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Darlington NGS Probabilistic Safety Assessment Report

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Revision Summary

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R000	May 2012	Initial Issue
R001	July 2015	Revised for 2015 DARA update
R002	March 2021	Revised for 2020 DARA update

Executive Summary

The objective of Probabilistic Safety Assessment (PSA) at Ontario Power Generation (OPG) Nuclear is to provide an integrated review of the adequacy of the safety of the current station design and operation for each nuclear power station. The station PSAs are required to comply with the Canadian Nuclear Safety Commission (CNSC) Regulatory Document REGDOC-2.4.2 [R-1].

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

The Darlington Nuclear Generating Station (NGS) PSA followed a quality assurance plan consistent with Canadian Standards Association standard CSA N286-12, Management System Requirements for Nuclear Power Plants [R-2]. The PSA used computer programs consistent with Canadian Standards Association standard CSA N286.7-16, Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants [R-3].

The PSA was prepared following methodologies consistent with industry good practices. The OPG PSA Methodologies have been accepted by the CNSC under compliance with REGDOC-2.4.2.

The baseline Darlington NGS PSAs are documented in several reports:

- A hazard screening assessment identifies the hazards that require assessment in a PSA model.
- The Level-1 and Level-2 internal events at-power PSA assesses the risk of severe core damage and radioactive releases from internal events occurring while the reactor is at power; i.e., it considers the challenges to reactor core cooling from accident sequences covering Design Basis Accidents and Beyond Design Basis Accidents including Severe Accidents while the reactor is at full power.
- The internal events outage PSA assesses the risk of severe core damage from internal events occurring while the reactor is in the Guaranteed Shutdown State (GSS); i.e., it considers the challenges to reactor core cooling from accident sequences during unit outages, including loss of shutdown heat sinks. It also provides an estimate of the risk of large release in GSS.
- The seismic PSA assesses the risk of severe core damage from seismic events occurring while the reactor is at full power, and provides an estimate of the risk of large release as a result of seismic events.
- The internal fire PSA assesses the risk of severe core damage and large release from internal fires occurring while the reactor is at full power.

- The internal flooding PSA assesses the risk of severe core damage from internal floods occurring while the reactor is at full power, and a bounding estimate of large release as a result of internal floods.

- The high wind PSA assesses the risk of severe core damage from high wind occurring while the reactor is at full power, and an estimate of large release as a result of high wind events.

- The non-reactor source PSA assesses the risk of radioactive releases from sources other than the reactor core.

The completion of the Darlington PSA shows that the severe core damage frequency and large release frequency for each hazard are less than OPG's safety goals.

1.0 INTRODUCTION

The objective of Probabilistic Safety Assessment (PSA) at Ontario Power Generation (OPG) Nuclear is to provide an integrated review of the adequacy of the safety of the current station design and operation for each nuclear power station. The station PSAs are required to comply with the Canadian Nuclear Safety Commission (CNSC) Regulatory Document REGDOC-2.4.2 [R-1].

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

The PSA for the Darlington Nuclear Generating Station (NGS) or Darlington Risk Assessment is referred to as DARA. The DARA studies provide an estimate of the station risk in its current configuration and are required for compliance with REGDOC-2.4.2. The PSA reflects the current station design and operation, is consistent with the OPG PSA methodology, and is consistent with industry good practices. The OPG PSA Methodologies have been accepted by the CNSC under REGDOC-2.4.2. A separate hazard screening assessment for internal and external events has been completed to confirm that no other identified hazards require detailed assessment in a PSA.

Development of the Darlington NGS PSA followed a quality assurance plan consistent with Canadian Standards Association standard CSA N286-12, Management System Requirements for Nuclear Power Plants [R-2]. The PSA used computer programs consistent with Canadian Standards Association standard CSA N286.7-16, Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants [R-3].

Ontario Power Generation has safety goals for Severe Core Damage[1] Frequency (SCDF) and Large Release[2] Frequency (LRF), Reference [R-4], as shown in Table 1. The intent of these goals is to ensure that the radiological risks arising from nuclear accidents associated with the operation of Ontario Power Generation's nuclear power reactors are low in comparison to risks to which the public is normally exposed. The baseline DARA studies show that the risk from the operation of Darlington NGS is low.

The first Darlington NGS PSA studies for S-294 [R-5] compliance were completed in 2011 and the previous update was completed in 2015. All of the Darlington PSA studies were revised in 2020 as part of the regular update cycle under REGDOC-2.4.2 compliance. The updates included:

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

- Station design, operation, and analysis information up to the study freeze date of December 31, 2018;
- A number of model and documentation enhancements;
- The incorporation of changes in Emergency Mitigating Equipment (EME) design since 2013;
- The incorporation of several Safety Improvement Opportunities (SIOs), which were implemented as part of Darlington NGS refurbishment; and
- The credit of Severe Accident Management Guidelines (SAMG) in the Level 2 PSA.

The current report summarizes the probabilistic safety assessments of the Darlington NGS described above and compares the results with Ontario Power Generation's safety goals as documented in Reference [R-4].

1.1 Objectives

The principal objectives of the DARA Studies are:

(1) To provide an integrated review of the adequacy of the safety of the current station design and operation;

(2) To prepare a risk model in a form that can be used to assist in safety-related decision making; and

(3) To assess risk results and ensure that they are acceptably low.

1.2 Scope

The baseline DARA probabilistic safety assessments are documented in eight separate reports - one hazard screening and seven PSA models, as follows:

(1) A hazard screening assessment for internal and external events, which identifies the hazards that require further detailed analysis in a PSA.

(2) A Level-1 internal events at-power PSA, which studies the risk of severe core damage from internal events (e.g., loss of coolant accidents, steam line breaks) occurring while the reactor is at full power; i.e., it considers the challenges to reactor core cooling from accident sequences covering Design Basis Accidents and Beyond Design Basis Accidents while the reactor is at full power. This report is referred to as DARA-L1P.

(3) A Level-2 internal events at-power PSA (DARA-L2P), which studies the frequency and composition of releases to the environment from severe core damage occurring due to events occurring within the station (e.g., loss of coolant accidents, steam line breaks) while the reactor is at full power. This PSA is the extension of the Level-1 PSA described in Item 2.

(4) An internal events outage PSA (DARA-L1O), which studies the risk of severe core damage from internal events occurring while the reactor is in the GSS; i.e., it considers the challenges to reactor core cooling from accident sequences during unit outages, including loss of shutdown heat sinks; and it provides an estimate of large release frequency as a result of internal events during GSS.

(5) A seismic PSA (DARA-SEISMIC), which studies the risk of severe core damage from seismic events occurring while the reactor is at full power, and provides an estimate of the risk of large release as a result of seismic events (i.e., earthquakes).

(6) An internal fire PSA (DARA-FIRE), which studies the risk of severe core damage and large release as a result of internal fire events (e.g., fires caused by station electrical equipment) occurring while the reactor is at full power.

(7) An internal flooding PSA (DARA-FLOOD), which studies the risk of severe core damage from internal floods (e.g., pipe breaks of plant systems) occurring while the reactor is at full power, and provides a bounding estimate of large release frequency as a result of internal flooding.

(8) A high wind PSA (DARA-WIND), which studies the risk of severe core damage from high wind events (e.g., severe thunderstorms, tornadoes) occurring while the reactor is at full power, and provides an estimate of large release frequency as a result of high wind events.

(9) A non-reactor source PSA, which studies the risk of releases to the environment from non-reactor sources of radioactivity.

The Darlington PSA models (reports 2-9 above) do not cover the following potential sources of risk:

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

These types of hazards are instead addressed through other screening or deterministic hazard studies, see Section 4.0. Consistent with industry practice, wilful acts (e.g., sabotage) are not modelled in the OPG PSAs.

The response of all Darlington NGS units to various initiating events is essentially identical, and it is generally only necessary to model a single unit, with this unit considered representative of all other units despite slight differences in design. Unit 2 was selected as the reference unit.

1.3 Organization of Summary Report

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

(a) A short description of the Darlington NGS station and units (Section 2.0);

(b) An overview of hazard screening method and the internal/external hazard screening assessment (Section 4.0);

(c) An overview of PSA methods and the Level 1 and Level 2 PSA (Section 3.0) and the methods used for Level 1 Analysis (Section 5.0) and Level 2 Analysis (Section 6.0);

(d) A discussion of the main results of the DARA studies (Section 7.0).

Appendix A contains a list of the abbreviations and acronyms used in this summary report.

2.0 PLANT DESCRIPTION

The following subsections provide a short description of the Darlington site and plant.

2.1 Site Arrangement

The Darlington NGS facility consists of four CANDU pressurized heavy water reactor units. The station was designed and constructed in the 1980s to early 1990s, with in-service dates ranging between October 1990 and June 1993. The station has four nuclear reactors, four turbine generators, and associated equipment, services and facilities, shown in Figure 1 and Figure 2. At full power each unit produces 2776 MW(th), generating a net output of 881 MW(e). The electrical output from each reactor-turbine generator set is generated at 22 kV, 60Hz and 0.85 power factor and delivered to the 500 kV switchyard. The turbine-generator set can operate for sustained periods if the reactor power is greater than 30% full power.

Each unit was originally designed and evaluated for a 30-year lifetime. OPG is currently working towards refurbishment of Darlington, which will extend the life of the station to 2055.

Each unit comprises a power source capable of operating independently of the other units with reliance on certain common services. The power generating equipment of each unit is a conventionally steam-driven turbine generator. The associated heat source is a heavy water (D_2O) moderated, pressurized heavy water cooled, natural uranium dioxide fuelled, horizontal pressure tube reactor. This type of nuclear steam supply is used in all electrical nuclear power stations built in the province of Ontario.

2.2 Buildings and Structures

The Darlington NGS contains the following buildings and structures:

(a) Four reactor building structures;

(b) Four reactor auxiliary bays;

(c) A powerhouse comprising four turbine halls, four turbine auxiliary bays, and a central service area;

(d) A vacuum structure;

(e) Four combined cooling and service water pumphouses;

(f) An emergency electrical power and water supply complex, consisting of an Emergency Service Water (ESW) pumphouse, emergency power supply generator sets buildings, emergency power supply fuel management structures, and emergency electrical rooms and associated tunnels;

(g) Two administrative buildings;

(h) A Water Treatment Building;

(i) Two Fuelling Facilities Auxiliary Areas (FFAAs), including two Irradiated Fuel Bays (IFBs);

(j) Two standby generator areas;

(k) A Heavy Water Management Building;

(l) Tritium Removal Facility;

(m) Flammable Material Storage Building;

(n) High-Pressure Gas Cylinder Storage Building;

(o) Sewage Treatment Plant;

(p) Emergency Response Team Facility;

(q) Hazardous Material and D_2O Storage Building;

(r) A Main Security Building and an Auxiliary Security Building;

(s) Darlington Waste Management Facility (DWMF); and

(t) Auxiliary Heating Steam Boiler House.

The general arrangement of the station is shown in Figure 2. The four units at the station are each numbered and referred to as Unit 1, Unit 2, etc. The common equipment is referred to as Unit 0.

The Reactor Building, Figure 3, is a rectangular reinforced-concrete building, which serves as a support and an enclosure for the reactor and some of its associated equipment. The portion

of the Reactor Building, which forms part of the containment envelope, is called the reactor vault.

The fuelling duct, which is connected to each of the reactor vaults, runs the length of the station under the vaults. It serves as a connection between the reactor and the Fuelling Facilities Auxiliary Areas at each end of the duct. A provision for future plant extension has been provided in the end wall of the fuelling duct in the Fuelling Facilities Area (east). A pressure relief duct connects the fuelling duct to the vacuum structure.

The containment envelope comprises the four reactor vaults, the fuelling duct, the pressure relief duct, the pressure relief valve manifold, the vacuum structure, the fuelling machine head removal area, and a fuel handling and service area at each end of the fuelling duct.

Each reactor vault is surrounded by a Reactor Auxiliary Bay. This building contains reactor auxiliaries and secondary circuits of low temperature, pressure, and generally of low radioactivity level. The Reactor Auxiliary Bay consists of a basement with concrete floors below elevation 100 m, and a conventional steel-frame structure with concrete floor slabs above elevation 100 m.

The Central Service Area is divided into the Central Service Area-Nuclear and the Central Service Area-Conventional. The Central Service Area serves the entire station. The Central Service Area-Nuclear contains facilities for fuelling machine head removal, treatment and storage of heavy water, spent ion exchange resins, and active wastes. It is located below grade in the south portion of the Central Service Area and is of reinforced-concrete construction. The Central Service Area-Conventional contains stores, laboratories, electrical, air conditioning equipment and the central control area. For the most part, it is of steel-frame construction with concrete floors. The central control area is located above the Central Service Area-Nuclear and is enclosed on all four sides by reinforced-concrete walls. The control area also has a reinforced-concrete roof.

Column Line 11 between turbine auxiliary building and reactor auxiliary building from elevation 100.0 m to 115.0 m is credited as a steam and flood protection barrier in the event of a secondary side or feed water line break. The wall and door between the RAB and the West FFAA on column line A, between elevations 107.5 m and 115 m are credited as a barrier to prevent steam and water released from a feed water break at 107.5 m el. (south of column line 11) from spilling into the West FFAA at elevations 100 m and 105.7 m.

The emergency electrical power and water supply complex is of reinforced-concrete construction throughout. The other buildings listed are of conventional steel-frame construction on reinforced-concrete foundations.

2.3 Reactor

The reactor consists of a cylindrical, horizontal, single-walled stainless steel vessel called the calandria. It provides containment for the heavy water moderator and reflector. It is axially penetrated by 480 calandria tubes. These tubes surround the pressure tubes, which contain the fuel and heavy water coolant. The calandria, the two end shields, and the shield tank form an integral, multi-compartment structure which contains the heavy water moderator and reflector, and the light water shielding. The end shields and shield tank (filled with light water)

provide part of the building operational shielding, as well as full shielding between the calandria and the reactor vault when the reactor is shutdown (see Figure 3).

2.3.1 Primary Heat Transport System

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

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

2.3.2 Steam and Feedwater System

The main role of the primary heat transport system is to transport the heat generated in the fuel channels to the steam generators. The role of the steam generators is to transfer this heat and boil the light water on the secondary side. The steam generated is then used to drive the turbine generators to convert the thermal energy to electrical power. After passing through the turbine the steam condenses. The condensate is returned via the feedwater (FW) system to the steam generators to continue the process.

2.3.3 Inter-Unit Feedwater Tie System

After an accident, if the normal feedwater supply to the steam generators is unavailable, the Inter-Unit Feedwater Tie (IUFT) system can provide a short-term source of water to the accident-unit steam generators. Along with the safety relief valves, the IUFT can be used to cool the heat transport system. The water is supplied by the feedwater system of an adjacent unit using a header that runs the length of the station. Feedwater supply to IUFT can come from the auxiliary feed pumps in any of the units. The IUFT system is automatically started when the water level in a steam generator drops below a set level.

2.3.4 Steam Generator Emergency Cooling System

The Steam Generator Emergency Cooling System (SGECS) provides an interim water supply to the steam generators following a postulated steam line or nozzle rupture and/or loss of feedwater supply. The automatic injection of SGECS water will maintain the steam generators as effective heat sinks for the heat transport system until such time as the ESW system is available.

SGECS comprises two water tanks and two air accumulators, with associated valves and piping. Each water tank is pressurized by one of the air accumulators and supplies water to two steam generators. The water tanks are filled with demineralized water from the feedwater system.

2.3.5 Steam Relief System

The steam relief system protects the steam generators from overpressure and is also used for rapid cooling of the primary heat transport system when needed. Three types of valves can be used to reject steam from the steam generators: the Atmospheric Steam Discharge Valves (ASDVs), the Condenser Steam Discharge Valves (CSDVs), and the Instrumented Steam Relief Valves (ISRVs). The ASDVs and ISRVs discharge steam into the atmosphere. The CSDVs discharge steam into the condenser, where the steam is condensed and returns to the feed cycle.

2.3.6 Shutdown Cooling System

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

2.3.7 Moderator System

During normal plant operation the moderator system is used to slow the neutrons produced by fission in order to sustain the chain reaction and maintain criticality. Heat is generated in the moderator by the neutrons as they slow down, and energy is transferred to the moderator from the calandria tubes, shell, tubesheets and, reactivity mechanisms. Additionally, a small fraction of the heat produced by the fuel is transferred to the moderator during normal at-power operation. The moderator heat is removed by the Moderator Circulation System that incorporates heat exchangers. After an accident, the moderator can be used as an additional heat sink to remove decay heat from the reactor. This additional heat sink is an important, unique feature of the CANDU reactor design.

2.3.8 Unit Control System

Each unit is operated and controlled independently by a dual Digital Control Computer (DCC) system. Important process variables and devices controlled by the dual computer system include:

(a) Reactivity control devices, which includes the liquid zone control valves, the adjuster, absorber and shut-off rods, and gadolinium poison addition into the moderator;

(b) Primary heat transport pressure and inventory control components such as the D_2O liquid feed and bleed valves, the D_2O steam bleed valves, and the pressurizer heaters;

(c) Steam generator level control system components such as the two large and one small level control valves per steam generator;

(d) Steam generator pressure control components such as the turbine governor valves, the CSDVs and the ASDVs; and

(e) Moderator temperature control system components such as the three temperature control valves in the service water side of the moderator heat exchangers.

2.3.9 Powerhouse Steam Venting System

The Powerhouse Steam Venting System (PSVS) is designed to vent steam from the powerhouse in the event of the secondary side piping failure, minimizing the effect of harsh environment on the equipment located in the powerhouse. The system consists of wall mounted, air and spring operated dampers of louvers located at a lower elevation on the powerhouse north wall and at a high elevation on the Reactor Auxiliary Bay walls, and dampers of gravity ventilators located on the roof of the Turbine Hall. The dampers of the louvers and gravity ventilators open automatically on a high temperature signal. The open flow areas at high elevations provide an escape route for steam, while the make-up air is supplied by the open dampers at the lower elevation.

2.3.10 Special Safety Systems

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

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

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

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

(d) Negative Pressure Containment (NPC) System.

2.3.11 Shutdown System No. 1

The primary method of quickly terminating reactor operation is the release of 32 gravity-drop, spring-assisted, neutron-absorbing shut-off rods. The shut-off rods are housed in 32 assemblies positioned vertically through the reactor core, with the rods themselves above the core during high power operation. The SDS1 system employs an independent, triplicated system which senses the requirement for reactor trip and de-energizes direct current clutches to release all of the shut-off rods into the reactor core.

2.3.12 Shutdown System No. 2

The second method of quickly terminating reactor operation is the rapid injection of neutron-absorbing gadolinium nitrate solution into the bulk moderator through eight horizontal nozzles. The SDS2 employs an independent, triplicated system which senses the requirement for this

rapid shutdown and opens fast-acting helium injection valves to force the gadolinium nitrate poison into the moderator.

The gadolinium nitrate solution is stored in eight tanks, connected to a horizontal injection nozzle in the calandria by stainless steel piping. Helium under pressure is stored in a tank that is isolated from the gadolinium nitrate tanks by a duplicated set of quick-opening valves. Opening of the valves causes the helium to pressurize the poison tanks, forcing the gadolinium nitrate solution through the injection nozzles and into the moderator.

2.3.13 Emergency Coolant Injection System

The Emergency Coolant Injection System (ECIS) automatically provides make-up cooling water to the heat transport system following a postulated Loss-Of-Coolant Accident (LOCA). The system also provides one of the long-term heat sinks for emergency core cooling. The ECIS, with most of its major equipment centralized in the central service area, is designed to serve all four units.

The ECIS does not operate during normal plant operation, but is in a poised standby mode.

For the initial high-pressure Emergency Coolant Injection (ECI) injection, light water coolant is drawn from the injection water storage tank and pumped to the affected unit. Upon depletion of the water stored in the injection water storage tank, a recovery mode (long-term injection) is established manually. During this long-term injection phase, a mixture of light (ECI) water and heavy (heat transport) water is drawn from the recovery sump in the pressure relief duct and is recirculated to the affected heat transport system. The Post-Accident Water Cooling System (PAWCS) can be used to cool the recirculated water, providing a long term heat sink.

2.3.14 Containment Systems

The containment system is a special safety system that forms an envelope around the nuclear components of the reactor and the reactor coolant system. It is composed of a number of systems and subsystems whose collective purpose is to prevent a significant release of radioactive material, which may be present in the containment atmosphere following certain postulated accident conditions, to the outside environment. The physical barrier, which minimizes the outflow of radioactive material, is called the containment envelope, and the system whose main purpose is to prevent the design pressure of the containment envelope from being exceeded following an accident is called the containment system. The containment system includes provisions for controlling and maintaining a negative pressure within the containment envelope before and after accidents. The containment system quickly reduces the containment pressure to a subatmospheric level following a large energy release within containment and, hence, minimizes uncontrolled releases to the outside environment. Containment includes an Emergency Filtered Air Discharge System (EFADS) to maintain containment at a sub-atmospheric pressure in the long term following a design basis accident, while providing a filtered discharge path to minimize long-term radioactive releases to the environment. Containment also includes a Containment Filtered Venting System (CFVS) which provides protection of containment against the potential of slow over-pressurization failure and reduces radioactive release to the atmosphere in the event of a Beyond Design Basis Accident (BDBA) or severe accident.

2.3.15 Support Systems

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

2.3.15.1 Electrical Power Systems

The electrical system of the Darlington NGS is designed to satisfy the high reliability requirements of nuclear systems. The design features dual (odd and even) bus arrangements for both unit and common systems, high capacity standby power supplies, and ample redundancy in equipment. There are four distinct classes of power (Classes IV, III, II, and I), as well as the Emergency Power System (EPS).

Class IV power is the main site electrical power supplied from a combination of the provincial electrical grid and the station generating unit transformers; Class III power is typically supplied by Class IV power, but has backup supplies and includes four standby generators; Class II is an AC power system to supply control and monitoring systems and is supplied by Class I power via inverters; Class I is a DC power system to supply control and monitoring system. Class I has battery backup supplies.

EPS is a separate power system consisting of its own on-site power generation (three Emergency Power Generators (EPGs)) and AC and DC distribution systems whose normal supply is from the Class III power system. The purpose of the EPS system is to provide power to selected safety-related loads following events postulated to impact the normal Class IV / III / II / I power distribution, including events that impact more than one unit.

2.3.15.2 Service Water Systems

The service water systems provide cooling water for various loads. The service water systems for Darlington NGS consist of:

(a) Low Pressure Service Water System: Each unit has a Low Pressure Service Water (LPSW) system taking untreated lake water from the forebay. This water is used to cool loads at low elevations. After passing through the various loads, the water is returned to the lake via the condenser cooling water discharge duct.

(b) Powerhouse Upper Level Service Water system: The Powerhouse Upper Level Service Water (PULSW) system supplies tempered water of 10°C in winter and untempered lake water in summer from the LPSW system to various continuously used equipment. This system serves all loads where potential heavy water freezing is a problem, as well as loads located at high elevations in the reactor building that are beyond the maximum pressure available from the LPSW system.

(c) Recirculated Cooling Water System: The Recirculated Cooling Water (RCW) system is a unitized closed loop system which supplies demineralized water to continuously used equipment. This system supplies cooling water to certain vital equipment requiring treated water, at a temperature above the freezing point of heavy water, at a pressure

sufficiently high to prevent localized boiling in certain heat exchangers, and of a quality sufficiently high to minimize corrosion, fouling, and activation by radiation.

(d) Emergency Service Water System: The Emergency Service Water (ESW) system is independent and physically separated from the normal water systems. It is primarily used to supply cooling water to essential safety-related loads when normal service water supplies are unavailable. One ESW system supplies the required loads for all four units. In order for this system to not remain dormant for long periods of time, it is used to supply the normal requirements of the IFB heat exchangers, secondary control areas (Group 2 ventilation), the Auxiliary Service Water System, and the fire water supply.

(e) Circulating Water System: The circulating water system is an open loop system to supply cooling water to the condensers to maintain the design backpressure of the turbine exhaust during full load operation. The circulating water is discharged back to the lake through the discharge duct.

(f) Auxiliary Service Water System: The auxiliary service water system supplies water for cooling purposes in the Central Service Area and other common areas. The system is supplied from the ESW system.

(g) Demineralized Water System: This system supplies make-up water to systems using demineralized water including RCW and the condensate make-up system.

(h) Domestic Water System: This system supplies hot and cold potable water to domestic fixtures in the station including the drinking fountains, showers, washrooms, and kitchens.

Failures of the auxiliary service water system, the demineralized water system and the domestic water system are not analyzed in detail as part of the PSA assessment.

2.3.15.3 Instrument Air Systems

The instrument air (IA) supply is a support system providing filtered and dry compressed air. This compressed air is used for various plant activities including operating valves and inflating airlock seals. Each unit has its own air supply, with certain key loads supplied by backup air from bottles, to ensure operability in the event of failure of the normal supply. On loss of unit instrument air, instrument air supply from another donor unit can be valved in manually via an inter-unit tie.

