9 4.11
	3(f)	4.1 4.2 4.8 4.10 4.11
	3(g)	4.9
	3(h)	4.9
	3(i)	4.5 4.12

	3(j)	5.0
	3(k)	4.11
	5(a)	3.0 4.5
	5(b)	3.0 4.5 4.9
	5(c)	4.3
	5(d)	4.5
	5(e)	4.3 4.5
	5(f)	4.4 4.5
	5(g)	4.1
	5(h)	4.1 4.12 4.13
	5(i)	4.1 4.2 4.3 4.7 4.8 4.9 4.10 4.11 4.12 4.14
	5(j)	4.7 4.9 4.11
	5(k)	4.1 4.5 4.7 4.9 4.10 4.11

	5(l)	4.2 4.7
	5(m)	4.2 4.3
	9	4.2
	10	4.2
	11	4.2
	12	4.2
	14	4.7
	14(1)	4.1 4.9 4.11
	14(4)	4.1
	14(5)	4.1
Legislation	Clause(s)	Application Cross-Reference
Nuclear Non-proliferation Import and Export Control Regulations	All	4.13
Legislation	Clause(s)	Application Cross-Reference
Nuclear Security Regulations	All	4.5 4.12
	3(b)	2.1
	16	2.1
	37(1), (2) and (3)	4.1
	38	4.1 4.2

Appendix B REGDOC-1.1.2 Mapping to Application and OPG Documents

No.	REGDOC-1.1.2 Section(s)	Application Cross-Reference	Application Supporting Documents (Pre-Licensing and Packages 1-3)
1.	**Introduction**		
1.1	Purpose	N/A	
1.2	Scope	N/A	
1.3	Relevant legislation	N/A	
1.4	CNSC contact information	N/A	
2.	**Licensing Basis, Process and Submission**		
2.1	Background	N/A	
2.2	Licensing process	N/A	
2.3	Structuring the licence application	N/A	
2.4	Completing the licence application	N/A	
2.5	Submitting the licence application	N/A	
N/A	N/A	Introduction	• Level 1 and 2 Schedule [R-24]
3.	**Applicant's General Information**		
3.1	Identification and contact information	1.2	
3.2	Facility and activities to be licenced	2.1	
3.3	Other relevant information	2.2	• PSAR [R-6]
4.	**Safety Policies, Programs, Processes, Procedures and other Safety and Control Measures**		

4.1	Management system	4.1	• Management System Report [R-47]
4.2	Human performance management	4.2	• Construction and Commissioning Training Plan [R-66] • Program Management Plan [R-31] • PSAR [R-6]
4.3	Operating performance	4.3	
4.4	Safety analysis	4.4	• PSA Methodology [R-42] • Hazard Analysis Methodology [R-23] • PSAR [R-6] • Darlington BWRX-300 Security Assessment [R-77] • PSA Results Report [R-40] • Hazard Analysis Results Report [R-76]
4.5	Physical design	3.0 4.5	• PSAR [R-6] • Site Evaluation Update Summary Report [R-5] • Commitments Report [R-2] • EIS Review Report [R-4] • Geotechnical Investigation Results Report [R-25] • REGDOCs, Codes & Standards Report [R-58] • DIQ [R-82] • Preliminary Fire Safe Shutdown Analysis Report [R-89] • Preliminary Fire Hazards Assessment Report [R-90] • Fire Protection System Preliminary Code Compliance Review Report [R-91]

			• Independent Third-Party Review of the Preliminary Fire Protection Design [R-92] • HFEPP [R-81] • Darlington BWRX-300 Security Assessment [R-77] • REGDOC-2.5.2 Compliance Matrix [R-70] • REGDOC-2.5.2 Justification for Alternative Approach • Project Design Plan • Fuel Qualification Report • Fuel Design Reports
4.6	Fitness for service	4.6	N/A
4.7	Radiation protection	4.7	• PSAR [R-6]
4.8	Conventional health and safety	4.8	• DNNP Health and Safety Plan [R-69]
4.9	Environmental protection	4.9 5.2 5.6	• EIS Review Report [R-4] • Commitments Report [R-2] • PSAR [R-6] • PPE Review Report [R-64]
4.10	Emergency management and fire protection	4.10	• PSAR [R-6]
4.11	Waste management	4.11	• PSAR [R-6] • Commitments Report [R-2]
4.12	Security	4.12	• Darlington BWRX-300 Security Assessment [R-77] • TRA Report [R-109]
4.13	Safeguards and non-proliferation	4.13	• DIQ [R-82]
4.14	Packaging and transport	4.14	
5.	**Other Regulatory Areas**		
5.1	Reporting requirements	5.1	
5.2	Public information and disclosure program	5.3	• Stakeholder Engagement Plan [R-68]

5.3	Indigenous engagement	5.4	• Indigenous Engagement Plan [R-126] • Commitments Report [R-2] • Indigenous Engagement Report [R-125]
5.4	Cost recovery and financial guarantees	5.5	
Appendix A	Legislative Clauses	Appendix A	N/A
Appendix B	Safety and Control Areas	Appendix B	N/A
Appendix C	Review Objectives for an Application for a Licence to Construct a Reactor Facility	Appendix C	N/A
Appendix D	Sample Format for Listing the Supporting Documentation	N/A	N/A
Appendix E	Sample Format for Listing Revisions to the Supporting Documentation	N/A	N/A

Appendix C Upcoming Supporting Documents

Document Title	Target Submission Date
Environmental Monitoring and Environment Assessment Follow-up Plan	Q4 2022
Radioactive Waste Management Plan	Q1, 2023
BWRX-300 Occupational Dose Assessment Report	Q1, 2023
PSAR Independent Review Report	Q1, 2023
PDP (Construction Phase)	Q1, 2023
Financial Guarantee (Construction Phase)	Q1, 2023
PDP (As-Built)	Q1, 2023
Construction Management Plan	Q1, 2023
Commissioning and Turn-over Program Management Plan	Q1, 2023
Seismic Investigation Results Report	Q1, 2023