In addition, the station has a common instrument air system to supply the central service area, FFAAs, vacuum structure, pumphouses, water treatment building, heavy water management building, and ESW pumphouse.

The service air system supplies compressed air to all areas in the station including the service area and other buildings. In addition, the service air system supplies the air requirements of the common instrument air system.

2.3.15.4 Powerhouse Ventilation System

The powerhouse ventilation system provides heating and cooling to the station buildings. Failures of this system are studied for the steam protected rooms in the powerhouse, reactor auxiliary bay and reactor building. Failure of the cooling and ventilation in these rooms may result in equipment failures in the support or mitigating systems.

2.3.15.5 Emergency Mitigating Equipment

As a result of the Fukushima event, OPG has implemented Emergency Mitigating Equipment (EME) for Darlington NGS. The EME was designed to cope with a total loss of heat sink caused by an extended loss of all AC power. EME also provides an additional potential mitigating function for a variety of accident sequences considered in the DARA studies that involve a total loss of heat sinks due to other causes.

The intent of EME is to restore selected reactor cooling and monitoring functions as much as possible using temporarily installed and portable equipment.

EME response is provided in two phases:

- Phase 1 via on-site rapidly deployable mobile equipment to restore selected reactor cooling and monitoring functions and to protect containment.
- Phase 2 via three 4 kV portable diesel driven generators stored at Pickering NGS to energize one Unit 0 4 kV EPS bus to restore specified Unit 0 and Unit EPS loads and to provide additional methods to protect containment.

Phase 1 EME consists of:

(a) EME Generator: A 150 kW 600V diesel generator.

(b) 2 large diesel pumps, each with suction, discharge hose and manifold to supply:

  (1) Steam generators;

  (2) Moderator.

  (3) Heat transport system;

  (4) End shield cooling; and

  (5) Irradiated fuel bays.

(c) Portable Uninterruptable Power Supplies (PUPSs): provided for each unit to power instruments for essential EME parameters if EPS and Class I batteries fail before connection of the Phase 1 EME diesel generator.

(d) Portable Instrumentation: Pressure Gauges (PGs) and associated connection fittings to allow monitoring of EME parameters if for any reason the normally installed instrumentation is unavailable (power cannot be restored, instruments failed, etc.).

(e) Portable Compressors: Two portable diesel driven compressors to provide a means of maintaining airlock seal integrity in the event of extended loss of AC power (loss of Class III and Class IV power where EPS is unable to restore power) by tying into the emergency backup bottle and airlock distribution panel locations for the airlock seals via quick connect fittings.

(f) Telecommunications Trailers: Specialized telecommunication equipment and satellite telephones dedicated for use during an emergency on site are distributed at key locations throughout the station.

The portable Phase 1 EME equipment would be moved from its storage location on site to pre-determined locations in the plant and connected to the designated tie-in points.

Phase 2 EME consists of:

(a) Three 4 kV portable diesel driven Generators deployed from Pickering NGS and staged on the Darlington site within 12 hours of the initiating event.

(1) The Generators are connected in parallel for tie in to the EPS buses.

(2) The Unit 0 4 kV bus will energize specified Unit 0 and Unit EPS 600 V and low voltage AC and DC buses. The loads will include one ESW pump, one Low Pressure Emergency Coolant Injection (LPECI) pump, EFADS, Vault Coolers, and a limited set of Unit 0 and Unit Group 2 equipment.

(3) Each Generator has a maximum capacity of 1.4 MW and the parallel connected Generator Set has a nominal rated capacity of 3 MW. The Generator Set digital controller has full synchronizing and load sharing capability.

After transition to Phase 2 EME power, station ESW will supply steam generator and moderator makeup, vault cooling, and PAWCS as required. The Phase I EME diesel driven pumps will continue to supply End Shield Cooling (ESC), IFB, and primary Heat Transport System makeup as required (unless primary HTS makeup via LPECI has been established).

2.4 Two-Group Separation

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

The following systems are Group 1:

- SDS1: Shutdown System No. 1

- SDC: Shutdown Cooling
- IUFT: Interunit Feedwater Tie
- FW: Feedwater
- Class IV, III, II, I Electrical Power
- Instrument air (normal distribution)

The Group 1 control functions are performed from the Main Control Room (MCR).

The following systems are Group 2:

- SDS2: Shutdown System No. 2
- ISRVs: Instrumented Steam Relief Valves
- EPS: Emergency Power Supply
- SGECS: Steam Generator Emergency Cooling System
- ESW: Emergency Service Water
- ECI, PAWCS: Emergency Coolant Injection and Post-Accident Water Cooling System
- Containment
- EFADS: Emergency Filtered Air Discharge System
- CFVS: Containment Filtered Venting System

The Group 2 systems are seismically qualified to withstand a Design Basis Earthquake (DBE) and designed to withstand the severe atmospheric conditions created by the design basis tornado. The Group 2 control functions are performed from secondary control areas.

3.0 OVERVIEW OF PSA METHODS

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

Risk provides a means of quantifying the degree of safety inherent in a potentially hazardous activity as well as a common basis for comparing the relative safety of dissimilar types of activities and industrial processes. One of the principles of the probabilistic safety assessment

process is that the larger the numerical value of risk for a particular event or combination of events, the more important the event is to safety. Thus, measures to reduce calculated risk improve the level of safety. Probabilistic Safety Assessment represents the process by which risk is quantified, leading to the identification of the dominant contributors to risk. If necessary, the dominant contributors can be used to create strategies to reduce risk and improve safety.

For a nuclear generating plant, the events studied are those leading to damage to fuel both in the core and out of core or releases of radioisotopes into the environment. Consistent with the requirements of REGDOC-2.4.2 [R-1], Ontario Power Generation has completed hazard screening, Level 1 and Level 2 PSA to assess the risk from Darlington NGS:

- A hazard screening assessment was performed to confirm which hazards can be screened out from probabilistic safety assessment, and identify which hazards need to be assessed by a PSA.

- Level 1 of the PSA assesses the frequency of varying degrees of fuel failures, which lead to release of radioactivity into containment.

- Level 2 of the PSA assesses the frequency and magnitude of the release of this radioactivity from containment to the outside environment.

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

Level 1 probabilistic safety assessments have been prepared for full reactor power operation for the following types of initiating events based on the hazard screening results:

- Internal initiating events (e.g., steam line break, loss of coolant accidents);

- Seismic events;

- Internal Fire (fires initiated by in plant sources, e.g., electrical equipment);

- Internal flooding (floods originating from water sources internal to the plant); and

- High winds (including both straight line winds and tornadoes).

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

An outage seismic PSA was not performed as the risk from a seismic event while a single unit is shutdown is bounded by the risk from seismic event while all units are at high power. The accident progression is slower when the unit is in outage, giving more time for operator action; and the time at risk while the unit is in outage is small compared to the time at-power. Thus, the risk is smaller for outage.

An outage high wind PSA was not performed as the risk from a high wind while a single unit is shutdown is bounded by the risk from high wind event while all units are at high power. The accident progression is slower when the unit is in outage, giving more time for operator action;

and the time at risk while the unit is in outage is small compared to the time at-power. Thus, the risk is smaller for outage.

An outage internal fire PSA was not performed as the overall risk of severe core damage due to fire while the unit is at-power is low; the time at risk during an outage is small; and the risk management controls during outage limit the risk of an internal fire.

An outage internal flood PSA was not done as the overall risk of Severe Core Damage (SCD) due to flooding is low. The low risk of SCD due to flooding is due to the low initiating event frequency, the physical separation of the Group 1 and Group 2 systems and the separation of odd and even equipment. As these factors are the same from both at-power and outage operation, a low at-power risk of SCD implies the outage risk will also be low.

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

- The Level 2 assessment for seismic events considers the likelihood of consequential failure of containment due to an earthquake, and then provides a bounding assessment of large release frequency due to seismic failure modes of containment following severe core damage caused by a seismic event.

- The Level 2 assessment of outage internal events reviews the potential for unique containment challenges or bypass pathways in the outage state caused by severe core damage from an internal initiating event occurring while the reactor is in the GSS.

- For the Level 2 assessment of fire events, a Level 2 fire PSA model was developed based on the Level 2 internal events and quantified to provide an estimate of large release frequency.

- Level 2 assessment for internal flooding considered flooding events inside and outside containment involving a single unit, flooding events involving 2 units and more than 2 units that will lead to SCD. Based on this assessment, a large release frequency estimate can be produced.

- The Level 2 high wind assessment considered the potential failure of containment systems due to wind impacts. Large release frequency is then estimated based on the Level 2 model which has been updated to include the impacts of the wind hazards.

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

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

4.0 HAZARD SCREENING METHODS

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

4.1 External Hazards Screening for Reactor Sources

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

4.1.1 Overview of External Hazards Screening Method

The external hazards screening method involves three main steps:

(1) Identify all the external hazards applicable to the site.

(2) Determine consequences of hazards and accident scenarios. Screen-out events qualitatively, based on the consequence of events.

(3) Determine likelihood of event occurring. Screen-out events quantitatively, based on the likelihood of event occurring.

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

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

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

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

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

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

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

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

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

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

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

At this stage of the screening, all qualitative criteria are examined and if the hazard still has not been screened out by any of the seven deterministic criteria, quantitative screening would be required. The OPG PSA Guide for External Hazard Screening recommends using the criteria for quantitatively screening of external events, as shown in Table 2.

Once a hazard has been subject to all qualitative and quantitative criteria, and it is not screened out, then a more detailed assessment using PSA is recommended.

4.1.2 Human-Induced External Hazards

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

4.1.3 Natural External Hazards

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

(a) Canadian and International regulations and standards, and

(b) Information on credible hazards at the plant site.

(c) Or alternatively, the RLC can be established for the corresponding screening frequency.

PSA screening analysis for natural external hazards was conducted in accordance with the methodology described in Section 4.1.1. A set of RLCs were defined and used in the screening analysis. Among the twenty-two natural hazards assessed, all of them were screened out, except earthquake, tornado, and high wind as they may cause some potential damages to certain SSCs, which may have impact on Group 2 systems. A list of the natural external hazards considered is presented in Table 4. Seismic and high wind (including straight-line winds and tornados) PSA assessments were performed; see details in Section 5.5 and Section 5.6, respectively. In addition, the hazards ice-storms, extreme temperatures and geomagnetic storms and solar flares are already accounted for in the internal events PSA (see Section 5.1).

4.1.4 Combined External Hazards

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

4.2 External Hazards Screening for Non-Reactor Sources - IFB

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

- Loss of IFB heat sink
- Loss of IFB water Inventory

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

- Reduced IFB water level can result in increased radiation fields in and near the IFB with the potential to inhibit corrective operator field actions such as equipment repair and IFB inventory make-up.

- Boiling of IFB water can result in harsh environment with the potential to cause IFB equipment failure and inhibit corrective operator field actions.

- IFB inventory leakage events (e.g., pipe break) can cause IFB equipment failure and inhibit corrective operator field actions

4.2.1 Human-Induced External Hazards

The methodology used for screening the human-induced external hazards for IFB is the same as described in Section 4.1.1. All human-induced hazards are screened out, and do not require a PSA. A list of the human-induced external hazards assessed is presented in Table 5.

4.2.2 Natural External Hazards

A list of natural external hazards were assessed through a hazard screening assessment to determine if they are applicable to the IFB at the Darlington NGS.

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

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

4.2.3 Combined External Hazards

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

4.3 External Hazards Screening for Non-Reactor Sources - UFDS

Once the fuel has resided in the irradiated fuel bays for a minimum of ten years, the residual decay heat is sufficiently low to allow this fuel to be moved to dry storage. The Used Fuel Dry Storage (UFDS) process operations can be broken down into six parts, which are:

- Receipt of empty Dry Storage Containers (DSCs);

- Prepare empty DSC for loading;

- Transfer operation;

- Fuel loading operation at Fuelling Facilities Auxiliary Areas (FFAAs);

- DSC processing operations at the DWMF; and
- Interim storage of DSCs in the storage buildings.

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

4.3.1 Human-Induced External Hazards

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

4.3.2 Natural External Hazards

Table 8 lists the natural external hazards and provides their screening analysis based on the approach adopted for the analysis of the human-induced hazards in Section 4.2.2. All natural external hazards are screened out, and do not require a PSA.

4.3.3 Combined External Hazards

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

4.4 Internal Hazards Screening for Reactor Sources

4.4.1 Overview of Internal Hazards Screening Method

The internal hazards screening method is similar to the external hazards screening method and involves three main steps:

(1) Identify all the internal hazards applicable to the site.

(2) Determine consequences of hazards and accident scenarios. Screen-out events qualitatively, based on the consequence of events.

(3) Determine likelihood of event occurring. Screen-out events quantitatively, based on the likelihood of event occurring.

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

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

4.4.2 Internal Hazards Screening Results

The internal hazards identification included mechanical, chemical, electrical hazards, initiated from the inside of the plant (such as turbine missiles, load drops, accidental release of chemicals, and electromagnetic interferences). The internal hazards identified are listed below:

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

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

4.5 Internal Hazards Screening for Non-Reactor Sources - IFB

Identification of internal hazards for IFB is based on the OPG PSA Guide for Internal Hazard Screening which classifies all the internal hazards as leading to the following bounding consequences:

- Loss of IFB Heat Sink (resulting from, e.g., random IFB cooling and support system failures, human errors, internal IFB fires, internal IFB flooding, reactor hazards that may impact IFB cooling system equipment operation); or

- Rapid/Slow Loss of IFB Inventory (resulting from, e.g., random IFB piping breaks, loss of inventory make-up, damage due to heavy load drops). In principle the same hazard types are considered as for the existing reactor PSAs.

The internal hazards are identified as follows:

- Loss of heat sink:
    - Random IFB cooling system failures (e.g., pumps, flow path, valving, control logic, etc.);
    - Random IFB support systems failures (e.g., power, air, water supply failure);
    - Human errors (e.g., due to maintenance and testing);
    - Internal IFB fires;
    - Internal IFB flooding;
    - Loss of IFB water Inventory
- Loss of IFB water inventory
    - Rapid loss of IFB water inventory;
    - Slow loss of IFB water inventory;
    - Damage due to heavy load drops from craning accidents.
- Loss of IFB make-up water
- Turbine generated missile
- Criticality
- Reactor events, e.g., secondary side line break (SSLBs) and LOCAs outside containment
- Reactor events leading to core damage
- Reactor events leading to containment failure causing a large release

The internal hazards that were not screened out were assessed further as part of the non-reactor PSA (see Section 6.13).

4.6 Internal Hazards Screening for Non-Reactor Sources – UFDS

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

The internal hazards are identified as follows:

- Mechanical missile impact;
- Explosions within the generating station main buildings;
- Release of oxidizing, toxic, radioactive or corrosive gases and liquids from onsite storage;
- Release of stored energy;
- Dropped or impacting loads
- Transportation impact (e.g., vehicles, movement of toxic on-site goods);
- Criticality
- Loss of support services to the UFDS (e.g. Electrical Power, Service Air, Heat, Ventilation and Air Conditioning (HVAC), etc.)
- Fires
- Electromagnetic interference; and
- Static electricity.

All internal hazards with the potential to impact UFDS have been screened out and do not require a PSA.

5.0 LEVEL 1 PSA METHODS

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

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

The Level 1 and Level 2 At-Power PSA models were used to aid in the development and quantification of the internal events outage, seismic, fire, internal flooding, and high wind PSAs.

5.1 Level 1 At-Power Internal Events

The Level 1 At-Power Internal Events PSA for Darlington NGS has been developed following the methodology for preparation of a Level-1 At-Power PSA as described in the Internal Events At-Power PSA Guide.

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

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

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

(c) Development of event trees. Event trees (ETs) are a tool that establishes what consequences can occur from a particular initiating event, given success or failure of the systems credited with mitigating the initiating event. Development of the DARA event trees is discussed in Section 5.1.3.

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

(e) Development of a component reliability database with, to the extent possible, information specific to Darlington NGS. The reliability database is needed to support the fault tree analysis mentioned above. The sources for the data in the component reliability database are discussed in Section 5.1.4.

(f) Assessment of the effect of human error on system performance using Human Reliability Analysis (HRA). The potential for human errors must be incorporated along with hardware failures in the system level fault trees and event trees, and the human error probabilities systematically estimated and assigned. Human errors are referred to as “human interactions” in DARA. The HRA is discussed in Section 5.1.5.

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

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

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

5.1.1 Initiating Events Identification and Quantification

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

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

Although DARA-L1P is an internal events PSA, it does include events associated with loss of off-site power (loss of the bulk electrical system) and events leading to failures in the service water intake. This is consistent with standard practice in PSA for nuclear power plants.

The objective of the initiating event selection task was to obtain as complete coverage as possible of credible initiating events. To create the initiating event list, past Ontario Power Generation probabilistic safety assessments were reviewed, as were the plant operating experience and station condition records, and other published PSAs. In addition, insight from the fault tree modelling, discussed in Section 5.1.4, identified other initiating events.

The complete list of initiating events considered in DARA-L1P is provided in Table 9.

The initiating events are quantified primarily using Bayes' Theorem. In a Bayesian approach, an assessment is made of generic (prior) experience that is then updated by station-specific experience. This technique allows general experience and knowledge about a given event to be combined with actual operating experience gained with the station under study. It is especially useful for quantifying the frequency of events unlikely to be experienced within the lifetime of a single station. This is the industry standard method.

5.1.2 Fuel Damage Categorization Scheme

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

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

(a) Sequences with the potential for loss of core structural integrity (severe core damage).

(b) Loss of fuel cooling requiring the moderator as a heat sink.

(c) Prolonged loss of heat sink.

(d) Inadequate cooling to fuel in one or more core passes following a large loss-of-coolant accident with successful Emergency Cooling Injection System initiation.

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

(f) Loss of Heat Transport System integrity followed by successful ECI initiation with no significant fuel damage.

The lower consequence threshold for significance is deemed to be the occurrence of a loss of heat transport system integrity resulting in ECI initiation. Although fuel damage is not likely, the event is considered to have the potential for significant economic consequence due to the downgrading of heavy water, and the loss of revenue due to prolonged shutdown of the accident unit. At the other extreme are the unlikely events that have the potential for severe consequences involving the loss of core structural integrity. Table 10 presents the FDCs used in DARA. These FDCs are also used to calculate the frequency of severe core damage, used for comparison to the relevant Ontario Power Generation safety goal. Severe core damage is defined to be the sum of the FDC1 and FDC2 frequencies.

5.1.3 Event Tree Analysis

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

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

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

Event tree analysis requires the following to be predefined:

(a) A list of initiating events to be considered.

(b) Definition of sequence end states.

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

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

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

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

5.1.4 Fault Tree Analysis

A fault tree (FT) is a logic diagram that models the possible causes of a particular fault, usually a system failure, and is used to calculate the probability that the fault occurs. In DARA, fault trees are used to quantify the probability of the failure of the mitigating systems that appear in the event trees discussed in Section 5.1.3, and for the support systems. Table 11 lists the systems modelled by fault trees in the DARA-L1P study. Figure 6 depicts the relationship between the event trees and fault trees. System fault tree analysis is used to calculate the probability of an event tree branch point given a specific set of events that fail the system.

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

The “CAFTA” software code is used for developing and quantifying the fault tree [R-6].

For example, consider emergency make-up water to the steam generators. For this system, the failure mode of interest might be “fails to supply adequate water to the steam generator when required”. Figure 7 shows a partially completed fault tree with this event at the top. Starting from this top event, the fault tree analyst poses the question “How can this event occur?”. The answers to this question become the inputs to this top event. For example, Figure 7 shows that ESW to the steam generators can fail if the piping fails due to water hammer, or if there is no flow from check valve NV42. For each of these contributors, the process of examining how they can occur is repeated, until no further insights can be obtained about the behaviour of the system. Typically, the fault tree is developed either to predefined system boundaries, or to the individual system components.

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

(a) System capability: For example, how much water flow is required for the steam generator to be a successful heat sink?

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

(c) Common cause failures: For example, if a pump has failed due to any number of causes will any of the remaining redundant pumps fail to operate due to the same cause of failure as the first?

(d) Failure criteria: For example, what fundamental failure modes lead to failure of ESW to the steam generators?

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

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

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

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

The nuclear industry has adopted a Bayesian approach for obtaining component failure rates. The Bayesian approach is based on the use of both the “prior knowledge” and the plant-specific data in deriving the failure rates. Three industry sources, U.S. Nuclear Regulatory Commission (NRC) [R-7], T-book [R-8], and Westinghouse Savannah River Company [R-9], were used for obtaining generic data. The DARA component reliability database is based on a Bayesian calculation of the equipment failure rates reflecting Darlington operational data from 1999 to 2018 inclusive.

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

5.1.5 Human Reliability Analysis

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

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

In principle, every piece of equipment or system in the plant is susceptible to failure because of human error; however, human errors that contribute directly to the failure of individual components are included in the equipment reliability database (i.e., reflected in the component failure rate) and need not be identified in fault trees. The human errors of interest to the fault tree analyst arise under five sets of circumstances:

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

(b) Where an annunciated equipment or system failure occurs but this failure is not responded to by the operator prior to an initiating event;

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

(d) Where an operator can fail to respond appropriately to bring the plant to a stable state following an initiating event (by not taking any action at all or taking the required action but in an inappropriate way); and

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

A human interaction in a fault tree identifies an opportunity for a human to make an error. Only those opportunities that arise in carrying out established plant operating practice are included; specifically, errors made during maintenance, testing, normal plant control, and post-initiating

event control and recovery activities. In most cases, these errors would be made while carrying out formal procedures. Random, spurious, wilful, or vengeful actions are not included.

In order to systematically quantify the human interactions in DARA, Ontario Power Generation uses a human interaction taxonomy. This taxonomy classifies the human interactions in DARA-L1P into three parts: Part 1 contains the simple interactions that, by definition, occur prior to an initiating event; Part 2 contains complex human interactions that occur prior to initiating events; and Part 3 contains the complex interactions that occur after an initiating event.

Simple human interactions have the following characteristics:

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

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

(c) They occur prior to an initiating event.

The task of assigning preliminary (screening) human error probabilities for the simple human interactions is made easier and faster using a simple method requiring only selection of an unmodified basic Human Error Probability (HEP) and predefined modifying factors. This method quantifies the human interaction based on the type of task, the location where the task is performed, whether the error can be detected in the main control room, and if any annunciations or inspections can detect the error. The simple human interactions are reviewed by the Human Reliability Assessment (HRA) Specialist. In some cases, the probability is requantified using the Technique for Human Error Rate Prediction (THERP) described in Reference [R-10].

For the complex human interactions that occur prior to initiating events, the same process may be followed to obtain a preliminary (screening) quantification. These human interactions are complex because they include system-level functions that involve more than just direct physical manipulation of a component, such as the setting of computer control program parameters or modes. The preliminary quantifications are then reviewed by the HRA Specialist on a case-by-case basis and if required are requantified using THERP methodology described in Reference [R-10].

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

For the post-initiating event complex human interactions, the preliminary (screening) human error probabilities are assigned based on three criteria: whether the task is straightforward, of average complexity, or very complex; the time available; and the quality of indication available

in the main control room to indicate that action is required. The preliminary quantifications are then reviewed by the HRA Specialist. Like the pre-initiating event complex human interactions, in some cases these probabilities are requantified using THERP methodology described in Reference [R-10].

5.1.6 Fault Tree Integration and Evaluation

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

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

The CAFTA software stores and evaluates the fault trees [R-6]. The CAFTA program was developed by Electric Power Research Institute (EPRI). The FTREX program is used to quantify the results [R-11].

The solution of a fault tree is a listing of the combination of an initiating event, equipment failures, and human errors that leads to the occurrence of the fault tree top event, with each combination containing the minimum number of failures that have to occur to cause the top event. Such combinations are also called minimal cutsets.

The solution of the fault tree calculated using CAFTA is truncated. That is to say, contributors below a certain frequency are not included in the solution. Truncation is necessary because of computational limits. The truncation level selected should be low enough that all significant contributors are captured. The Level 1 At-Power PSA Guide recommends that the solution of the integrated fault tree for each FDC be truncated at either 4 orders of magnitude below the most likely minimal cutset in that FDC or at 1E-12 occ/yr, whichever is the highest. For FDC2, the top cutset frequency is in the 3E-08 occ/yr range, so a truncation of approximately 3E-12 occ/yr is used.

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

- Importance analysis to identify systems and components that are important to the FDC results;

- Parametric uncertainty analysis to determine the lower and upper limits of the two-sided 90% confidence interval for the frequency of each FDC; and

- Sensitivity analysis used to evaluate the impact on the results of a number of assumptions made in the event tree analysis and fault tree analysis, as well as assumptions impacting the quantification of initiating events, undeveloped events, and human error events.

Recall from Section 3.0 that risk has two components: the frequency of occurrence and the consequences. Section 5.1.1 to 5.1.6 describe the methods used to quantify the frequency of occurrence of the fuel damage categories. The Level 1 analysis is used as an input to the Level 2 analysis (see Section 6.0). The remaining subsections in Section 5.0 describe the differences in methodology for Level 1 assessment for the outage state, for internal fire, internal flood, seismic, and high wind initiating events.

5.2 Outage Internal Events

DARA-L1P considers internal events occurring at 100% full power operation. However, the Darlington NGS has periods of planned outage to perform routine maintenance and testing that cannot be done during full power operation. Typically, a unit has a planned outage for less than 10% of the operating cycle. The reactor power continues to decrease exponentially after reactor trip. Reactor power is typically around 0.6% full power on the first day of an outage.

The 2020 DARA-L1O assessment was developed following the methodology for preparation of a Level-1 Outage PSA as described in the OPG Outage PSA Guide. The 2011 model and the 2015 model were used as the basis for developing the 2020 bounding assessment described in Section 5.2.8.

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

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

5.2.1 Plant Operational State (POS) Identification and Analysis

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

Firstly, Pre-Plant Operational States (Pre-POSs) are identified; Pre-POSs are defined as unique outage plant configurations wherein all parameters of interest (system configuration and parameters, e.g., heat transport system pressure, primary heat sink, HTS pressure) are considered stable for the duration of the state. Pre-POSs are the highest resolution of the outage states. The Pre-POSs are grouped into POSs. For DARA-L1O, eight pre-POSs were identified and have been grouped into three representative POSs. The three POSs are used in other aspects of the Outage PSA, including accident sequence analysis using event trees. Table 12 provides a summary of the final POSs used in the DARA-L1O model. The parameters used to define the POSs are listed in the leftmost column.

5.2.2 Initiating Event Identification and Quantification

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

The process described below was used to identify, group and quantify outage state initiating events:

- The outage IE identification process uses a number of different steps and different sources of information, so that the basis for the Outage PSA is as comprehensive as possible.

- The identified IEs are grouped on the basis of similar mitigation requirements, in order to simplify the accident sequence analysis.

- The frequency of occurrence of each initiating event (or IE group) is estimated, so that the overall risk of core damage can be calculated.

Table 13 presents the list of outage initiating events for the Darlington NGS Level 1 Outage PSA, and which POS each initiating event can occur in. Some initiating events can occur only in specific plant configurations. For example, ice-plugs are used during some maintenance activities on the HTS, but can only be used while the HTS is depressurized. So the ice-plug failure initiating event can only occur during the POSs with a depressurized HTS (POSC and POSD).

5.2.3 Outage Event Tree Analysis and Fuel Damage Category (FDC) Analysis

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

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

(1) For each unique IE/POS combination, identify the mitigating systems credited for the IE based on a review of the accident analysis and plant operating procedures.

(2) Determine the end states of interest in the ET analysis. For DARA-L1O, these are the outage fuel damage categories as shown in Table 14.

(3) Develop the accident sequence logic depending on the success and failure of the mitigating functions credited for the IE.

(4) Add the branch point label for each mitigating system failure as the logic is being developed on the failure branch of the ET.

(5) Assign a FDC to each ET sequence end state.

5.2.4 Outage System Fault Tree Analysis

The fault tree analysis process for the internal outage PSA is the same as for the at-power PSA. However, the fault tree models are significantly different to reflect the outage configurations of the system.

The system FT models are specific to the outage PSA. Each fault tree includes a brief overview of the system analyzed, top event definitions, assumptions, failure criteria, FT diagram, data table, results expressed as minimal cutsets, system failure probability and importance indices. Table 11 lists the systems modelled by fault trees in DARA-L1O.

5.2.5 Reliability Data Analysis

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

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

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

5.2.6 Human Reliability Analysis

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

combined with system and/or equipment failures in the ET to provide accident sequence frequencies.

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

5.2.7 Model Integration, Quantification, and Additional Analyses

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

For each consequence category, the magnitude of the associated consequence needs to be calculated. The product of frequency and consequence is calculated for each category and summed to obtain an overall estimate of risk. These are used in absolute terms to assess the overall safety design adequacy, and in relative terms to identify the dominant risk contributors. The acceptability of the Darlington NGS risk estimates is judged based on comparison with the safety goals established by OPG [R-4].

Similar to the At-Power PSA, additional elements (see Section 5.1.6) supplement the baseline solution in order to gain an additional understanding of the station risk.

5.2.8 DARA-L1O 2020 Bounding Assessment

The 2020 DARA-L1O update is a bounding assessment, undertaken in accordance with the principle in REGDOC-2.4.2 that the level of detail in a PSA should be consistent with the level of risk. It was prepared in accordance with the OPG Level 1 Outage PSA Guide.

The overall objective of 2020 DARA-L1O analysis was to provide an updated severe core damage frequency (SCDF) estimate for 2020 reflecting the current Darlington design and operation to the extent practical for a limited scope bounding assessment. This has been accomplished as follows:

(1) A full scope quantitative update was completed for the outage Plant Operational State (POS) parameters (as described in Section 5.2.1), outage initiating event (IE) frequencies (as described in Section 5.2.2), component failure rates, and frequencies of planned test and maintenance procedures (described in Section 5.2.3), based on the incorporation of recent Darlington NGS experience up to the study freeze date of December 31, 2018.

(2) The potential impact of event tree and fault tree model changes from the 2020 DARA-L1P study was reviewed, with applicable changes made to the Level 1 Outage event trees and fault trees.

(3) The integrated DARA-L1O model was constructed from the updated outage event trees and fault trees.

(4) The integrated DARA-L1O model has been requantified in order to obtain a revised set of baseline cutsets for severe core damage.

5.3 At-Power Internal Fire

The 2020 DARA-FIRE assessment was developed following the methodology for preparation of an Internal Fire PSA as described in the OPG Fire PSA Guide. The 2020 model and analysis are significantly altered from the 2015 DARA-Fire, with fire damage consequence assessments performed using explicitly selected equipment and cable routing information.

The OPG Fire PSA Guide has been developed based on NUREG/CR-6850 [R-12]. The major activities of the Fire PSA methodology and its application in the development of the DARA-FIRE assessment are summarized in the subsections below.

An internal fire PSA is built from the internal events PSA for the corresponding plant operational state. The scope of the DARA-FIRE model is limited to internal fires initiated with the analysis unit at power with the potential to cause severe core damage. Internal fires considered are those resulting from ignition events within fixed equipment (e.g., electrical panels, pumps, etc.) as well as transient ignition events resulting from human activities in the plant (e.g., combustible material storage, hot work, etc.).

The DARA-FIRE model considers sequences that result in severe core damage. Severe core damage is defined as the sum of the FDC1 and FDC2 frequencies. Severe core damage at Darlington is dominated by the FDC2 frequency. In the fire PSA, FDC1 sequences (failure to shutdown the reactor) represents a very small portion of the sequences leading to SCD due to the low frequency in the internal events model. The fail-safe design of the two shutdown systems (SDS1 and SDS2) and the physical separation of SDS1 and SDS2 make it unlikely that a fire could impact both systems. This limited the number of fire scenarios with the potential to impact more than one channel of one SDS and reduced the probability of a fire-induced failure to trip.

The DARA-FIRE analysis used the Darlington NGS Fire Safe Shutdown Analysis (FSSA) as a starting point to select the components required for safe shutdown following a fire.

5.3.1 Phased Approach to Fire PSA

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

The fire PSA logic is based on the internal events PSA logic. As the fire PSA is developed based on the internal events PSA, the major tasks in the fire PSA are associated with

identifying possible fire scenarios, the zones the fires can impact, affected equipment and cables, and selection of representative internal events sequences and quantifying the consequences of the fire scenarios.

The Fire PSA methodology is broken down into 18 tasks:

- Task 1 – Plant Boundary Definition and Partitioning
- Task 2 – Fire PSA Component Selection
- Task 3 – Fire PSA Cable Selection
- Task 4 – Qualitative Screening
- Task 5 – Fire-Induced Risk Model
- Task 6 – Fire Ignition Frequencies
- Task 7 – Quantitative Screening
- Task 8 – Scoping Fire Modelling
- Task 9 – Detailed Circuit Failure Analysis
- Task 10 – Circuit Failure Mode Likelihood Analysis
- Task 11 – Detailed Fire Modelling
- Task 12 – Post-Fire Human Reliability Analysis
- Task 13 – Seismic-Fire Interactions Assessment (outside the scope of the Darlington NGS Fire PSA; a seismically-induced internal fire and internal flood risk evaluation is undertaken as part of the Darlington NGS Seismic PSA)
- Task 14 – Fire PSA Level 1 Quantification
- Task 15 – Uncertainty and Sensitivity Analysis
- Task 16 – Fire PSA Documentation
- Task 17 – Fire PSA Level 2 Quantification
- Task 18 – Alternate Unit Assessment

The integration of these tasks is shown in Figure 9. The methods prescribed in the Fire PSA Guide are iterative. Several of the tasks listed above involve calculation of severe core damage frequency due to fires in various plant locations. With each subsequent calculation, the methods used to assess the risk for the various scenarios are refined. This iterative

approach is used to identify high risk areas and to focus the detailed fire analysis on these areas. A brief summary of the methodology used for DARA-FIRE is provided in the following sections.

5.3.2 Plant Partitioning

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

5.3.3 Fire PSA Component and Cable Selection

The development of a fire PSA requires identifying components necessary for safe shutdown and long-term decay heat removal following a fire. A fire can affect the equipment credited for safe shutdown by either being in the same area as the credited equipment or by being in the same area as the cables related to the credited equipment. For example, a fire in the same area as the power cables for a pump could result in failure of the pump, even if the pump itself was remote from the fire.

The purpose of this task is to identify the equipment to be explicitly credited in the fire PSA, and determine where in the plant the equipment and cables necessary for their credited function are located.

The set of components selected for explicit credit in the Fire PSA following a fire includes the systems credited in the Darlington FSSA. In addition, Group 2 functions not credited in the FSSA are selected, such as ESW to the moderator and ESW to the Primary Heat Transport (PHT) system. A subset of Group 1 systems powered by Class III power was also selected, including Auxiliary Feedwater, the IUFT, Shutdown Cooling and the Moderator System. To support these front-line systems, support functions such as power supply from the standby generators, LPSW, instrument air and room HVAC were all considered and included in the selected component set as required. EME make-up to the steam generators, HTS and calandria are also credited, including all electrically or pneumatically controlled valves in the injection pathways that can be misaligned by fire, as well as the availability of an interim heat sink from SGECS or gravity fed flow from the Deaerator Storage Tank to provide time for the deployment of the EME.

In addition to the explicit selection of mitigating systems, the impact of fire-induced Multiple Spurious Operation (MSO) has been addressed in this 2020 update of DARA-Fire as part of this task. The MSO assessment started with the list of unscreened scenarios identified for Darlington NGS and each scenario was either further dispositioned, or resulted in the selection of components for which control and/or power cables were traced in order to identify fire scenarios which can initiate the MSO scenario.

Once the equipment to be explicitly credited following a fire event had been identified, the locations and routing of all cables that impact this equipment were identified. This was completed through a simplified circuit analysis process that identified wires with the potential to impact the credited component function, and identification of the cables containing these wires. The route through the plant, including the PAUs in which the cables appear, was established for each cable through a review of the cable tray route sequence against cable tray layout drawings. The cable routing information was compiled in a database and used to determine the fire PSA components potentially affected by postulated fires at different plant locations.

5.3.4 Qualitative Screening

The PAUs, described in Section 5.3.2, may be screened to eliminate those PAUs where the contribution of fire risk to severe core damage is expected to be relatively low or nonexistent compared to other PAUs. The screening criteria considered the following:

- The type of equipment in the PAU;
- The types of ignition sources in the PAU, and the ability to introduce transient ignition sources into the area;
- Impact of the ignition sources on mitigating systems.

Due to specifics of the Darlington NGS design and the analysis approach, this specific task has been excluded from the 2020 DARA-Fire update and instead, quantification of SCDF for all scenarios were performed as described in Section 5.3.12.

5.3.5 Fire-Induced Risk Model

This task involves the development of a logic model that reflects plant response following a fire. This includes modelling the plant response to fire-induced events and modifying the internal events PSA to reflect postulated equipment failures.

The DARA-L2P model was modified and manipulated to produce a fire-induced risk model capable of quantifying both the SCDF and LRF based on the gate selected for quantification. This included incorporating the modified human error event probabilities as described in Section 5.3.11, and incorporating model logic changes specific to the fire analysis such as the addition of fire specific failure modes (e.g., hot shorts). It also included the identification of events in the fire model to be set to “failed” to represent the unavailability of the equipment should they be failed in the fire scenarios.

5.3.6 Fire Ignition Frequencies

To calculate the risk due to an internal fire, the Fire Ignition Frequencies (FIFs) for each PAU must be assessed. The frequencies were calculated based on generic data in NUREG-2169 [R-13] and the plant populations of fixed ignition sources (e.g., pumps, electrical equipment), as well as information regarding plant operations affecting transient ignition sources (e.g., transient material storage, staff occupancy) identified by plant walkdowns and other appropriate means.

The Darlington NGS fire PSA project is limited to Unit 0 and Unit 2. The calculation of FIFs for Unit 0 and Unit 2, however, required calculation of FIFs for all of the PAUs that are within analysis boundary. This was accomplished by:

(1) Conducting Fixed Ignition Sources (FISs) counting walkdowns of Unit 0 and Unit 2 PAUs;

(2) Conducting a 2019 walkdown to confirm additions or removal of Unit 0 and Unit 2 FISs identified through the review of Darlington NGS engineering design changes for the five year period 2014 – 2018;

(3) Conducting a 2019 walkdown of risk-significant Unit 0 and Unit 2 PAUs (as identified in the 2015 DARA-Fire) to identify any other changes to the FIS inventory occurring since the initial 2011 walkdowns; and

(4) Assuming that Unit 2 is spatially representative of the other three operating units, replicating the Unit 2 FISs walkdown data for PAUs in Units 1, 3 and 4.

The Darlington NGS fire experience data was reviewed to determine the applicability of using the NUREG-2169 generic data [R-13]. The qualitative review of CANDU operating experience with fire events found the use of US experience documented in NUREG-2169 [R-13] would result in under-estimating the fire frequency for some types of ignition sources. Therefore, the US generic ignition frequencies were updated using a Bayesian approach to incorporate the Darlington NGS fire experience.

The 2020 DARA-Fire also incorporated consideration of the impact of refurbishment activities on the potential for transient fires in the operating units and adjusted the transient ignition frequencies accordingly in affected areas.

The FISs fire frequency, the transient ignition sources fire frequency and the total FIF were calculated for each PAU.

5.3.7 Quantitative Screening

The development of a fire PSA allows for a quantitative screening of PAUs based on contribution to SCD for a given PAU. This task estimates SCD frequency for each compartment as well as the cumulative risk associated with the screened compartments (i.e., those not retained for detailed analysis). With the information from the fire model and FIFs (described in Sections 5.3.5 and 5.3.6), the contribution to severe core damage by PAU can be calculated. Based on the severe core damage contribution of each PAU, the areas of the plant are further screened, using industry standard screening criteria from Reference [R-12].

Due to specifics of the Darlington NGS design and the analysis approach, this specific task has been excluded from the 2020 DARA-Fire update and instead, quantification of SCDF for all scenarios were performed as described in Section 5.3.12.

5.3.8 Scoping Fire Modelling

The scoping fire modelling refines the initial frequency results obtained in the quantitative screening process. The scoping fire modelling is used to develop explicit fire scenarios within the PAUs. This task involves the use of generic fire models for various fire ignition sources so that simple rules can be used to define and screen fire ignition sources (and therefore fire scenarios) in an unscreened fire compartment. Fire scoping models are developed for all fire areas.

This task has two main objectives:

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

To accomplish these goals, the scoping fire modelling refines the calculation of SCD frequency for each PAU.

5.3.9 Detailed Circuit Failure and Failure Mode Likelihood Analysis

The development of a fire PSA requires detailed circuit failure analysis and circuit failure mode and likelihood analysis. Detailed circuit failure analysis involves identifying how the failure of specific cables impacts the components credited in the Fire PSA. For example, not only can a fire result in failure of equipment, the fire may also result in spurious actuation of equipment, due to possible failure mode of the cables and control logic associated with the equipment.

Circuit failure mode and likelihood analysis task involves the evaluation of the relative likelihood of various circuit failure modes (e.g. failure to operate when required, spurious operation). This added level of resolution applies to those fire scenarios that are significant contributors to the risk.

Circuit analysis was not performed for cables required in the FSSA, because the cable information is already assessed as part of the FSSA. The scope of DARA-FIRE circuit analysis included cable failure mode and failure mode likelihood analysis of components added to the scope of credited safe shutdown equipment credited in the fire PSA, see Section 5.3.3. This task includes, for risk significant components, analysis of circuit operation and functionality to determine whether the cable's fire induced failure could result in undesirable equipment operation. In such cases, a probabilistic assessment of the likelihood that a fire induced failure causes a spurious operation is performed. Given that fire induced cable damage occurs, an appropriate conditional probability is assigned.

5.3.10 Detailed Fire Modelling

Detailed fire modelling was used to perform fire ignition source (scenario) specific fire modelling to address risk significant scenarios in cases where (1) the scoping fire modelling described in Section 5.3.8 produced overly conservative results, or (2) to address the potential fire scenarios not readily addressed by scoping fire modelling. The detailed fire modelling included:

- Explicit treatment of fires in the MCR complex to address fire induced MCR abandonment;
- Explicit analysis of the potential for the formation of damaging hot gas layer from ignition sources in risk-significant PAUs;
- Explicit analysis of multi-compartment scenarios.

The abandonment times for operators in the Darlington NGS Main Control Room (MCR) envelope were assessed for electronic equipment fires and for transient combustible fires within the MCR envelope.

The purpose of the hot gas layer analysis is to determine the probability that a fire originating from a given ignition source in a PAU can generate a layer of hot gases in the PAU that is sufficient to damage all equipment in the PAU, rather than only the equipment within its original zone of influence.

The purpose of multi-compartment analysis is to calculate the probability of compartment interaction caused by a hot gas layer propagation between compartments. The calculation is the product of multiplying the probability of a hot gas layer in the PAU (i.e., the probability that the fire creates a hot gas layer) by the PAU barrier failure probability (i.e., failure of fire doors, dampers and penetrations). The multi-compartment analysis used the hot gas layer development timing defined in Reference [R-14].

5.3.11 Post-Fire Human Reliability Analysis

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

For each fire-related basic event that represents a post-initiator operator action modelled as human failure, HEP multipliers were developed for fire PSA adjustments. The method to apply the HEP adjustment considered the following factors:

- Location (either inside the MCR actions or outside the MCR actions);
- Time available (based on DARA-L1P HRA documentation);
- Availability of indications and controls necessary to diagnose and execute the action;
- Availability of path to the equipment for field actions

Based on the factors above, the baseline HRA value from the PSA may be retained, the HRA value may be multiplied by a factor in the range of 2 to 30, or no credit for the operator action may be taken (failure of operator action assigned a probability of 1).

The 2020 DARA-Fire has also introduced new fire-specific human actions that were not contained in the DARA-L1P model, and has included a fire-specific analysis of the HEP for EME deployment following a fire event.

5.3.12 Fire Level 1 PSA Quantification

The development of a fire PSA requires the integration of the fire ignition frequencies with the damage consequences assessed for each scenario. The damage consequences are imposed on the risk model to quantify a conditional core damage probability (CCDP) given the occurrence of the initiating fire. The combination of the scenario ignition frequency and CCDP defines the SCDF for the scenario. The DARA-Fire SCDF quantification has been performed using the EPRI code FRANX 4.4 [R-15] which incorporates the output of all the previous fire tasks. The integrated SCDF for all scenarios is used to determine the total fire risk.

The development of the fire risk quantification is typically an iterative process. As various analysis refinement strategies are developed, they are incorporated into the fire risk model.

The scoping fire modelling (Section 5.3.8) provided a conservative and simplified means to develop an initial risk estimate. Additional model quantifications to calculate the severe core damage frequency are performed iteratively as additional analysis refinements are incorporated. This includes information gathered during walkdowns conducted for scoping modelling (Section 5.3.8) and additional analysis of other Darlington NGS design inputs (e.g., equipment and cable tray layout drawings) to refine treatment of PAUs that had high estimated SCDFs. This refinement typically divided risk significant PAUs into multiple fire initiating events (scenarios) to represent the individual fire ignition sources. In some cases, multiple fire ignition sources in a PAU were grouped and treated as a single fire initiating event so long as such grouping did not result in overly conservative risk estimates.

5.3.13 Assessment of Unit-to-Unit Differences

The scope of work resulted in specific numerical results for the Unit 2 PAUs and other site PAUs that are common to all four units. Quantification of separate SCDFs and release frequencies for Units 1, 3, and 4 are not specifically included. Because fire risk characterization is needed for the entire plant site, the anticipated symmetry / consistency in the design and construction of the entire four unit site is relied upon to support the applicability of the risk results for the analyzed unit to the other units.

A side-by-side comparison of the Unit 1, 3 and 4 PAUs to the analyzed Unit 2 PAUs was created using fire zone information from the FSSA and the FHA. Equipment layout drawings and general arrangement drawings were also consulted. A walkdown was performed to assess the differences between the units. The walkdown confirmed the physical differences between the units are relatively minor. Although ongoing refurbishment introduces more significant differences in design, it is recognized that it is a temporary condition. All units will be similar in design post-refurbishment. The top contributing scenarios are not impacted by any of the identified differences and no new scenarios were identified that would be expected to contribute significantly to fire-induced risk.

5.3.14 DARA-FIRE 2020

The 2020 DARA-FIRE assessment was prepared according to the OPG Fire PSA Guide. The overall objective of the 2020 DARA-FIRE report was to provide the risk of SCDF due to internal fire events. This has been accomplished as follows:

(1) Update of PAUs to reflect the fire zone definitions in the updated FHA and FSSA.

(2) Update of the FIFs to include use of the latest U.S. industry guidance [R-16] and generic ignition frequency data [R-13], including a conservative treatment of the impact of Darlington NGS-specific fire experience.

(3) The CCDPs for the fire scenarios are quantified using a fire-induced risk model derived from the latest 2020 DARA-L1P model, described in Section 5.1. The updated 2020 DARA-L1P model includes all relevant engineering and operational changes up to the study freeze date of December 31, 2018, including credit of Phase 1 EME.

(4) The explicit selection of equipment and cable tracing to perform the consequence assessment.

(5) The integration of the ignition frequency, and damage consequence for each fire scenario to generate scenario SCDFs.

5.4 At-Power Internal Flood

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

The 2020 DARA-FLOOD assessment was developed following the methodology for preparation of an Internal Flood PSA as described in the OPG Flood PSA Guide. The 2015 model and analysis were used as the basis for developing the 2020 assessment described in Section 5.4.7.

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

The flooding analysis is focused on two primary objectives: areas of the plant that contain equipment from both Group 1 and Group 2 systems (referred to as “pinch points”), or areas which might completely disable all of Group 1 or Group 2, as these areas represent the highest potential for degradation of the plant mitigation capability; and conservative estimation of risks associated with the other areas of the plant. A major input into the Internal Flooding PSA is the At-Power Internal Events PSA (DARA-L1P). The At-Power Internal Events PSA is used to determine which components need to be evaluated for flooding impacts, and is also used as the basis for the quantification of the internal flooding severe core damage frequency.

The construction of the Internal Flood PSA requires the following steps:

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

(2) Identification of Flood Sources.

(3) Internal Flood Qualitative Screening.

(4) Potential Flood Scenario Characterization.

(5) Internal Flooding Initiating Event Frequency Estimation.

(6) Flood Consequence Analysis.

(7) Evaluate Flood Mitigation Strategies.

(8) PSA Modelling of Flood Scenarios

(9) Internal Flooding Level 1 PSA Quantification.

(10) Sensitivity and Uncertainty Analysis.

(11) Support Task – Plant Walkdowns.

Figure 10 shows the tasks for the flooding PSA.

The flooding PSA focuses on sequences that lead to severe core damage (FDC1 and FDC2) caused by an internal flood. Failure to shutdown sequences (FDC1) are not quantified as the frequency of FDC1 is several orders of magnitude lower than FDC2 in the DARA-L1P model (see Table 19) and the potential for flooding events to adversely affect the shutdown systems, which fail safe on loss of power or loss of actuation inputs, is minimal.

5.4.1 Identification of Flood Areas, SSC and Flood Sources

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

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

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

5.4.2 Internal Flood Qualitative Screening

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

The following rules were used when screening

- The area is outside of Unit 2 (the reference unit) or Unit 0 (common unit);
- The area is evaluated as a Screen 1 in the FSSA (see Section 5.3.4);
- The area contains no Group 1 equipment affecting FDC2;
- The area contains no Group 2 equipment affecting FDC2;
- The area contains no credible flood source, or credible flooding propagation paths into the area of the collocation.

The unscreened areas are the pinch-point areas for the flooding assessment

5.4.3 Potential Flood Scenario Characterization and Consequence

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

- Type of flood source, including the type of pressure boundary failures (e.g., spray, large leak, major structural failure), capacity of the flood source (e.g., unbounded lake source, closed tank);
- Through-wall flow rate or spill rate;
- Flood location;
- Time to reach the critical flood volume (e.g., to submerge equipment, or lead to propagation into another area);
- The impact on the SSCs modelled in the PSA.

5.4.4 Internal Flooding Initiating Event Frequency Estimation

This step identifies flooding induced initiating events and estimates their frequency of occurrence. The flooding failure rates are based on generic EPRI data from Reference [R-17].

5.4.5 Flood Mitigation Strategies

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

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

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

5.4.6 Internal Flooding Accident Sequence and Level 1 PSA Quantification

This step includes the finalization of flood scenario development and completing internal flood accident sequence models based on modifying the internal events PSA model. The 2020 DARA-FLOOD followed a quantification methodology more in line with the Internal Events PSA methodology where all scenarios are captured in a single-top model and the quantification did not use CCDP as in the previous assessment. A generalized simplified flood scenario event tree was used as the basis for developing flood scenarios. For the scenarios developed as part of the previous assessment, the event trees and associated CCDP have been used as the basis for developing the flood specific fault tree logic. The failure or successful isolation of breaks, as modelled in the simplified flood scenario event trees, along with the respective CCDP cases define the combinations of events that need to be added to mitigating system fault trees for the single-top model used in the 2020 DARA FLOOD.

Qualitative sensitivity and uncertainly analyses were included as part of the quantification of the 2020 DARA-FLOOD model.

5.4.7 DARA-FLOOD 2020

The 2020 DARA-FLOOD assessment was prepared according to the OPG Flood PSA Guide. The overall objective of the 2020 DARA-FLOOD report was to provide the 2020 DARA-FLOOD results. This has been accomplished as follows:

(1) Update of the piping rupture frequencies with the latest EPRI data [R-18].

(2) Assessment of postulated flooding scenarios impact on deployment of the EME, including accessibility of the deployment locations and the associated HEPs. Generally, the flooding scenarios credit EME for preventing severe core damage using the same logic modelled in DARA-L1P.

(3) Re-quantification of Severe Core Damage Frequency (SCDF) for all postulated flood scenarios.

(4) The qualitative screening, flood area identification, and flood source identification are based on the FSSA.

5.5 At-Power Seismic

The DARA-SEISMIC assessment has been developed following the methodology for preparation of a seismic PSA as described in the OPG Seismic PSA Guide. The major activities of the Seismic PSA methodology and its application in the development of the DARA-SEISMIC assessment are summarized in the subsections below.

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

The DARA-SEISMIC model considers sequences that result in severe core damage (i.e., end states FDC1 and FDC2). Accident sequences that postulate a failure to shutdown the reactor (i.e., end state FDC1) are not explicitly assessed following a seismic event. Failure to shutdown following a seismic event is highly unlikely as SDS1 and SDS2 are diverse, highly reliable, have a fail-safe design, and are seismically robust.

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

- **Phase 1 - PSA-Based Seismic Margin Assessment (SMA)** - In Phase 1, a PSA-based SMA is performed based on the methodology described in Reference [R-19]. This focused approach uses a plant model based on DARA-L1P with the addition of new seismic failure modes; the new seismic failure events are developed from a seismic margin approach with generic variabilities and the seismic risk is calculated based on a point estimate format that does not include a full uncertainty analysis.

- **Phase 2 - Seismic PSA (SPSA)** – In Phase 2, the Phase 1 results are used to identify the most effective approach to convert the Phase 1 risk-based seismic margin study into a SPSA. Uncertainty in the seismic hazard and seismic fragilities are included, propagated, and displayed in the final quantification of risk estimates of the plant for significant risk contributors.

For Darlington, a Phase 2 Seismic PSA study was performed.

Major elements of the Darlington NGS SPSA consist of the following tasks:

- Task 1 - Seismic Hazard Characterization
- Task 2 - Plant Logic Model and Seismic Equipment List Development
- Task 3 - Seismic Response Characterization
- Task 4 - Plant Walkdown and Screening Reviews
- Task 5 - Seismic Fragility Development

- Task 6 - Seismic Level 1 PSA Quantification
- Task 7 - Level 2 Evaluation (see Section 6.11)
- Task 8 - Alternate Unit Analysis
- Task 9 - Seismic PSA Documentation

The integration of these tasks is shown in Figure 11.

5.5.1 Seismic Hazard Characterization

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

The site-specific seismic hazard curve is used to define the earthquake characteristics used in the PSA analysis

5.5.2 Plant Logic Model Development

This task involves two related but separate sub-tasks: development of the accident progression logic for the risk quantification model, and development of the Seismic Equipment List (SEL), which lists the structures, systems and components credited in the seismic PSA. This task relies upon the internal events PSA and other safe shutdown analyses to define the functions, systems, and components required to mitigate seismic initiating events. The seismic model was updated to credit systems and equipment modified or replaced since the last PSA update (e.g., new EME design, CFVS).

A starting point for the SEL is the fire safe shutdown equipment list. The SEL credits systems that are seismically qualified (e.g., SDS2, ESW, ECI, EPS, EPGs, and required support systems), or seismically assessed (e.g., EME) with preventing SCD over the entire seismic hazard range. Since at lower magnitude earthquakes it is likely that the majority of DNGS systems are still fully operational and capable of performing their SCD mitigating function, selected non-seismically qualified DNGS systems are credited in the lower portion of the seismic hazard.

The SSCs in the reference unit (i.e., Unit 2) and the common systems (i.e., Unit 0) are modelled in the SPSA model, are assessed in later SPSA tasks (e.g., fragility development).

5.5.3 Seismic Response Characterization

The next step in the seismic PSA is to characterize how the station buildings respond to a seismic event. The response of the building will not be the same on each elevation. For example, the small earthquakes occasionally experienced in southern Ontario are typically

undetectable to people in the basement or lower floors of buildings, but can be easily detected by people in the higher floors of tall buildings.

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

In Phase 1, a generalized scaling approach is used to calculate the structural response of the site buildings. This method is based on the existing DBE seismic response analyses for the site buildings, prepared as part of the design for the Darlington NGS, with updates to reflect the shapes of the new seismic hazard curves. In addition to characterizing the overall building response, this task defines the local accelerations for the credited equipment. In Phase 2, seismic responses analyses were performed for selected site structures, considering soil-structure interaction (SSI) and ground motion incoherence (GMI) analysis. Insights from the SSI/GMI analyses of these structures were used to refine the response of these structures to the seismic event. The potential for seismically induced soil liquefaction was considered.

5.5.4 Plant Walkdown and Screening Reviews

Plant walkdowns were required to assess the relative vulnerability of equipment to seismic challenges. The walkdowns were performed by fragility experts in order to document the basis for screening equipment in, based on susceptibility, or out, based on ruggedness, of the SPSA. The plant walkdowns included reviews of the SEL items in one unit and the items in the systems common to all four units. The 2020 DARA-SEISMIC update included a walk down to assess SEL SSCs including:

- Group 2 SSCs;
- Spatial interactions;
- Seismically Induced Internal Fires and Floods (SIIFF) sources that were not screened in the previous SIIFF studies;
- EME storage building and deployment paths;
- MCR access paths;
- Group 1 SSCs that can be credited with a magnitude up to 0.1g; and
- SSCs installed due to refurbishment activities (e.g., temporary containment boundary components).

5.5.5 Seismic Fragility Development

The likelihood that a given piece of equipment will fail for a given seismic hazard is based on the fragility of the equipment. The fragility of the equipment is a conditional failure probability

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

Preliminary fragilities were determined through a combination of walkdown review of the as-installed configurations, experience-based estimates, and equipment-specific fragility calculations using the Conservative Deterministic Failure Margin (CDFM) methodology with generic representations for the variability [R-20]. In some cases, more refined fragilities were derived using the Hybrid Method, or the Separation-of-Variable method [R-20], for risk-significant SSCs.

For the 2020 DARA-SEISMIC, the fragility analysis includes consideration of the findings from the seismic walkdown, potential spatial interactions, the impact of SIIFFs, and the impact of seismically induced soil failures.

5.5.6 Seismic Level 1 PSA Quantification

Quantification of the seismically induced SCDF requires the integration of the seismic hazard developed in Section 5.5.1, the plant logic model developed in Section 5.5.2, selected portions of the DARA-L1P integrated model for systems credited in the SPSA, and seismically induced failures of credited SSCs.

In the development of DARA-SEISMIC model, the seismic hazard curve was divided into discrete ground motion intervals for the purposes of quantification. Eight intervals were used to represent the seismic hazard; Table 15 shows the intervals used for DARA-SEISMIC. These intervals are treated as the initiating events in the DARA-SEISMIC study. Their frequencies are calculated as the annual exceedance frequency at the beginning of interval minus the annual exceedance frequency at the end of the interval. The information on the seismic response of the buildings and the seismic fragility of the credited SSCs, developed in Sections 5.5.3 to 5.5.5, was used to calculate the probability of seismically induced failures in each interval. The EPRI code FRANX 4.4 [R-15] was used to model the seismically induced initiating events and SSC failures in the DARA-SEISMIC model. Quantification of the DARA-Seismic model is performed for each seismic hazard interval to represent the risk over the entire seismic hazard range. The seismic PSA presents the risk of severe core damage for earthquakes with a frequency up to 1E-04 occurrences per year (recurrence interval of 10,000 years or less). Consistent with DARA-L1P, importance, uncertainty and sensitivity analyses were performed to identify key insights.

5.5.7 Alternate Unit Analysis

The Unit 2 is the analysis unit, with one unit undergoing a refurbishment outage (i.e., Unit 3), and the remaining units operating at full power. There is the possibility that unique physical differences between units could contribute to somewhat different seismic response from unit to unit. However, the results of the Level 1 analysis for Unit 2 show that the seismic risk is driven by correlated failures of shared systems (i.e., common portions of EPS and ESW). Thus, unit-to-unit differences are not expected to have significant impact on the DARA-Seismic results and no further assessment is deemed necessary at this time.

5.6 At-Power High Wind

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

The primary steps in developing the high wind PSA are identifying the high wind hazard, identifying the high wind targets, developing wind-borne missile fragilities for the high wind targets, evaluating the fragility of the high wind targets, developing the high-level plant logic, and quantifying the high wind scenarios. The high wind PSA was created based on the internal events At-Power PSA, DARA-L1P.

Figure 14 shows how each step feeds into the overall DARA-Wind study. The methodology applied in the high wind hazard assessment uses a high level approach in determining fragilities based on wind capacity. The approach is realistic with conservative assumptions to simplify the analysis where needed.

5.6.1 High Wind Hazard Analysis

The first step in the high wind PSA is to identify the potential contributing wind hazards at the site. The primary hazard includes straight winds (thunderstorms and extratropical cyclones), hurricanes and tornadoes. The wind hazard curve is developed for peak gusts in open terrain at 10 m height. Terrain, height, and averaging time adjustments were performed to adjust gust wind data to 3 second gust speed at a height of 10 m in flat open terrain. Figure 15 shows an example of high wind hazard curves.

Similar to the seismic results, the high wind results are reported for high winds with a frequency up to 1E-04 occ/yr.

5.6.2 Plant Logic Model Development

This task involves two related but separate sub-tasks: development of the event tree logic for the risk quantification model, and identification of target systems, structures, and components (SSCs) that are included in the high wind PSA model. The high wind plant logic model examines the response of plant SSCs to the defined high wind hazard. It then combines this response with the response of the plant to the initiating event, given the degraded condition of the plant's SSCs and the challenges faced by the operator due to the wind hazard. The focus of the high wind analysis is estimation of severe core damage frequency for a single reference unit, with consideration of the common unit and adjacent unit impacts on the reference unit.

5.6.3 Analysis of Windborne Missile Risk

Windborne missile fragility is defined as the probability of target damage (failure) from windborne missiles for a given value of peak gust wind speed. Wind-borne missile risk includes:

(1) Missiles that hit/damage an exterior target.

(2) Missiles that enter a building and hit an interior target.

(3) Missiles that originate within a building and hit an interior target.

The windborne missile risk analysis considered the risk from all potential missiles at and near the site. Missile data were collected from the site walkdown, plant layout and SSC drawings.

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

The windborne missile risk considered failure of building components in the determination of flying missile risk and missile fragilities for targets. The failed building components (such as cladding, roof top equipment, roof elements, and loose contents) were assumed to be available missiles at appropriate wind speeds associated with the failure of the building envelope components for that building type.

The windborne missile fragilities were represented by missile hit, missile penetration, perforation, spall, or other damage relationship appropriate for the target.

5.6.4 High Wind Fragility Development

Wind fragility is defined as the conditional probability of failure for a given value of peak gust wind speed. The general objective of the wind fragility study is to assess the aerodynamic wind forces which may result in damage to buildings housing safety-related equipment and their contents and to determine associated uncertainty.

High wind capacities and corresponding fragilities were developed for the identified targets. For each wind hazard type, the fragility of screened-in targets was assessed using an advanced code-based methodology. This method applies a code-based approach with code and load-effect calculations and considers wind direction, terrain roughness, blockage, and structure enclosure state. The mean fragilities were used in the risk quantification to represent the nominal point estimate fragility of a given component.

5.6.5 High Wind Hazard Site Walkdown

The high wind hazard walkdown includes a walkdown of credited SSCs and a missile survey. The walkdowns of SSCs were performed in order to confirm all the structures and their condition, vulnerability of the equipment, etc. The walkdowns of the windborne missile survey were conducted and covered each missile source zone at the entire site. The survey collected data on the types, numbers, and locations of potential missiles (e.g., construction materials, equipment, automobiles, signs, trees, and vulnerable structures that are likely to fail in windstorms).

5.6.6 Plant Response Model Quantification

Quantification of the high wind PSA models requires the integration of the wind hazard curves from Section 5.6.1 and the fragility curves from Section 5.6.4 along with the non-high wind or random failure modes according to a Boolean representation of ways the plant response is assumed to lead to core damage.

This task involves the integration of the high wind hazard and fragility information with the overall plant logic model, by adding the fragility information to appropriate sequences and basic events in the plant logic model.

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

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

6.0 LEVEL 2 PSA METHODS

Section 5.0 described the methods used for the Level 1 PSA assessments of Darlington NGS. In the Level 1 PSA, the goal was to quantify the frequency of fuel damage. Once the fuel has been damaged, there is the potential for radioactive material to be released from the fuel into containment. The Darlington NGS design includes a containment system (described in Section 2.3.14) to prevent the release of any radioactive material in the station from being discharged into the environment.

The Level 2 PSA studies the system failures and accident phenomena that might result in a release to the environment, and the timing and magnitude of the release. This information is combined with the Level 1 DARA-L1P model to quantify the frequency of possible releases.

The DARA-L2P model has been developed following the methodology for preparation of a Level-2 PSA as described in the Level 2 PSA Guide. The major activities of the Level-2 PSA methodology and its application in the development of DARA-L2P are summarized in the subsections below.

6.1 Interface with Level 1 PSA

The Darlington Level 1 At-Power Internal Events PSA (DARA-L1P) generates results in the form of frequencies of nine Fuel Damage Categories, described in Section 5.1.2, representing a wide range of possible outcomes. The possible outcomes include the most severe involving failure to shutdown (FDC1) to relatively benign where there are no fuel failures and release is limited to the equilibrium fission product inventory of the Heat Transport System (HTS) (FDC9). A subset of the FDCs (1-7), those that involve release of significant quantities of

fission products from the core, is used to develop the interface between Level 1 and Level 2. Subsets are grouped into Plant Damage States (PDSs). The PDSs serve to reduce number of the sequences assessed in the Level 2 analysis to a manageable number while still reflecting the full range of possible accident sequences and their impacts on the plant.

Only two FDCs are used to represent the range of sequences that result in severe core damage, FDC1 for rapid accident progression resulting from failures to shut down the reactor when required and FDC2 for all other sequences. FDC1 is conservatively assumed to cause early consequential containment failure and is assigned to a unique PDS, PDS1.

FDC2 is not assumed to result in immediate containment failure and was subdivided into three PDSs (2-4) to examine the potential for random and consequential failures of containment systems that could eventually lead to enhanced release to the environment:

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

Random containment system failures are associated only with PDS2 and were identified by means of a Bridging Event Tree (Figure 16) that led to the creation of five subcategories, labelled PDS2A-E.

As described in Section 1.0, Unit 2 is the reference unit for the PSA study. In order to develop the logic for PDS3, additional simplified modelling of the other three units was undertaken to partition the FDC2 logic into sequences that impact a single unit, and sequences that could impact more than one unit.

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

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

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

The Level 1 PSA is used to identify initiating events that are the largest contributors to the frequency of the plant damage state. These sequences are then reviewed to select a representative sequence that bounds the consequence. The approach follows the guidance of the IAEA as this method selects a sequence that “largely bounds” the PDS. The representative sequences chosen for each PDS are summarized in Table 16.

6.2 Containment Event Tree Analysis

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

A CET is a logic model that addresses uncertainties in the ability to predict the potential impacts of accident progression and associated physical phenomena on containment response. Figure 17 shows a simplified CET. CET branch points are not built from system based "success criteria" but from questions that are intended to ascertain the magnitude of phenomenological challenges to the containment boundary and its continued integrity at a given stage of accident progression (e.g., *"Is containment integrity maintained?"* or *"Does core concrete interaction occur?"*). The CET branch points represent major events in accident progression and the potential for fission product release to the environment. The CET also represents the evolution of the progression with time so the same nodal question may appear more than once in the tree as conditions inside containment change. The focus of the CET is to estimate the probabilities of the various ways that containment failure may occur leading to a release to the environment.

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

6.3 Containment Fault Trees

The containment fault trees developed as part of the level 2 PSA are the following:

CEI:	Impairment of Containment Integrity Avoided
ACU:	Reactor Vault Cooling System Condenses Steam
IGN:	Hydrogen Igniters Control Possible Hydrogen Burn
CFVS	Containment Filtered Venting System (not credited in the baseline DARA-L2P assessment)
EFADS:	Emergency Filtered Air Discharge System Filters and Vents (not credited in the baseline DARA-L2P assessment)

The fault tree models documented in the Level 2 PSA are listed in Table 11. Fault tree representations for failure of these containment functions have been developed, reflecting the likelihood that random equipment failure or human error will prevent the operation of the system on demand or during the mission.

Containment system fault trees are required for quantification of the frequencies of the end-states PDS2A – PDS2E in the Level 1/Level 2 PDS2 bridging event tree, which is shown in

Figure 16. Containment failures arising as a consequence of severe accident progression are addressed in the CET.

6.4 Release Categorization

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

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

Seven RCs cover the full range of possible releases and provide enough discrimination to evaluate safety goal frequencies. An eighth category is used to represent basemat melt-through, when the core debris is postulated to penetrate the floor of the fueling machine duct. Table 17 presents the release categories used in the DARA-L2P analysis. Large release frequency (LRF) is defined to be the sum of RC1 through RC3.

6.5 MAAP-CANDU Analysis

MAAP-CANDU (Modular Accident Analysis Program – CANDU) is a severe accident simulation code for CANDU nuclear stations [R-21]. It is used to calculate the consequences of severe accidents and is designated as a CANDU Owners Group (COG) Industry Standard Toolset (IST) code. MAAP-CANDU originated from MAAP developed for Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) systems by Fauske and Associates (FAI) and is part of the EPRI suite of probabilistic safety assessment tools.

MAAP-CANDU can simulate the response of a CANDU power plant during severe accident sequences. The code quantitatively predicts the evolution of a severe accident starting from full power conditions given a set of system faults and initiating events through events such as core melt, primary heat transport system failure, calandria vessel failure, shield tank failure, and containment failure. Severe accident analysis carried out using MAAP-CANDU is the cornerstone of the Level 2 PSA. There are at least five distinct roles for the code, as outlined below:

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

- To generate results to support systematic sensitivity and uncertainty analysis; and
- To provide information related to plant environmental conditions.

6.6 Severe Accident Management Guidelines

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

SAMG documentation is treated as guidance, compared to AIM response, which uses procedures. The type of actions included in SAMG range from recovery of systems typical in the prevention of severe core damage (i.e. ECI, moderator cooling) to crediting systems or injections lineups in non-traditional ways that are not typically included in the AIM response. While Phase 1 EME is used prior to the entry into SAMG as a prevention mechanism, it can also be used within the SAMG framework if not successful in preventing severe core damage.

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

6.7 Integration of the Level 1 and 2 PSA

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

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

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

6.8 Level 2 Outage Assessment

Given the low risk of fuel damage from internal events occurring while the unit is in GSS, a full Level 2 study of the outage risks was not performed. Instead a bounding assessment of the large release was performed while the unit is in outage.

The at-power Level 2 assessment (DARA-L2P) demonstrated that a large release can only occur if severe core damage has occurred, so the large release frequency while the unit is in outage can be bounded by the frequency of severe core damage while the unit is in outage.

Nonetheless, an LRF estimate was performed by identifying the following:

- Outage sequences leading to LRF

- Single-unit vs multi-unit cutsets
- Fraction of single unit sequences leading to LRF
- Fraction of multi unit sequence leading to LRF.

The plant configuration in each POS was reviewed for potential containment failures (random failures, containment bypass, or consequential containment failure). A limited number of outage specific considerations were identified that might impact the severe accident progression.

Additional MAAP-CANDU analysis was performed to assess the consequences of the identified outage sequences.

6.9 Level 2 Fire Assessment

The Level 2 assessment of internal fire risk was evaluated using the fire-induced risk model described in Section 5.3.5, which was developed based on the DARA-L2P model. The DARA-Fire LRF quantification has been performed using the EPRI code FRANX 4.4 [R-15] which incorporates the output of all the previous fire tasks. Since the fire-induced risk model has been prepared to quantify SCDF and LRF, and the fire scenario impact includes consideration of Level 2 equipment credited in the FSSA, the LRF is quantified as described in Section 5.3.12 with selection of a different top event in the fire-induced risk model for quantification of LRF.

6.10 Level 2 Flood Assessment

The LRF is estimated using the 2020 Level 1 Darlington NGS Internal Flood PSA SCD sequences.

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

- Cutsets involving single unit with flooding event inside the containment;
- Cutsets involving single unit with flooding event outside the containment;
- Cutsets involving two units;
- Cutsets involving more than two units, which will be referred to as Multi-Unit.

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

6.11 Level 2 Seismic Evaluation

The Level 2 seismic evaluation included the following tasks:

- Develop the Level 2 SEL;
- Perform walkdown of Level 2 SSCs;
- Estimate of the seismic fragility of Level 2 SSCs;
- Estimate of LRF due to seismic events; and
- Evaluate the robustness of containment response to seismic events.

The development of the Level 2 SEL was performed in the same manner as the Level 1 SEL, as described in Section 5.5.2. Walkdowns of Level 2 SSCs were performed with those of Level 1, described in Section 5.5.4. Level 2 fragilities were calculated using the same techniques as those described in Section 5.5.5.

The estimate of LRF was performed by analyzing the Level 1 SCDF results, quantified in Section 5.5.6. The SCDF cutsets were divided into those that represented containment failure (e.g., containment bypass), and those for which additional failures are required to cause large releases. Containment failure SCD cutsets were treated as contributing directly to LRF. In those SCD scenarios that did not fail containment, their contribution to LRF was calculated considering:

- Insights from DARA-L2P (e.g., accident progression, phenomenological failures of containment);
- Random failure of containment; and
- Seismically induced failure of containment.

The evaluation of the robustness of containment response to seismic events was performed, based on examination of the limiting fragilities (i.e., those SSCs with the least seismic capacity in seismic events) for the containment system components.

6.12 Level 2 High Wind Assessment

The Level 2 high wind assessment was performed using insights from the Level 2 At-Power Internal Events PSA. To estimate the LRF, the Level 1 and Level 2 Models were used with specific hazards added. This approach has the advantage that the fraction of each type of cutset (e.g. single unit, multi-unit) that leads to LRF are quantified directly with the logic developed from the Level 2 CETs and fault trees.

6.13 Non-Reactor Source PSA

While the hazard screening analysis had screened out all hazards associated with the UFDS facility, selected internal and external natural hazards for the fuel in the IFB were screened in. Bounding simplified quantitative assessments were used for the following hazards.

- Loss of heat sink and loss of inventory

- Earthquake
- External flooding
- Extreme temperature
- Snow / snowpack
- Freezing rain
- Ice storms
- Tornado / high winds
- Geomagnetic storm and solar flare
- Internal fires
- Internal flooding

An assessment of interactions between accident progressions in reactor units and IFB was also conducted.

7.0 SUMMARY OF RESULTS

7.1 Frequencies of Severe Core Damage and Large Release

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

- Frequency of severe core damage; and
- Frequency of large release.

Table 18 compares the results of the PSA studies described in Sections 5.0 and 6.0, with the OPG safety goals for individual hazards on a per-unit basis.

OPG has both safety goals and administrative goals. The safety goal represents the limit of tolerability of risk exposure above which action shall be taken to reduce risk. The administrative safety goal represents the desired objective towards which the facility should strive to the extent practicable.

The results in Table 18 show that the severe core damage frequency results for individual hazards is below the OPG Safety Goal of 1E-04 per reactor-year. Moreover, most of the severe core damage frequency results are below the OPG Administrative Safety Goal target of 1E-05 per reactor-year. Similarly, the large release frequency results are below the OPG Safety Goal of 1E-05 per reactor-year, with most of the results being below the OPG Administrative Safety Goal of 1E-06 per reactor-year.

The internal events PSAs assess the full range of fuel damage and release categories defined in Table 10. The frequencies of fuel damage categories for the at-power internal events PSA (DARA-L1P) is presented in Table 19. The results in Table 19 show that failure to shutdown is a negligible contributor to severe core damage frequency. The frequency of fuel damage for outage internal events (DARA-L1O) by POS is presented in Table 20. The outage results in Table 18 show that the risk is below the OPG Administrative Safety Goal.

As described in Section 6.1, the fuel damage categories used as end states in the Level 1 PSA are partitioned into PDSs to use as inputs into the Level 2 PSA. Table 21 presents the frequencies of the PDSs, and Table 22 presents the results of DARA-L2P.

7.2 Conclusions

The PSA for the Darlington NGS (DARA) is performed in accordance with CNSC Regulatory Document REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants. The 2020 DARA update uses methodologies for which upfront CNSC's acceptance had been obtained. It addresses Level 1 and Level 2 PSA aspects for various internal and external events, for both at-power and outage operating conditions, including internal events, internal fire, internal flood, seismic, high winds, non-reactor sources, as well as an external and internal hazard screening assessment.

The 2020 DARA results demonstrate that the Darlington station satisfies OPG's safety goal for all internal and external hazards considered, and hence represents very low public risk. OPG continues to meet industry good practices through periodic PSA updates to account for operating experience, improvements in analysis methods, and changes at the station.

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Figure 1: Site Area

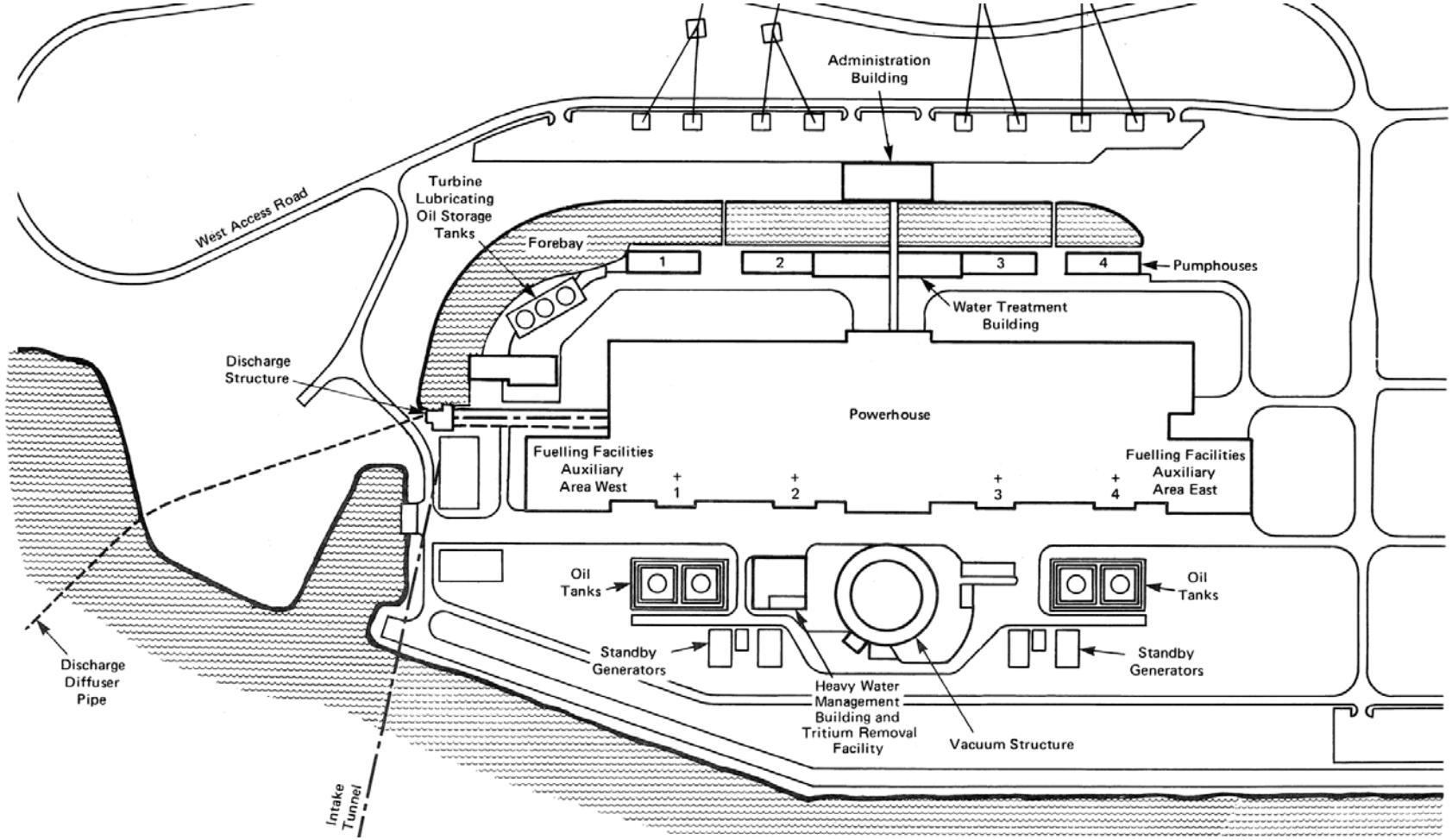
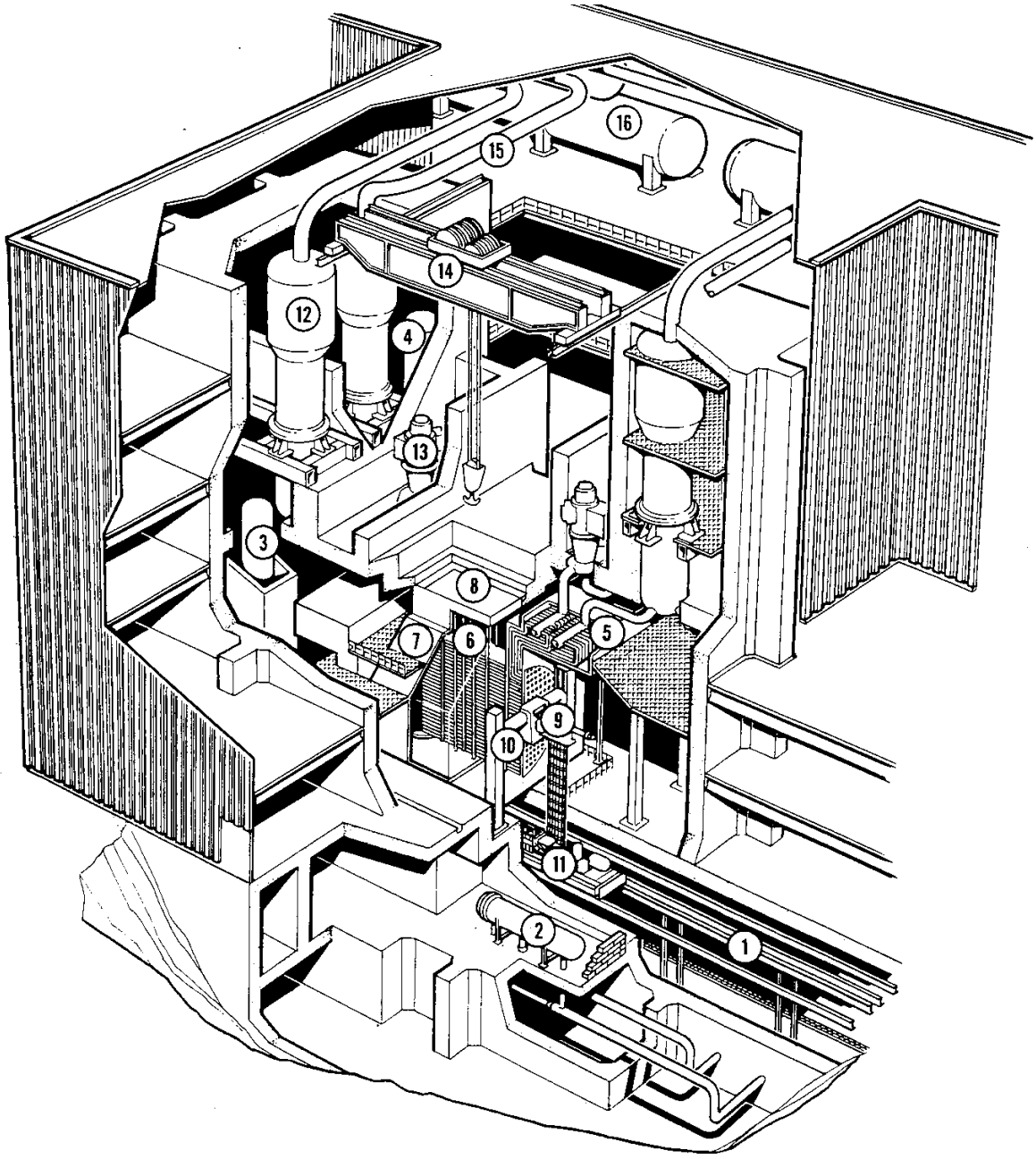


Figure 2: Darlington Station General Arrangement



20000 0005:1

1 Fuelling Duct
2 Shutdown Cooling Heat Exchanger
3 Pressurizer
4 Heavy Water Storage Tank
5 Feeder Cabinet
6 Calandria
7 Shield Tank
8 Reactivity Mechanism Deck
9 Fuelling Machine Head
10 Fuelling Machine Bridge Column
11 Fuelling Machine Transport Trolley
12 Steam Generator
13 Heat Transport Pump
14 Bridge Crane
15 Main Steam Line
16 Deaerator

Figure 3: Darlington NGS Reactor Building

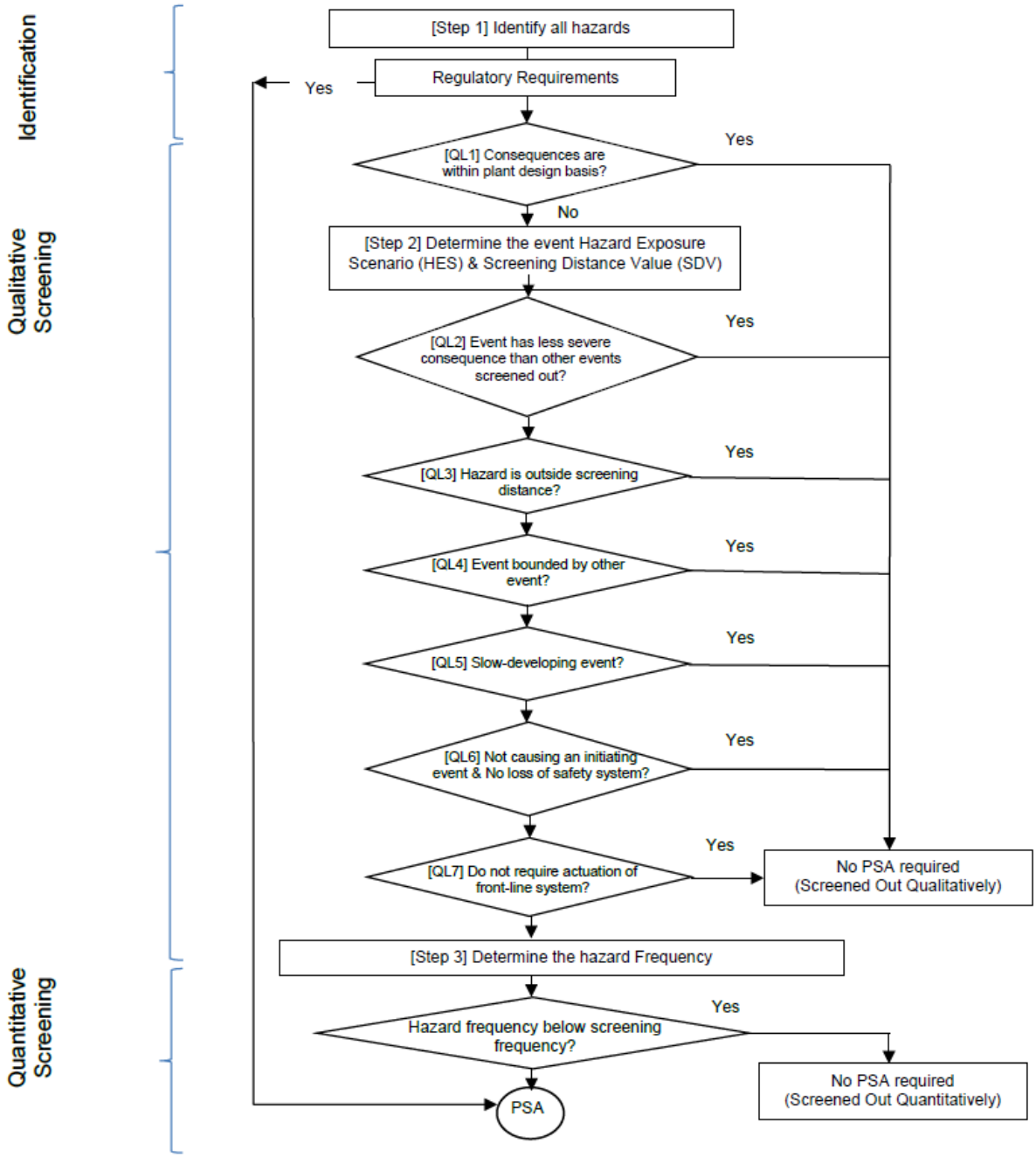


Figure 4: Hazard Screening Steps

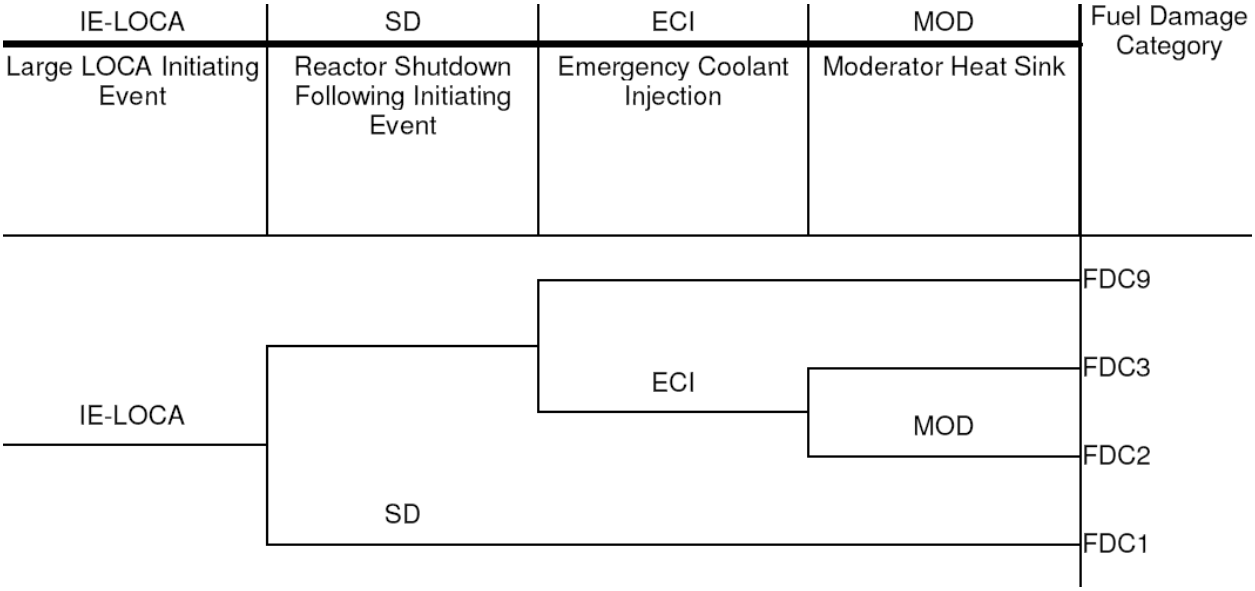


Figure 5: Example LOCA Event Tree

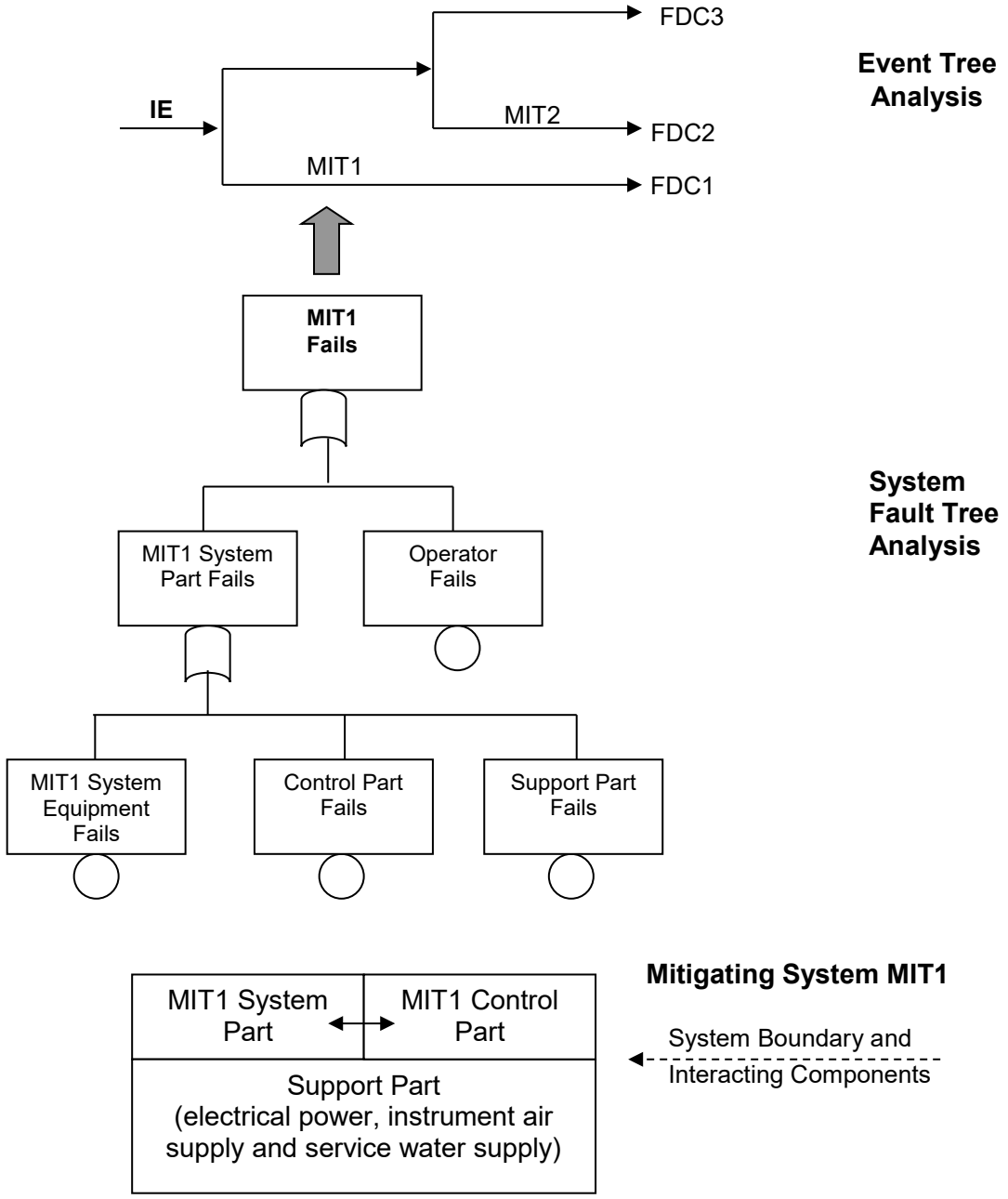


Figure 6: Fault Tree and Event Tree Integration

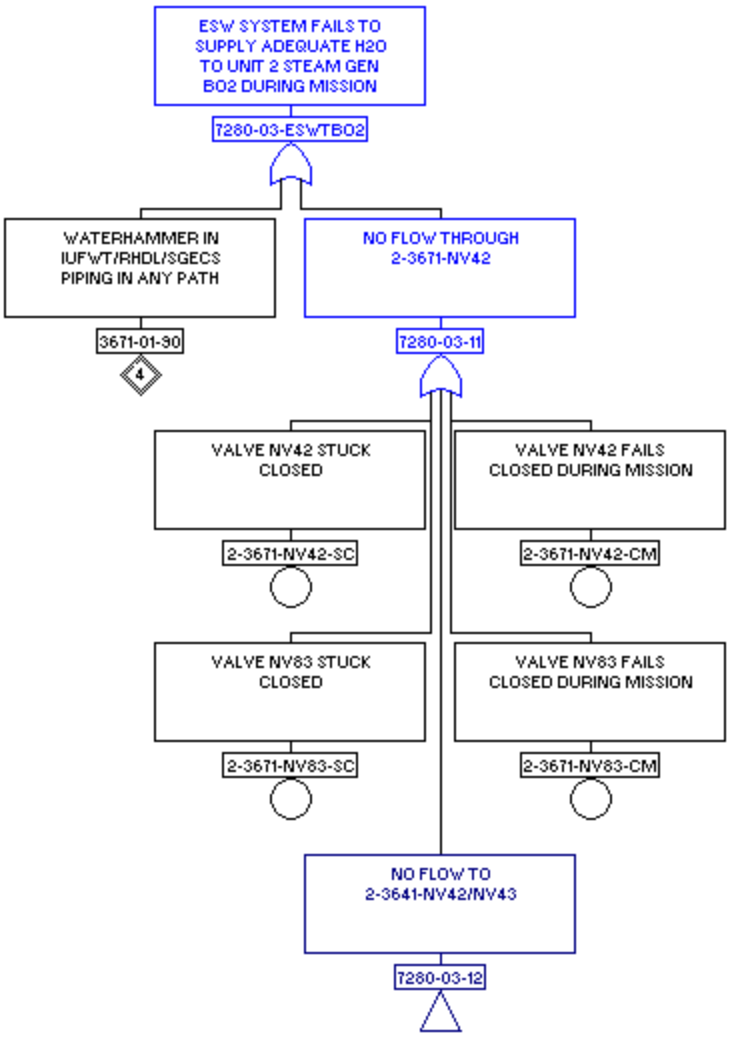


Figure 7: Example Fault Tree

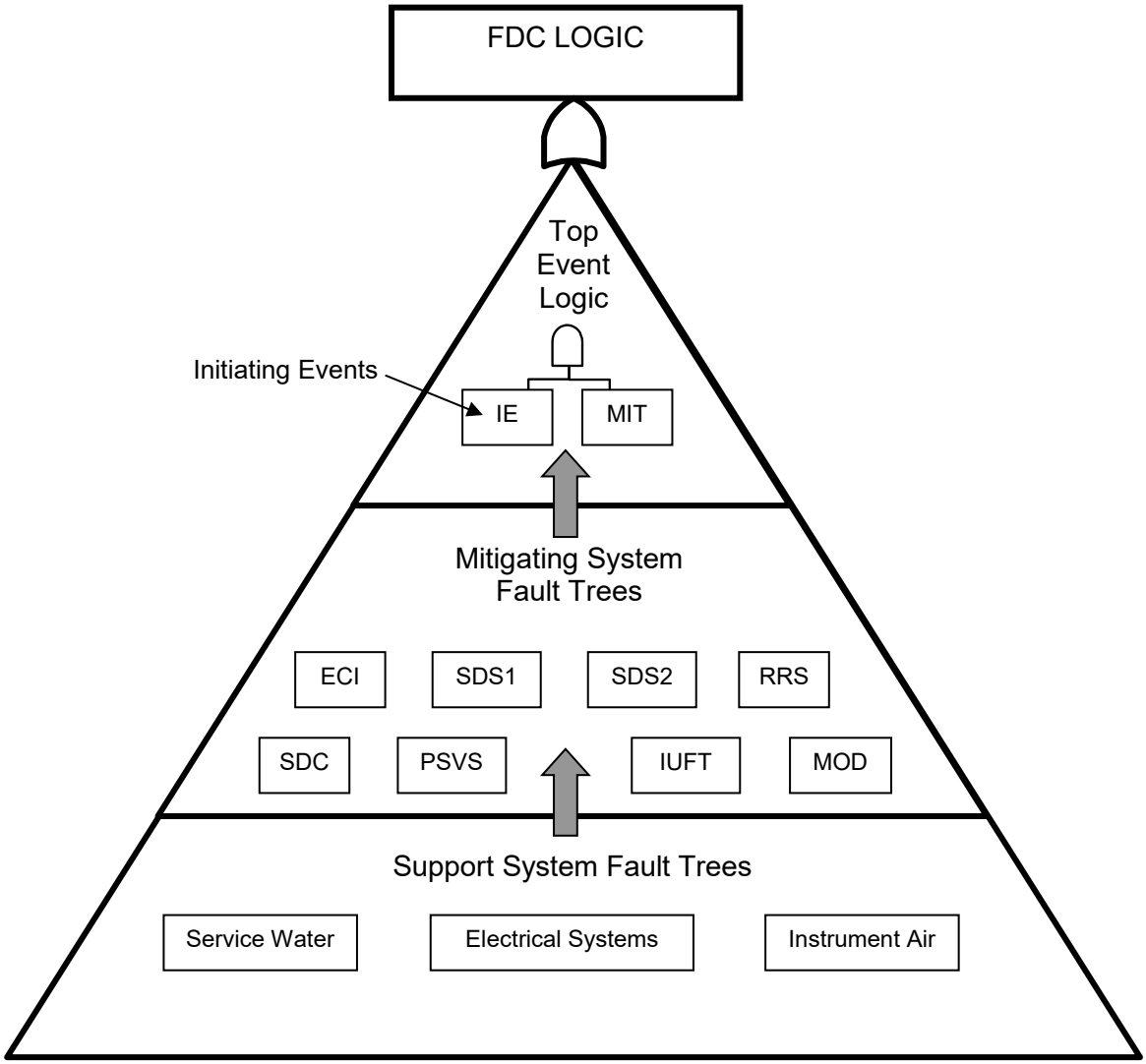


Figure 8: Fault Tree Integration

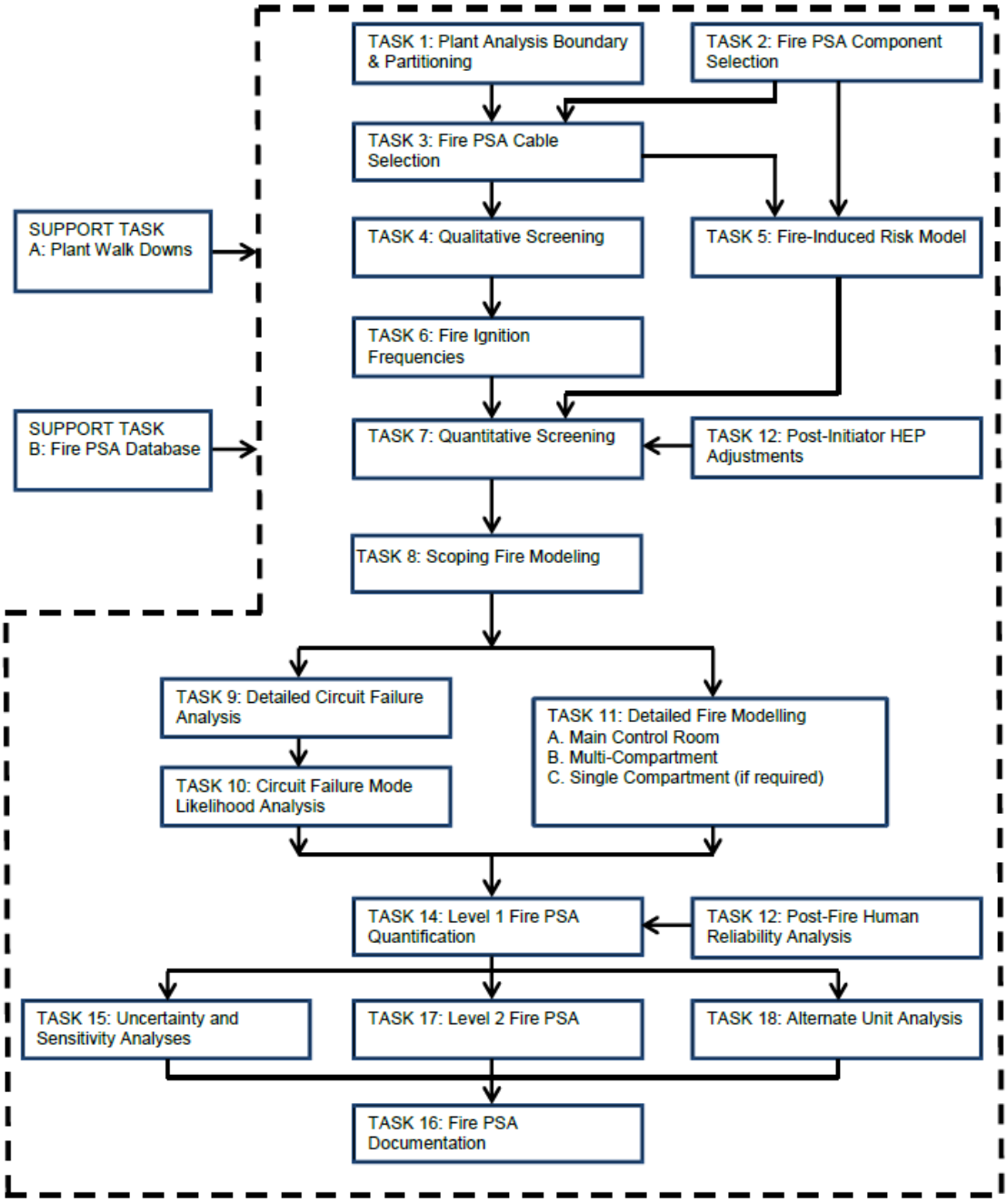


Figure 9: Fire PSA Tasks

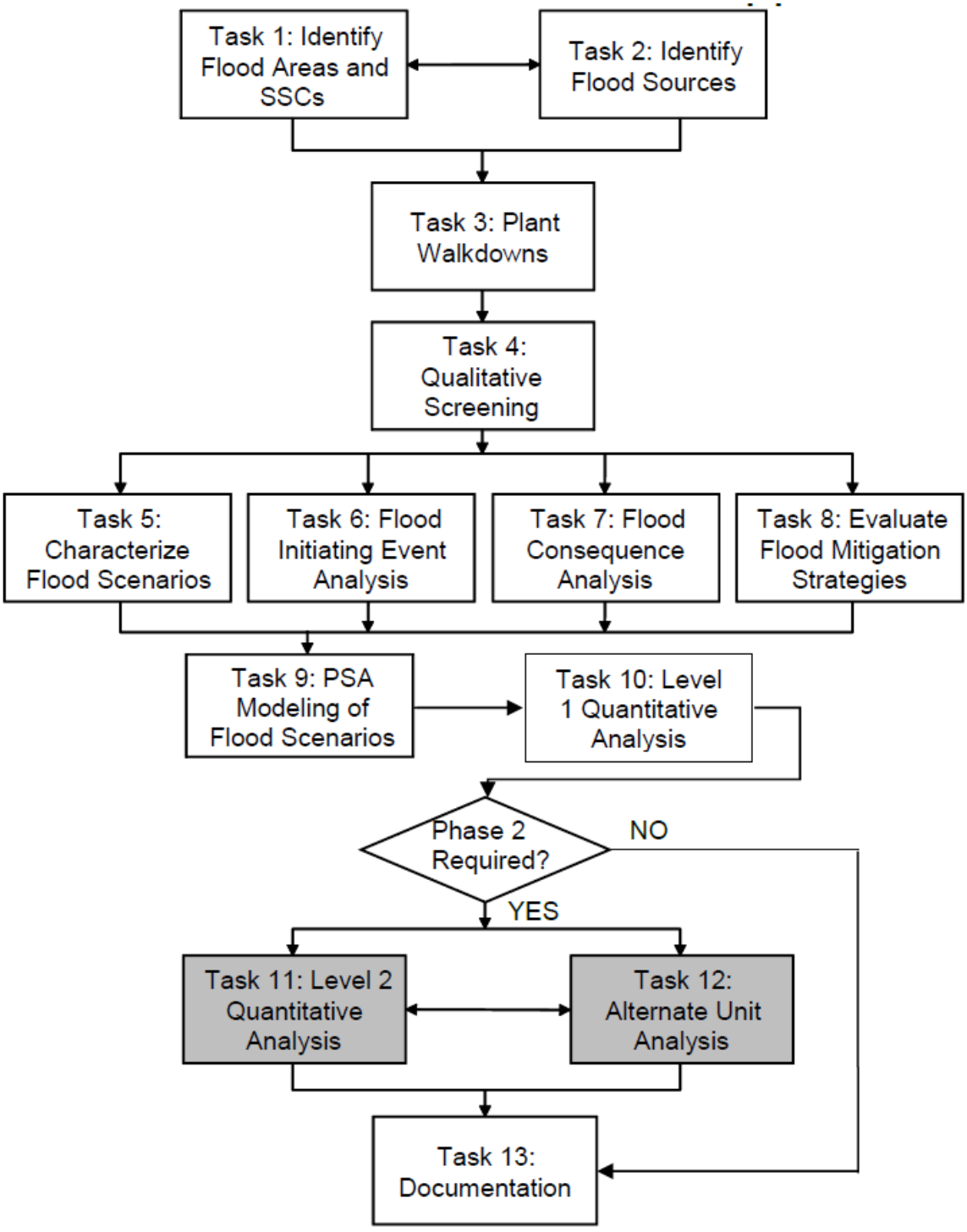


Figure 10: Internal Flood Phase 1 Tasks

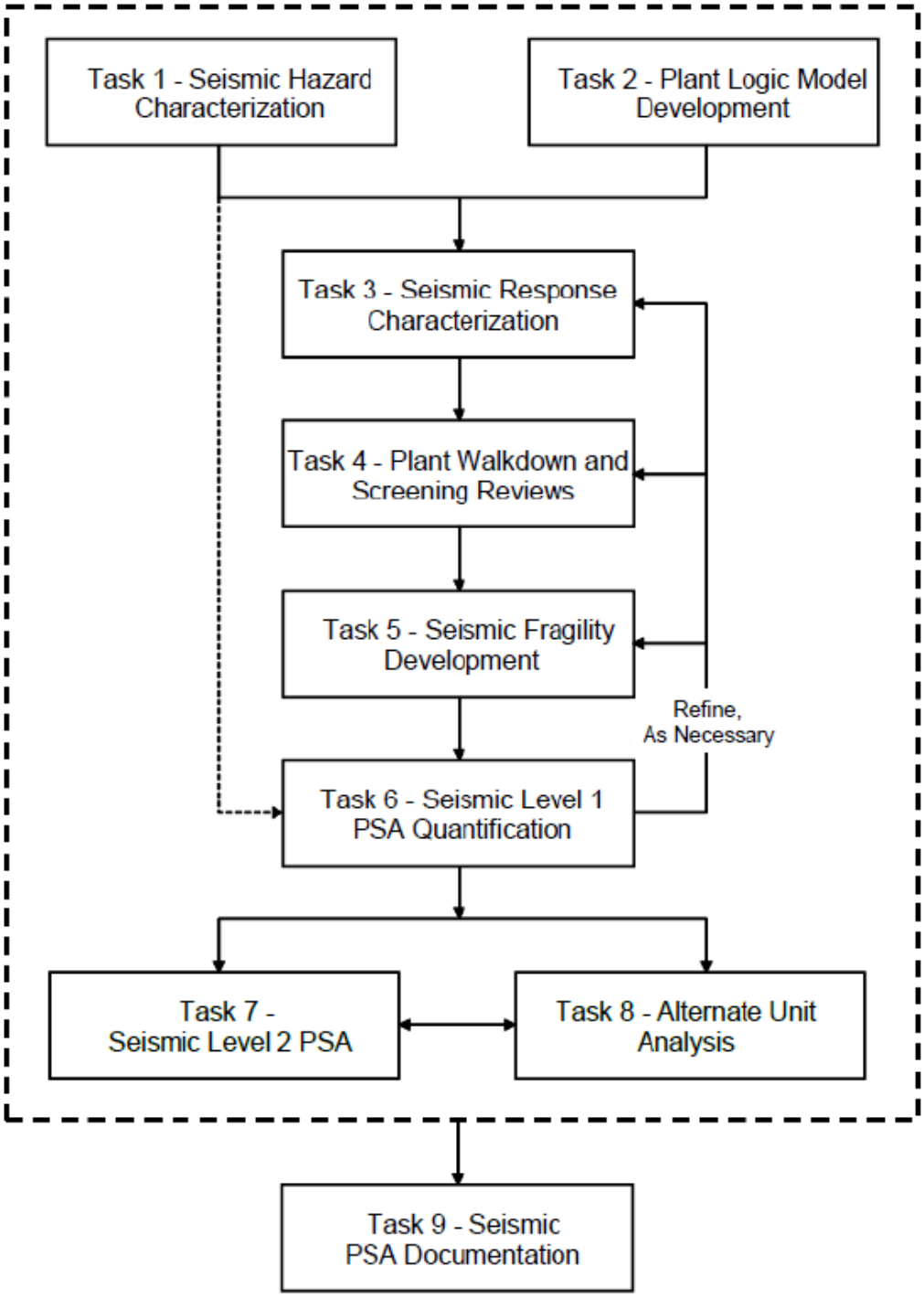


Figure 11: Seismic PSA Tasks

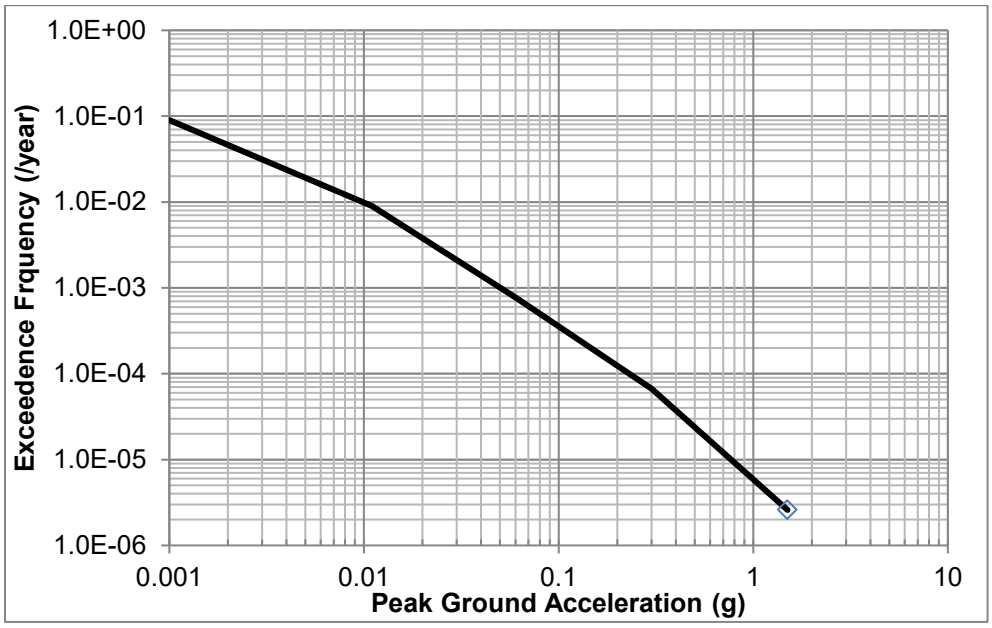


Figure 12: Example Seismic Hazard Curve

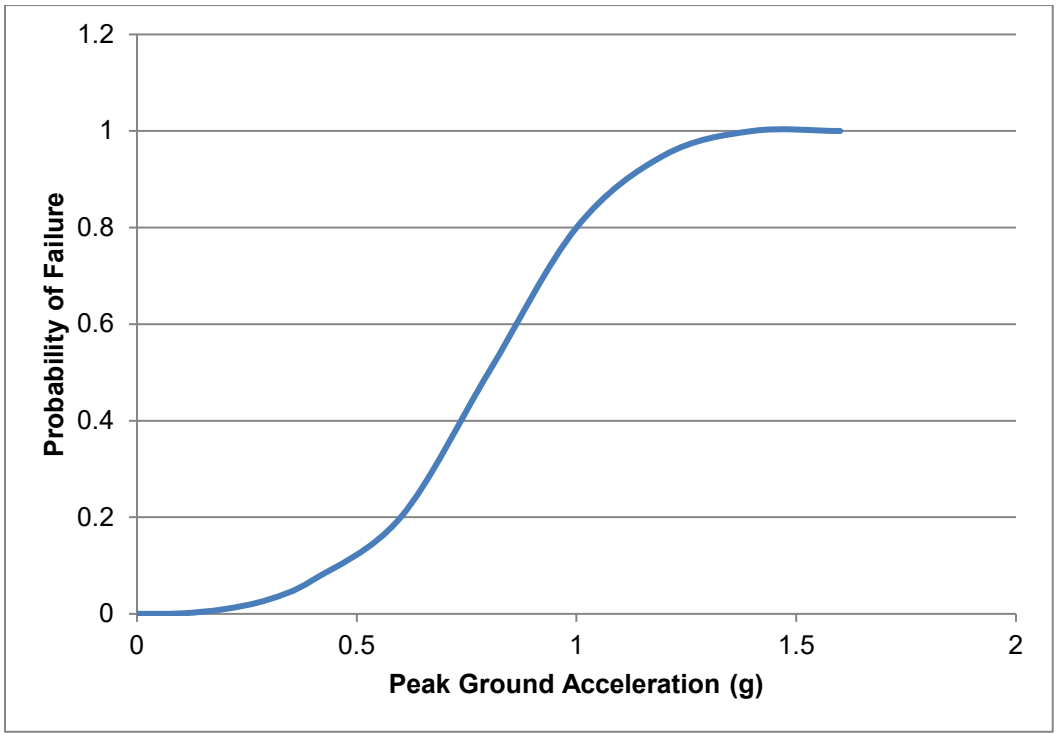


Figure 13: Example Fragility Curve

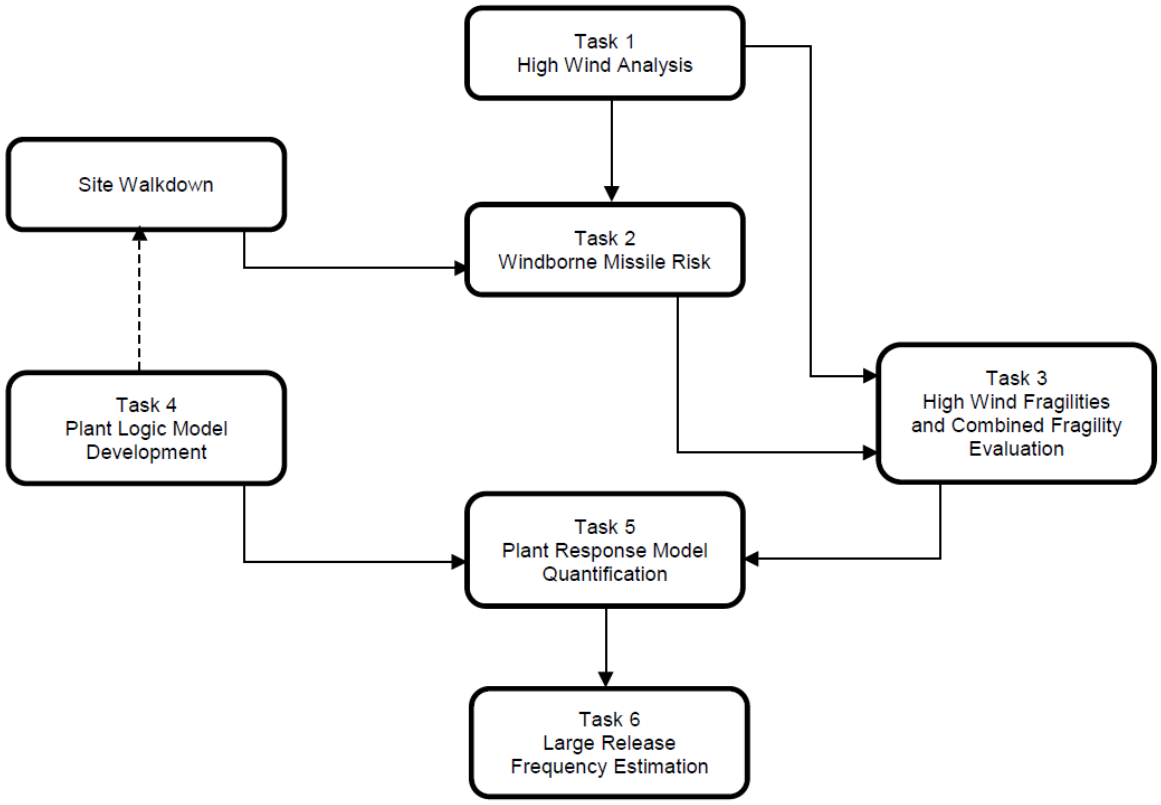


Figure 14: Overall OPG High Wind PSA Method

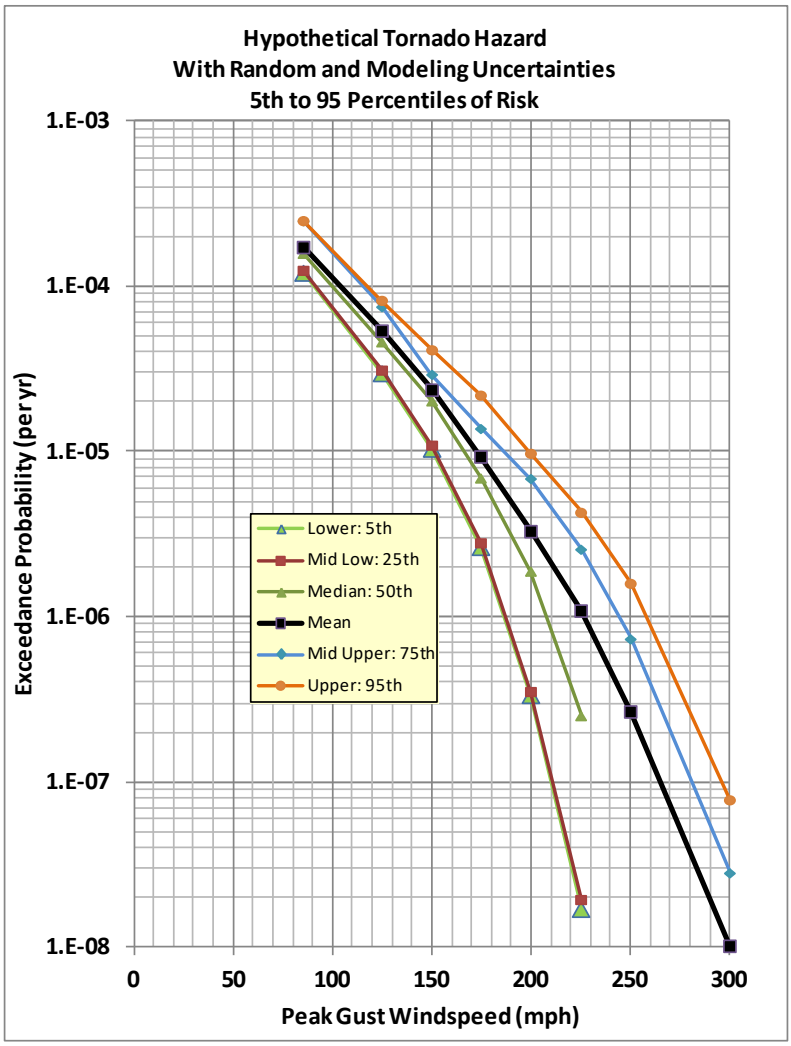


Figure 15: Example of High Wind Hazard Curves

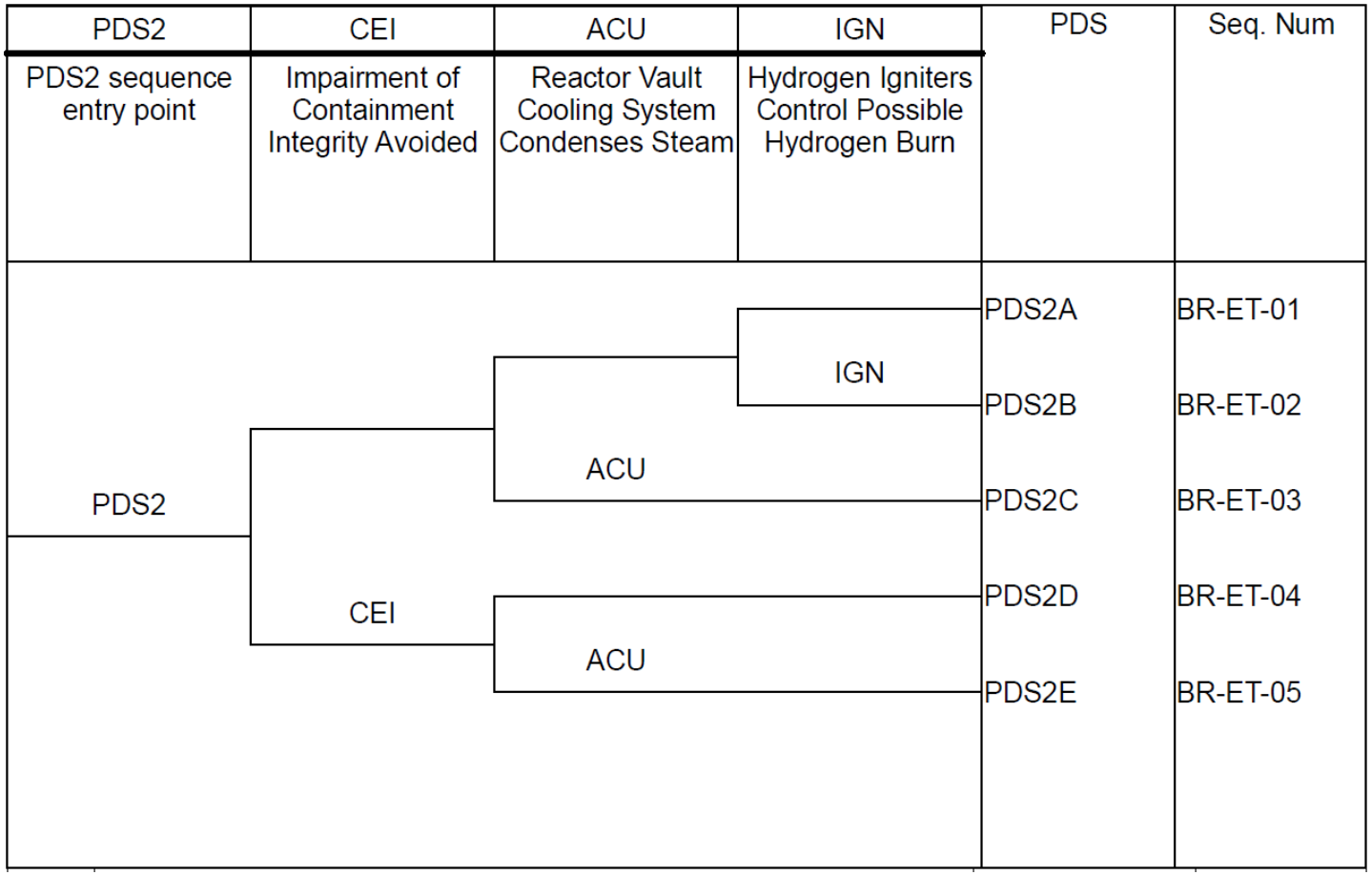


Figure 16: Darlington NGS Bridging Event Tree

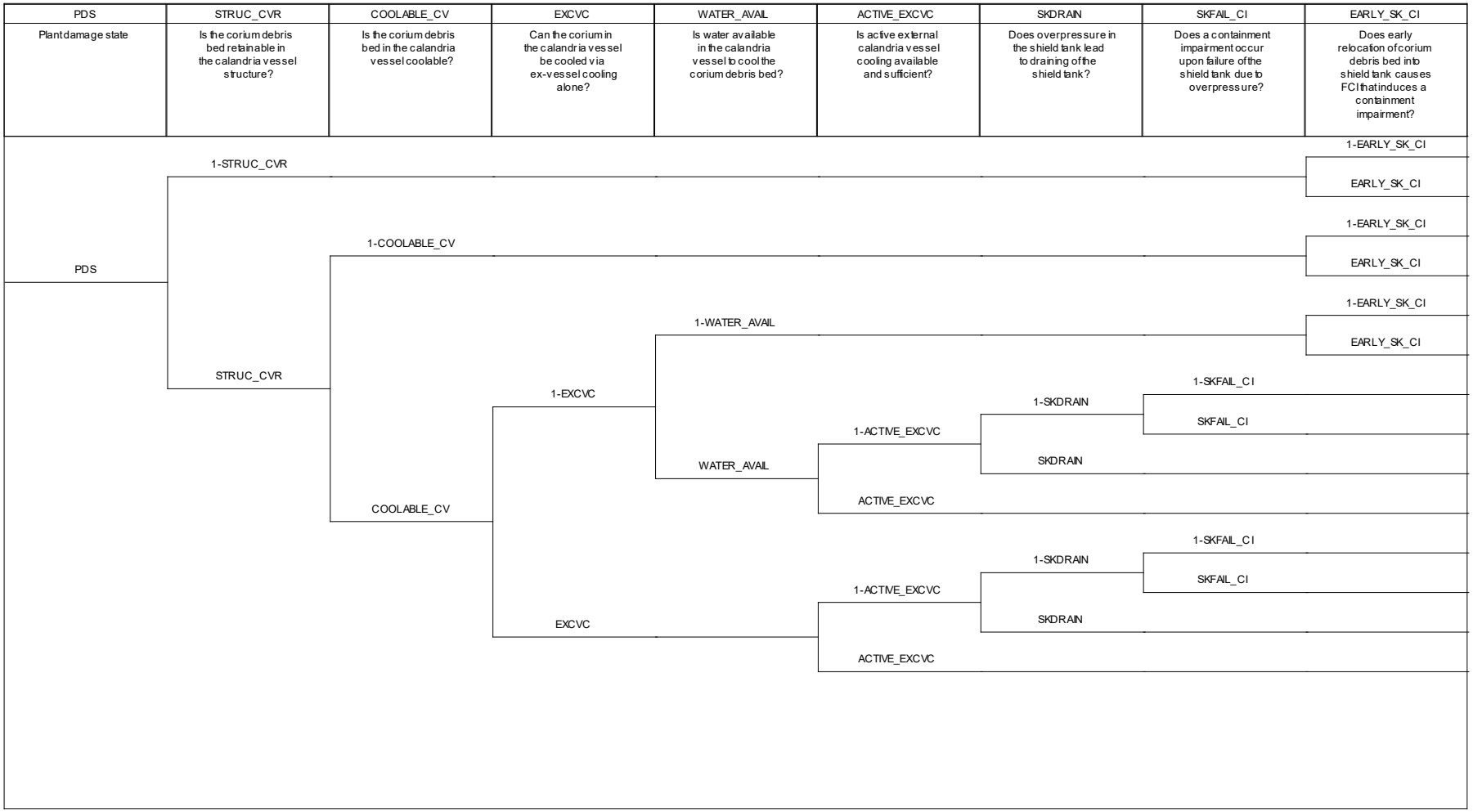


Figure 17: Simplified Containment Event Tree

Table 1: OPG Safety Goals

CRITERIA	AVERAGE RISK (PER YEAR)	
	Administrative Safety Goal	Safety Goal
Severe Core Damage (per unit)[1]	10^{-5}	10^{-4}
Large Release (per unit)[2]	10^{-6}	10^{-5}

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

[2] Large Release is a release greater than 1E14 Bq of Cs-137. OPG's Safety Goals are described in Reference [R-4].

Table 2: Quantitative Hazard Screening Criteria

Criteria	Description	Direct Containment Bypass or Failure (Note)	Applicability of Screening Criteria to Reactor and/or Non-Reactor Sources
QN1	SCDF < 10^{-6}/yr	No	QN-1 and QN-2 apply only to the reactor sources and not to the non-reactor sources
QN2	Design Basis Hazard Frequency, < 10^{-5}yr and CCDP < 0.1	No	
QN3	SCDF < 10^{-7}/yr	Yes	This QN applies to the reactor sources only. An equivalent QN for non-reactor sources of LRF < 10^{-7}/yr will be considered
QN4	Design Basis Hazard Frequency, < 10^{-6}/yr and CCDP < 0.1	Yes	This QN applies to the reactor sources only. An equivalent QN for non-reactor sources will be considered as follows: Design Basis Hazard Frequency, < 10^{-6}/yr and conditional large release probability (CLRP) < 0.1
QN5	IE or Hazard may be screened out if can be shown that their frequency is < 10^{-7} year.	Not Applicable	This QN applies to both the reactor and the non-reactor sources.

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

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

Human-Induced Hazard Description	Screening
Small Airplane Crash	Screened out
Military Jet Crash	Screened out
Large Airplane Crash	Screened out
Train Accidents causing Toxic Chemical Release	Screened out
Train Accidents causing Explosion	Screened out
Road Transportation Accidents	Screened out
Small Marine Transportation Accidents	Screened out
Large Marine Transportation Vessels Accidents	Screened out
Stationary Nuclear Accidents	Screened out
Stationary Non-Nuclear Accidents causing Toxic Chemical Release	Screened out
Stationary Non-Nuclear Accidents causing Explosions	Screened out
Industrial Underground Blasts	Screened out
Industrial Dusts	Screened out
External Fires	Screened out
Orbital Debris Crash	Screened out

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

Natural Hazard Description	Screening
Earthquake	Screened in
Slope Instability	No Hazard
Subsidence	No Hazard
Soil Failure	No Hazard
Probable Maximum Flood (PMF)	Screened out
Floods due to Runoffs	Screened out
Floods due to Rivers	No Hazard
Floods due to Waves	Screened out
Floods due to Seiche	No Hazard
Floods due to Tsunami	No Hazard
Floods due to Ponds and Dams	No Hazard
Floods due to Ice-Jamming	Screened out
Extreme Temperatures	Screened In
Snow/Snowpack	Screened out
Freezing Rain	Screened out
Avalanche	No Hazard
Ice Storm	Screened out (Impact on Class III) Screened in (Impact on PSVS)
Tornado/ High Wind / Hurricane	Screened in
Lightning	Screened out
Meteorites	Screened out
Geomagnetic Storms and Solar Flares	Screened in
Animals	Screened out

Table 5: Summary of Criteria Applied for Screening of Human-Induced External Hazards for Non-Reactor Sources - IFB

Human-Induced External Hazard	Screening
Large Aircraft Impact	Screened Out
Small Aircraft Impact	Screened Out
Train Accidents causing Explosion	Screened Out
Train Accidents causing Toxic Chemical Release	Screened Out
Road Transportation and Traffic Accidents	Screened Out
Marine Transportation Hazards	Screened Out
Stationary Nuclear Accident Stationary Non-Nuclear Accidents causing Toxic Chemical Release Stationary Non-Nuclear Accidents causing Explosions)	Screened Out
Industrial Underground Blasts Industrial Dusts	Screened Out
External Fires	Screened Out
Orbital Debris Crashes	Screened Out

Table 6: Summary of Criteria Applied for Screening of Natural Hazards for Non-Reactor Sources – IFB

Natural Hazard Description	Screening
Earthquake	Screened in
Slope Instability	No Hazard
Subsidence	No Hazard
Soil Failure	No Hazard
Probable Maximum Flood (PMF)	Screened In
Floods due to Runoffs	Screened out
Floods due to Rivers	No Hazard
Floods due to Waves	Screened out
Floods due to Seiche	No Hazard
Floods due to Tsunami	No Hazard
Floods due to Ponds and Dams	No Hazard
Floods due to Ice-Jamming	Screened out
Extreme Temperatures	Screened In
Snow/Snowpack	Screened In
Freezing Rain	Screened In
Avalanche	No Hazard
Ice Storm	Screened In
Tornado/ High Wind / Hurricane	Screened in
Lightning	Screened out
Meteorites	Screened out
Geomagnetic Storms and Solar Flares	Screened in
Animals	Screened out

Table 7: Summary of Criteria Applied for Screening of Human-Induced External Hazards for Non-Reactor Sources - UFDS

Human-Induced External Hazard	Screening
Large Aircraft Impact	Screened Out
Small Aircraft Impact	Screened Out
Train Accidents causing Explosion	Screened Out
Train Accidents causing Toxic Chemical Release	Screened Out
Road Transportation and Traffic Accidents	Screened Out
Marine Transportation Hazards	Screened Out
Stationary Nuclear Accidents Stationary Non-Nuclear Accidents causing Toxic Chemical Release Stationary Non-Nuclear Accidents causing Explosions	Screened Out
Industrial Underground Blasts Industrial Dusts	Screened Out
External Fires	Screened Out
Orbital Debris Crashes	Screened Out

Table 8: Summary of Criteria Applied for Screening of Natural Hazards for Non-Reactor Sources – UFDS

Natural Hazard Description	Screening
Earthquake	Screened Out
Slope Instability	No Hazard
Subsidence	No Hazard
Soil Failure	No Hazard
Probable Maximum Flood (PMF)	Screened out
Floods due to Runoffs	Screened out
Floods due to Rivers	No Hazard
Floods due to Waves	Screened out
Floods due to Seiche	No Hazard
Floods due to Tsunami	No Hazard
Floods due to Ponds and Dams	No Hazard
Floods due to Ice-Jamming	Screened out
Extreme Temperatures	Screened Out
Snow/Snowpack	Screened out
Freezing Rain	Screened out
Avalanche	No Hazard
Ice Storm	Screened out
Tornado/ High Wind / Hurricane	Screened out
Lightning	Screened out
Meteorites	Screened out
Geomagnetic Storms and Solar Flares	Screened out
Animals	Screened out

Table 9: Darlington At-Power Internal Events PSA Initiating Events

Category	Label	Description
Forced Shutdown	FSD	All reactor trips not included in other initiating events
LOCA	LOCA1A	A rupture within the capacity of the D_2O transfer system and above the lower LOCA threshold (discharge rate 1-12 kg/s)
	LOCA1A-OC	(discharge rate 1-12 kg/s outside containment)
	LOCA1B	A rupture within the capacity of the D_2O feed pump but beyond that of the D_2O transfer system (discharge rate 12-40 kg/s)
	LOCA1B-OC	(discharge rate 12-40 kg/s outside containment)
	LOCA1C	A rupture within the capacity of two D_2O feed pumps but beyond the capacity of one D_2O feed pump (discharge rate 40-70 kg/s)
	LOCA2A	Small breaks within the capacity of the auxiliary moderator heat sink (break discharge rate 70-220 kg/s)
	LOCA2B	Small breaks (discharge rate 220-1000 kg/s)
	LOCA3	Transition breaks. Partial breaks which exhibit system response characteristics in between those of small and large breaks (initial discharge rate 1000-2000 kg/s)
	LOCA4	Large breaks which lead to significant flow degradation in the core (initial discharge rate >2000 kg/s)
	LOCATOP	A LOCA2 size break in HT piping connected to the top of the pressurizer
	LOCA1-SF	Stagnation feeder break in LOCA1 range
	LOCA2-SF	Stagnation feeder break in LOCA2 range
	LOCA2-SDC	A LOCA2 size break in the PHT-SDC interface piping inside an SDC room
Pressure Tube Rupture	PTF	Pressure tube break resulting in a discharge rate in excess of 1 kg/s
Pressure Tube Leak	PTL	Pressure tube break resulting in a discharge rate of less than 1 kg/s
End-fitting Failure	EFL1WAGA	LOCA1A size break inside annulus gas bellows
	EFL1WAGB	LOCA1B size break inside annulus gas bellows
	EFL1WAGC	LOCA1C size break inside annulus gas bellows
	EFL1OAGA	LOCA1A size break outside annulus gas bellows
	EFL1OAGB	LOCA1B size break outside annulus gas bellows
	EFL1OAGC	LOCA1C size break outside annulus gas bellows
	EFL1FMIA	LOCA1A size break involving the fuelling machine
	EFL1FMIB	LOCA1B size break involving the fuelling machine
	EFL1FMIC	LOCA1C size break involving the fuelling machine
	EFL2WAG	LOCA2 size break inside annulus gas bellows
	EFL2OAG	LOCA2 size break outside annulus gas bellows
	EFL2FMI	LOCA2 size break involving the fuelling machine
Steam Generator Tube Rupture	SGTB1	SG single tube break (initial discharge rate 1 kg/s – 12 kg/s)
	SGTB2	SG multiple tube break (>12 kg/s)
Loss of HT Pressure Control (Low)	LRVO	One or more liquid relief valves fail open (base event)
	FVFC	Both D_2O feed valves fail closed (base event)

Category	Label	Description
	SBVO	Any pressurizer steam bleed or relief valve fails open
Loss of HT Pressure Control (High)	PHFO	Pressurizer heaters energized spuriously
	BVFC	Both HT bleed valves fail closed
	FVFO	Any D_2O feed valve fails open
	FP2S	Inadvertent start-up of inactive feed pump
	BCLCVFC	Bleed condenser level control valves fail closed
	PSBVFC	Pressurizer steam bleed valves fail closed when required open
HT Pressure and Inventory Control Failures	D2OFDL	Pipe break in D_2O feed system upstream of check valve NV61
	FBSICL	Feed/bleed system pipe break inside containment
	XSPR	Bleed condenser spray valve CV12 opens spuriously
HT Pump Trip	HTPT1	Pump trip in 2/2 mode
Channel Flow Blockage	LFB	Channel flow reduced by 70% or more
Moderator Failure	LOCOOL	Loss of moderator cooling resulting in setback
	SLOMA	Loss of moderator inventory within capacity of moderator D_2O recovery system (discharge rate 1-70 kg/s)
	LLOMA	Loss of moderator inventory beyond capacity of moderator D_2O recovery system (discharge rate >70 kg/s)
Loss of End Shield Cooling	LOESHS	Loss of end shield heat sink
	LOESF	Total loss of end shield flow
	LOESI1	Non-isolable pressure boundary rupture
	LOESI2A	Rupture upstream of V15/16 where isolation leads to loss of circulation
	LOESI2B	Rupture upstream of V15/16 where isolation does not lead to loss of circulation
Steam Line Break	SSLB1	Small break that requires reactor shutdown but does not cause global harsh environment
	SSLB3	A Feedwater Line Break downstream of the last check valve before the steam generator (assumed to be in SG1 flowpath)
	100SBH-ADJN	100% Steam Balance Header (SBH) Break in a unit adjacent to the analyzed unit, North of Column Line 11 with potential for in-plant environmental consequences
	100SBH-U3	Unit 3 100% Steam Balance Header Break in a unit remote to the analyzed unit, North of Column Line 11 with potential for in-plant environmental consequences
	100SBH-U4	Unit 4 100% Steam Balance Header Break in a unit remote to the analyzed unit, North of Column Line 11 with potential for in-plant environmental consequences
	100SBH-U2N	Unit 2 100% Steam Balance Header Break, North of Column Line 11 with in-plant environmental consequences
	SRV	Any ISRV, ASDV or CSDV opens spuriously
Loss of Feedwater to Steam Generators	LOFWB	LOFW resulting in reactor trip but greater than 3% full flow remains
	LOFWC	LOFW to less than 3% full flow
Feedwater Line Break	SFLB1	Break resulting in reactor shutdown but with sufficient water remaining to remove decay heat

Category	Label	Description
	100LFB-ADJN	100% Feedwater Line Break in an Adjacent Unit, North of Column Line 11
	100FLB-U3	Remote Unit (Unit 3) 100% Feedwater Line Break, North of Column Line 11
	100FLB-U4	Remote Unit (Unit 4) 100% Feedwater Line Break, North of Column Line 11
	100FLB-U2N	Unit 2 100% Feedwater Line Break, North Column Line 11, Causing Total Loss of Feedwater
	100FLB-U2S	Unit 2 100% Feedwater Line Break, South of Column Line 11, Causing Total Loss of Feedwater
	FLBSG	Isolable break downstream of LCVs resulting in total loss of feedwater to one steam generator (assumed to be in SG1 flowpath)
	FLBCOND1	Break in condensate system resulting in total loss of feedwater
Turbine Trip	TT	All turbine trips not included in other initiating events
Loss of Condenser Vacuum	LOVAC	Loss of condenser vacuum resulting in turbine trip
High Pressure Reheater Drains Line Break to Steam Generator	RDLB	Break in lines between steam generators and second check valve (assumed to be in SG1 flowpath)
Loss of Condensate Flow	LOCOND	Total loss of condensate flow to deaerator
Unplanned Bulk Increase in Reactivity	UFBIR	Unplanned fast (>0.2 mk/s) bulk increase in reactivity
	USBIR	Unplanned slow (<0.2 mk/s) bulk increase in reactivity
Unplanned Regional Increase in Reactivity	URIR	Local neutron overpower
Loss of Computer Control	WDTOX	Controlling computer stall
	DCCF	Dual computer failure
	DCCUF	Unsafe failure of DCC leading to reactor power increase
	HTPF SGLCF SGPCF MTCF DLCF	Failure 'off' of an individual control program on both computers
Loss of Low Pressure Service Water System	LOLPSW	Total loss of LPSW flow out of header L205
	LOPH	Loss of flow to pumphouse
	LOTH	Loss of flow to turbine hall
Loss of Recirculated Cooling Water System	LORCW	Total loss of RCW flow
Loss of Powerhouse Upper Level Service Water	LOPULSW	Total loss of PULSW flow
Loss of Instrument Air	TLOIA	Total loss of instrument air out of line L17
Loss of Cooling to F/M in Transit	LOFMCIT	Loss of cooling to fuelling machine in transit
Loss of Bulk Electricity Supply	LOBES	Loss of Bulk Electrical Supply (BES)
Loss of Switchyard	LOSWYD	Loss of both switchyard buses BU1 and BU2
	LOCL4	Total loss of Unit Class IV 13.8 kV power

Category	Label	Description
Loss of Power to Unit Class IV 13.8 kV Bus	LOBU1	Loss of power to Unit Class IV 13.8 kV bus BU1
	LOBU2	Loss of power to Unit Class IV 13.8 kV bus BU2
	LOBU3	Loss of power to Unit Class IV 13.8 kV bus BU3
	LOBU4	Loss of power to Unit Class IV 13.8 kV bus BU4
Partial Loss of Unit Class IV Power	FS1CB2	Loss of Unit Class IV 13.8 kV buses BU1 and BU3 due to 1CB2 failing short
	FS2CB2	Loss of Unit Class IV 13.8 kV buses BU2 and BU4 due to 2CB2 failing short
Partial Loss of Unit Class III Power	LOBU7	Loss of power to Unit Class III 4.16 kV bus BU7
	LOBU8	Loss of power to Unit Class III 4.16 kV bus BU8
	LOBU13	Loss of power to Unit Class III 600 V bus BU13
	LOBU14	Loss of power to Unit Class III 600 V bus BU14
	LOBU15	Loss of power to Unit Class III 600 V bus BU15
	LOBU16	Loss of power to Unit Class III 600 V bus BU16
Partial Loss of Unit Class II 120 V Power	LOBUA3	Loss of Unit Class II 120 V ac bus BUA3
	LOBUB3	Loss of Unit Class II 120 V ac bus BUB3
	LOBUC3	Loss of Unit Class II 120 V ac bus BUC3
Partial Loss of Unit Class II 45 V Power	LO45VA	Loss of Unit Class II 45 V dc at panel 2383-11
	LO45VB	Loss of Unit Class II 45 V dc at panel 2859-21
	LO45VC	Loss of Unit Class II 45 V dc at panel 3485-C1
Partial Loss of Unit Class I 48 V Power	LOBUA4	Loss of Unit Class I 48 V dc bus BUA4
	LOBUB4	Loss of Unit Class I 48 V dc bus BUB4
	LOBUC4	Loss of Unit Class I 48 V dc bus BUC4
	LOBUA141	Loss of Unit EPS 48 V dc bus BUA141
	LOBUB141	Loss of Unit EPS 48 V dc bus BUB141
Loss of Forebay	FOREBAY	Loss of Forebay leading to loss of Circulating Water System; may also lead to loss of Low Pressure Service Water and/or Emergency Service Water
ECI Blowback	BLOWBACK	Blowback of HTS D_2O at high pressure outside containment via ECI piping
Powerhouse Freeze	PHFREEZE	Spurious opening of powerhouse venting dampers during extreme cold outside condition.
ESW Blowback	ESW-BLBK	Blowback of HTS D_2O at high pressure outside containment via ESW piping

Table 10: DARA Fuel Damage Categories

FDC	Definition	Typical Events in FDC
1	Rapid loss of core structural integrity.	Positive reactivity transient and failure to shutdown.
2	Slow loss of core structural integrity.	Loss of coolant accident (LOCA) with failure of ECIS and failure of moderator heat sink.
3	Moderator required as heat sink in the short-term (< 1 hr after reactor trip).	LOCAs of LOCA2B size or greater and failures of ECIS on demand or during mission.
4	Moderator required as heat sink in the intermediate term (1 to 24 hr after reactor trip).	LOCAs of LOCA2A size or greater and failure of Emergency Coolant Recovery (ECR). Total loss of secondary side heat sink with ECI successful.
5	Moderator required as heat sink in the long-term (> 24 hr after reactor trip).	LOCA1 and failures of D_2O make up and ECR.
6	Temporary loss of cooling to fuel in many channels.	LOCA4.
7	Single channel fuel failure with sufficient release of steam or radioactivity to initiate automatic containment button-up.	In-core LOCA with end-fitting release End-fitting LOCA2B and fuel ejection. LOCA2A stagnation feeder break.
8	Single channel fuel failure with insufficient release of steam or radiation activity to initiate automatic containment button-up.	Large flow blockage (no end-fitting release). LOCA1 stagnation feeder break.
9	LOCAs with no fuel failure (ECIS successful); potential for significant economic impact.	LOCA2A, LOCA2B and LOCA3. LOCA1 with no D_2O makeup.

Table 11: List of Systems Modelled by Fault Trees

System Name	L1 At-Power	L1 Outage	Level 2 At-Power
Heat Transport Liquid Relief, Pressure and Inventory Control and D_2O Storage Systems	Y	Y	*
Heat Transport Circulation System And Heat Transport Pump Gland Seal LOCA	Y	Y	*
Shutdown Cooling System	Y	Y	*
Moderator System	Y	Y	*
Boiler Feedwater System	Y	Y	*
Condensate and Makeup Systems	Y	Y	*
Steam Generators Emergency Cooling System	Y	Y	*
Steam Relief and Bypass System	Y	Y	*
Digital Control Computer System	Y	Y	*
OH180 Programmable Controller and PK Buffer System	Y	N	*
Class IV Power Distribution System	Y	Y	*
Class III Power Distribution System	Y	Y	*
Class II Power System	Y	Y	*
Class I Power System	Y	Y	*
Emergency Power Supply System	Y	Y	*
Standby Generators	Y	Y	*
Emergency Power Generators System	Y	Y	*
Low Pressure Service Water System	Y	Y	*
Recirculated Cooling Water System	Y	Y	*
Powerhouse Upper Level Service Water System	Y	Y	*
Emergency Service Water System	Y	Y	*
Unit Instrument Air System	Y	Y	*
Common Instrument Air System	Y	Y	*
Reactivity Control System	Y	N	*
Shutdown System No. 1	Y	N	*
Shutdown System No. 2	Y	N	*
Emergency Coolant Injection System	Y	Y	*
Emergency Coolant Injection System: Blowback	Y	N	*
Inter-Unit Feedwater Tie System	Y	Y	*
D_2O Recovery and Transfer Systems	Y	Y	*
Room Air Conditioning System	Y	Y	*
Hostile Environment Events (including Powerhouse Emergency Venting System)	Y	Y	*
Annulus Gas System	Y	N	*
Emergency Mitigating Equipment	Y	Y	*
Containment Envelope Integrity (CEI) System	N	N	Y
Reactor Vault Atmosphere Cooling System	N	N	Y

System Name	L1 At-Power	L1 Outage	Level 2 At-Power
Post-Accident Hydrogen Ignition System	N	N	Y
Emergency Filtered Air Discharge System	N	N	Y**
Containment Filtered Venting System	N	N	Y**

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

** The system is developed as a fault tree model, however, it is not included in the Level 2 At-Power baseline integrated model.

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

Table 12: DARA-L1O Plant Operational State Definition

Input Parameter	Plant Operational State (POS)		
	A	C	D
GSS	OPGSS or RBGSS	OPGSS or RBGSS	DGSS or MD-RBGSS
Moderator State	Calandria Full	Calandria Full	Calandria Drained
HTS Inventory Level	Full	LLDS	LLDS
HTS Boundary Configuration	Closed	Closed or Open or Abnormal IC / OC	Closed or Open or Abnormal IC / OC
HTS Temp (Nominal)	60°C	30°C	30°C
HTS Pressure	Pressurized	Depressurized	Depressurized
Primary Heat Sink (Circulation)	HTS Pumps or SDC Pumps Note 1	SDC Pumps	SDC Pumps
Primary Heat Sink (Heat Removal)	SDC HXs, Bleed Cooler, or Boiler Blowdown Note 2	SDC HXs	SDC HXs
Backup Heat Sink (Circulation)	Various (SDC, NC, HTS Pumps and Steam Generators, Bleed Cooler)	Various (SDC, NC, HTS Pumps and Steam Generators)	Various (SDC, NC)
Backup Heat Sink (Heat Removal)			
Time after Shutdown at Start of POS (days for decay heat load)	1.0	4.8	28.4

Note 1: If HTS pumps are the primary shutdown heat sink circulation method, then SDC pumps are the backup (and vice versa).

Note 2: Boiler blowdown can only be used later in the outage. The limiting decay heat load of 1.0 day after shutdown is used here for the general definition of POS A; however, the shutdown heat sinks fault tree includes modelling to allow for use of boiler blowdown for a fraction of POS A that represents time later in the outage

Table 13: Initiating Events (IEs) for Darlington Level 1 Outage PSA

Outage IE Label	IE Definition	POS Applicability		
		A	C	D
Loss of Moderator Inventory				
LOMA	Loss of moderator inventory leading to a drained moderator	Y	N	N
Failures of the HT or SDC System Boundaries				
LOCA1	Small non-isolatable breaks inside containment from a pressurized HTS, within the capacity of two D_2O feed pumps	Y	N	N
LK1A	Small non-isolatable leak inside containment from a depressurized HTS, within the capacity of D_2O transfer	N	Y	Y
LK1B	Small non-isolatable leak inside containment from a depressurized HTS, within the capacity of one D_2O feed pump	N	Y	Y
LK1C	Small non-isolatable leak inside containment from a depressurized HTS, within the capacity of two D_2O feed pumps	N	Y	Y
LLOCA	Non-isolatable breaks inside containment from a pressurized HTS, beyond the capacity of two D_2O feed pumps	Y	N	N
LOCA2-OUTAGE	Non-isolatable breaks inside containment from a depressurized HTS, beyond the capacity of two D_2O feed pumps	N	Y	Y
LOCA1-OC	Small breaks outside containment from a pressurized HTS, within the capacity of one D_2O feed pump	Y	N	N
LK1-OC	Small leak outside containment from a depressurized HTS, within the capacity of one D_2O feed pump	N	Y	Y
LK1-SDCIS	Leak in piping within the SDC system when in service, within the capacity of two D_2O feed pumps	Y	Y	Y
LLOCA-SDCIS	Large break in piping within the SDC system when in service, beyond the capacity of two D_2O feed pumps	Y	Y	Y
PTF	Pressure tube failure	Y	N	N
PTL	Pressure tube leak (initial discharge rate less than 1 L/s)	Y	Y	Y
SGTB1	Steam generator tube break within the capacity of two D_2O feed pumps	Y	N	N
SGTB2	Steam generator tube break beyond the capacity of two D_2O feed pumps	Y	N	N
SDCHXTB1	SDC HX tube break within the capacity of two D_2O feed pumps	Y	Y	Y
SDCHXTB2	SDC HX tube break beyond the capacity of two D_2O feed pumps	Y	N	N
ICEPLUGS	Failure of liquid nitrogen supply to all ice plugs	N	Y	Y
Intrinsic System Failures for Primary Heat Sink				
SDC-COOL	Failure of SDC HXs to remove heat	Y	Y	Y
SDC-FLOW	Loss of HTS forced circulation using the SDC pumps	Y	Y	Y
2HTPT	2 or more heat transport pumps trip (2 in one loop)	Y	N	N
SDC-INV-LLDS	Loss of HTS inventory in Low Level Drained State (LLDS) (no rupture) leads to failure of forced circulation using SDC pumps	N	Y	Y
SDC-MV	Spurious closure of SDC isolating Motorized Valve (MV)	Y	Y	Y
Pressure and Inventory Control System Failures				
LOPIC	Failure of HTS pressure and inventory control (no pressure boundary failure) while HTS is pressurized in solid mode	Y	N	N

Outage IE Label	IE Definition	POS Applicability		
		A	C	D
PIC-LOC	Loss of HTS inventory through HTS P&IC pressure boundary while pressurized in solid mode	Y	N	N
Large Pipe Breaks or Other Events in Operating Units with Effects on Outage Unit				
100SBH-ADJN	100% Steam Balance Header (SBH) Break in a unit adjacent to the analyzed unit, North of Column Line 11 with potential for in-plant environmental consequences	Y	Y	Y
100FLB-ADJN	Adjacent Unit 100% Feedwater Line Break, North of Column Line 11	Y	Y	Y
100SBH-U3	100% SBH Break in remote Unit 3, North of Column Line 11 with potential for in-plant environmental consequences	Y	Y	Y
100SBH-U4	100% SBH Break in remote Unit 4, North of Column Line 11 with potential for in-plant environmental consequences	Y	Y	Y
100FLB-U3	100% Feedwater Line Break in remote Unit 3, North of Column Line 11 with potential for in-plant environmental consequences	Y	Y	Y
100FLB-U4	100% Feedwater Line Break in remote Unit 4, North of Column Line 11 with potential for in-plant environmental consequences	Y	Y	Y
EVAC-CNMT	Internal event, not originating from U2, that leads to an evacuation of the outage unit work areas inside containment	Y	Y	Y
Electrical System Failures				
LOBES	Loss of Bulk Electricity System	Y	Y	Y
LOSWYD	Loss of Switchyard	Y	Y	Y
LOCL4	Loss of Class IV	Y	Y	Y
LOBU1	Loss of power to Unit Class IV 13.8 kV bus BU1	Y	Y	Y
LOBU2	Loss of power to Unit Class IV 13.8 kV bus BU2	Y	Y	Y
LOBU3	Loss of power to Unit Class IV 13.8 kV bus BU3	Y	Y	Y
LOBU4	Loss of power to Unit Class IV 13.8 kV bus BU4	Y	Y	Y
LOBU5	Loss of power to Unit Class IV 13.8 kV bus BU5	Y	Y	Y
LOBU6	Loss of power to Unit Class IV 13.8 kV bus BU6	Y	Y	Y
FS1CB2	Loss of Unit Class IV 13.8 kV buses BU1 and BU3 due to 1CB2 failing short	Y	Y	Y
FS2CB2	Loss of Unit Class IV 13.8 kV buses BU2 and BU4 due to 2CB2 failing short	Y	Y	Y
LOBU7	Loss of power to Unit Class III 4.16 kV bus BU7	Y	Y	Y
LOBU8	Loss of power to Unit Class III 4.16 kV bus BU8	Y	Y	Y
LOBU13	Loss of power to Unit Class III 600 V bus BU13	Y	Y	Y
LOBU14	Loss of power to Unit Class III 600 V bus BU14	Y	Y	Y
LOBU15	Loss of power to Unit Class III 600 V bus BU15	Y	Y	Y
LOBU16	Loss of power to Unit Class III 600 V bus BU16	Y	Y	Y
LOBUA3	Loss of Unit Class II 120 V ac bus BUA3	Y	Y	Y
LOBUB3	Loss of Unit Class II 120 V ac bus BUB3	Y	Y	Y
LOBUC3	Loss of Unit Class II 120 V ac bus BUC3	Y	Y	Y
LO45VA	Loss of Unit Class II 45 V dc at panel 2383-11	Y	Y	Y
LO45VB	Loss of Unit Class II 45 V dc at panel 2859-21	Y	Y	Y
LO45VC	Loss of Unit Class II 45 V dc at panel 3485-C1	Y	Y	Y

Outage IE Label	IE Definition	POS Applicability		
		A	C	D
LOBUA4	Loss of Unit Class I 48 V dc BUA4	Y	Y	Y
LOBUB4	Loss of Unit Class I 48 V dc BUB4	Y	Y	Y
LOBUC4	Loss of Unit Class I 48 V dc BUC4	Y	Y	Y
LOBUA141	Loss of EPS 48 V dc bus BUA141	Y	Y	Y
LOBUB141	Loss of EPS 48 V dc bus BUB141	Y	Y	Y
Failures of Other Support Systems				
LOLPSW	Total loss of low pressure service water	Y	Y	Y
LOPULSW	Total loss of powerhouse upper level service water	Y	Y	Y
LORCW	Total loss of recirculated water flow	Y	N	N
TLOIA	Total loss of instrument air	Y	Y	Y
FOREBAY	Forebay severe condition	Y	Y	Y
ESW-BLBK	Emergency service water blowback	Y	Y	Y

Table 14: Summary of Fuel Damage Categories for DARA-L1O

FDC	Definition	Typical Outage Events in FDC
1-SD	**Rapid loss of core structural integrity.**	**Positive reactivity transient during outage and failure to terminate the event.** [Note 1]
2-SD	**Slow loss of core structural integrity.**	**LOCA with failure of HTS make-up and failure of the moderator heat sink.**
3	Moderator required as heat sink in the short term (< 1 hr after reactor shutdown).	Not applicable to Outage PSA. Unit has been shutdown for greater than 1 hour and therefore the short term moderator heat sink is not required.
4	Moderator required as heat sink in the intermediate term (1 to 24 hr after reactor shutdown).	Not applicable to Outage PSA. Unit has been shutdown for >24 hours and intermediate term moderator heat sink not required.
5-SD	**Moderator required as heat sink in the long term (> 24 hr after reactor shutdown).**	**LOCA1 with failure of D_2O make-up and ECR.**
6	Temporary loss of cooling to fuel in many channels.	Not applicable to Outage PSA.
7-SD	**Single channel fuel failure with sufficient release of steam or radioactivity to initiate automatic containment button-up.**	**In-core LOCA and fuel ejection. Large flow blockage. LOCA1 stagnation feeder break.**
8	Single channel fuel failure with insufficient release of steam or radiation activity to initiate automatic containment button-up.	Not applicable to Outage PSA (single channel events adequately covered by FDC7-SD.
9-SD	**HTS leaks with no fuel failure (ECIS successful); potential for significant economic impact.**	**LOCA1 with failure of D_2O make-up.**

Note 1: Potential initiating events representing inadvertent criticality during an outage have been screened out of DARA-L1O on the basis that they have an extremely low frequency. Similarly, the likelihood of an inadvertent criticality during the mission is assumed to be negligible when compared to the other causes of severe core damage during an outage. Therefore, no DARA-L1O event tree sequences are assigned to the FDC1-SD end state.

Table 15: Seismic Hazard Bins

BIN	Bin Seismic Range (g)	Representative Ground Motion PGA (g)	Seismic Bin Frequency (occ/yr.)
%G1 (Bin 1)	0.01 Note 1 - 0.1	0.03	3.82E-03
%G2 (Bin 2)	0.1 - 0.16	0.13	1.05E-04
%G3 (Bin 3)	0.16 - 0.4	0.25	8.20E-05
%G4 (Bin 4)	0.4 - 0.64	0.51	1.47E-05
%G5 (Bin 5)	0.64 - 0.9	0.76	5.39E-06
%G6 (Bin 6)	0.9 - 1.4	1.12	3.20E-06
%G7 (Bin 7)	1.4 - 2	1.67	1.04E-06
%G8 (Bin 8)	>2	2.20 Note 2	6.20E-07 Note 3

Note 1: The beginning of the first seismic hazard bin was defined as 0.01g PGA. Since the Darlington NGS DBE is 0.08g, little seismic risk contribution was expected below the 0.01g PGA.

Note 2: The representative ground motion value for the final interval is calculated as 1.1 x the lower bound ground motion magnitude of the final interval.

Note 3: The seismic bin frequency of the last seismic interval (%G8) was defined as the exceedance frequency at the beginning of the interval.

Table 16: Summary of Selected Accident Sequence

PDS	Representative Accident Sequence
PDS1	No representative sequence defined.
PDS2A	LOCA2A, with loss of moderator cooling and failure of ECI.
PDS2B	LOCA2A, with loss of moderator cooling and failure of ECI, combined with failure of hydrogen igniters.
PDS2C	LOCA2A, with loss of moderator cooling and failure of ECI, combined with failure of reactor vault Air Conditioning Units (ACUs).
PDS2D	LOCA2A, with loss of moderator cooling and failure of ECI, combined with containment envelope impairment.
PDS2E	LOCA2A, with loss of moderator cooling and failure of ECI, combined with containment envelope impairment and failure of reactor vault ACUs.
PDS3-2U	2-Unit blackout with failure of FW, IUFT, IA, SDC, ESW, ECI.
PDS3-4U	100% steam line break in Unit 2, loss of Class IV and III power and EPS affecting 3 or more units, with PSVS success.
PDS3-4U-PSVS	100% steam line break in Unit 2 with PSVS failure, affecting all at-power units.
PDS4	Gland seal LOCA, failure of ECI and moderator cooling.
PDS4-BLBK	ECI Blowback event with failure of moderator cooling.
PDS5	LOCA2 end fitting failure plus failure of ECI, with the moderator providing a long term heat sink, and failure of containment isolation.
PDS6	Multiple steam generator tube rupture with failure of ECI, with the moderator providing a long term heat sink.

Table 17: Darlington NGS Release Categorization Scheme

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

Note: The prefix ‘D’ refers to Darlington.

Table 18: Summary of DARA Severe Core Damage and Large Release Frequency Results

Model	Severe Core Damage Frequency (occurrences per reactor year)	Large Release Frequency (occurrences per reactor year)
Internal Events At-Power	1.7E-06	7.9E-07
Internal Events Outage	4.7E-07	4.6E-07
Internal Fire At-Power	2.8E-05	9.1E-06
Seismic At-Power	7.4E-06	7.4E-06
Internal Flooding At-Power	4.9E-08	1.3E-08
High Wind At-Power	1.9E-06	1.7E-06
Non-Reactor Sources	N/A	7.1E-08
OPG Safety Goal	1E-04	1E-05
OPG Administrative Safety Goal	1E-05	1E-06

Table 19: DARA Level 1 At-Power Internal Events Fuel Damage Results

Fuel Damage Category	Baseline Predicted Frequency (/yr)
FDC1	<<1E-09
FDC2	1.7E-06
FDC3	1.5E-05
FDC4	2.9E-04
FDC5	7.5E-06
FDC6	4.9E-06
FDC7	9.6E-04
FDC8	2.1E-03
FDC9	2.1E-02
Severe Core Damage Frequency FDC1 + FDC2	1.7E-06

Table 20: Frequencies of Fuel Damage Categories for DARA-L1O

Fuel Damage Category	Plant Operating State	Time-Average Note 1 Frequency (/yr)
FDC2-SD	POS A	3.4E-09
	POS C	3.0E-08
	POS D	9.4E-07
Severe Core Damage Note 2	(all)	9.8E-07

Note 1: Time-average FDC results are on a reactor-year basis, using the weighted duration and outage frequency from the POS analysis.

Note 2: FDC2-SD represents Severe Core Damage for the DARA-L1O model.

Table 21: Plant Damage State Frequency

PDS	Predicted Frequency (occ/yr)
PDS1	2.5E-11
PDS2	9.8E-07
PDS3-2U	4.5E-07
PDS3-4U	2.0E-07
PDS4	3.2E-07
PDS5*	1.1E-03
PDS6*	1.6E-04

*PDS5 and PDS6 sequences are limited core damage sequences.

Table 22: Release Category Frequencies for DARA L2P

Release Category	Baseline Predicted Frequency (occ/yr)
D-RC1	2.9E-07
D-RC2	3.1E-07
D-RC3	2.3E-07
D-RC4*	0
D-RC5	1.4E-07
D-RC6	2.1E-07
D-RC7	1.1E-06
D-RC8*	0

* No sequences above the truncation limit were identified in which a release was predicted in the range of magnitude and timing corresponding to the definitions of RC4 and RC8.

Appendix A: Acronyms

Acronym	Definition
ACU	Air Conditioning Unit
AIM	Abnormal Incident Manual
ASDV	Atmospheric Steam Discharge Valve
BES	Bulk Electrical System
BWR	Boiling Water Reactor
CANDU	CANadian Deuterium Uranium
CCDP	Conditional Core Damage Probability
CDFM	Conservative Deterministic Failure Margin
CEI	Containment Envelope Integrity
CET	Containment Event Tree
CFVS	Containment Filtered Venting System
CLRP	Conditional Large Release Probability
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CSDV	Condenser Steam Discharge Valve
D_2O	Deuterium Oxide (Heavy Water)
DARA	Darlington NGS Probabilistic Safety Assessment
DARA-FIRE	Darlington Internal Fire Probabilistic Safety Assessment
DARA-FLOOD	Darlington Internal Flooding Probabilistic Safety Assessment
DARA-L1O	Darlington Level 1 Outage Internal Events Probabilistic Safety Assessment
DARA-L1P	Darlington Level 1 At-Power Internal Events Probabilistic Safety Assessment
DARA-L2P	Darlington Level 2 At-Power Internal Events Probabilistic Safety Assessment
DARA-SEISMIC	Darlington Seismic Probabilistic Safety Assessment
DARA-WIND	Darlington High Wind Probabilistic Safety Assessment
DBE	Design Basis Earthquake
DCC	Digital Control Computer
DGSS	Drained Guaranteed Shutdown State
DSC	Dry Storage Container
DWMF	Darlington Waste Management Facility

Acronym	Definition
ECI	Emergency Coolant Injection
ECIS	Emergency Coolant Injection System
ECR	Emergency Coolant Recovery
EFADS	Emergency Filtered Air Discharge System
EME	Emergency Mitigating Equipment
EPG	Emergency Power Generator
EPRI	Electric Power Research Institute
EPS	Emergency Power System
ESC	End Shield Cooling
ESW	Emergency Service Water
ET	Event Tree
FAI	Fauske and Associates
FDC	Fuel Damage Category
FFAA	Fuelling Facilities Auxiliary Area
FHA	Fire Hazard Assessment
FIF	Fire Ignition Frequency
FIS	Fixed Ignition Source
FSSA	Fire Safe Shutdown Analysis
FT	Fault Tree
FW	Feedwater
GSS	Guaranteed Shutdown State
HEP	Human Error Probability
HES	Hazard Exposure Scenario
HRA	Human Reliability Analysis
HT	Heat Transport
HTS	Heat Transport System
HX	Heat Exchanger
HVAC	Heat, Ventilation and Air Conditioning
IAEA	International Atomic Energy Association
IA	Instrument Air
IC	Inside Containment
IE	Initiating Event
IFB	Irradiated Fuel Bay
ISRV	Instrumented Steam Relief Valve

Acronym	Definition
IST	Industry Standard Toolset
IUFT	Inter-Unit Feedwater Tie
LLDS	Low Level Drained State
LOCA	Loss-Of-Coolant Accident
LPECI	Low Pressure Emergency Coolant Injection
LPSW	Low Pressure Service Water
LRF	Large Release Frequency
MAAP	Modular Accident Analysis Program
MCR	Main Control Room
MD-RBGSS	Moderator Drained Rod-Based Guaranteed Shutdown State
MSO	Multiple Spurious Operation
MV	Motorized Valve
MW	Megawatt
NC	Natural Circulation
NGS	Nuclear Generating Station
NPC	Negative Pressure Containment
NRC	Nuclear Regulatory Commission (U.S.)
NUREG	Nuclear Regulation
OC	Outside Containment
OPG	Ontario Power Generation
OPGSS	Over Poisoned Guaranteed Shutdown State
OSR	Operational Safety Requirements
PAU	Physical Analysis Unit
PAWCS	Post-Accident Water Cooling System
PDS	Plant Damage State
PHT	Primary Heat Transport
PK	Programmable Controller
PMF	Probable Maximum Flood
POS	Plant Operational State
PSA	Probabilistic Safety Assessment
PSF	Performance Shaping Factor
PSVS	Powerhouse Steam Venting System
PULSW	Powerhouse Upper Level Service Water
PUPS	Portable Uninterruptable Power Supply

Acronym	Definition
PWR	Pressurized Water Reactor
RBGSS	Rod-Based Guaranteed Shutdown State
RC	Release Category
RCW	Recirculating Cooling Water
RLC	Review Level Condition
RRS	Reactor Regulating System
SAMG	Severe Accident Management Guideline
SBH	Steam Balance Header
SCD	Severe Core Damage
SCDF	Severe Core Damage Frequency
SDC	Shutdown Cooling
SDS	Shutdown System
SDV	Screening Distance Value
SEL	Seismic Equipment List
SGECS	Steam Generator Emergency Cooling System
SIIFF	Seismically Induced Internal Fires and Floods
SIO	Safety Improvement Opportunity
SMA	Seismic Margin Assessment
SPSA	Seismic Probabilistic Safety Assessment
SSC	Systems Structures and Components
SSLB	Secondary Side Line Break
THERP	Technique for Human Error Rate Prediction
UFDS	Used Fuel Dry Storage