LEVEL 1 INTERNAL FLOOD PSA ACCIDENT SEQUENCE QUANTIFICATION FOR POINT LEPREAU REFURBISHMENT PROJECT

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Abstract

A Probabilistic Safety Assessment (PSA) for Point Lepreau Generating Station has been completed as part of the plant Refurbishment Project. The main objective of this PSA is to provide insights into plant safety design and performance, including the identification of dominant risk contributors, and compare options for reducing risk. The scope of this assessment covers Level 1 and 2 PSA and includes internal events during full power and shutdown, internal fires and internal floods as well as a PSA-based seismic events margin assessment for full power operation. This paper describes the major steps in the Level 1 analysis for internal flood events and discusses the key results of this study.

1. Introduction

The Point Lepreau Refurbishment (PLR) Probabilistic Safety Assessment (PSA) is the first Level 2 PSA for both internal and external (fire, flood, and seismic) events performed in Canada. Similar recent CANDU[1] 6 PSAs include a Level 2 PSA for internal and external events for Wolsong 2, 3, and 4 and a Level 1 PSA for internal events for Qinshan CANDU 1 and 2. Following the lead of Point Lepreau PSA, Gentilly-2 (G-2) has also begun the preliminary work for a Level 1 PSA for internal and external events.

Both Atomic Energy of Canada Limited (AECL) and New Brunswick (NB) Power PSA analysts participated in the PSA. For most systems, NB Power provided reliability models and AECL incorporated common cause failures (CCFs) and human reliability analysis. All PSA methodologies and analysis reports were reviewed by NB Power. Component failure and operation data was provided by NB Power for input to the analysis models.

The Level 1 and Level 2 PSA was performed in order to evaluate the summed severe core damage frequency (SCDF) and the summed large release frequency (LRF) for the refurbished Point Lepreau plant. Following accident sequence quantification (ASQ) for internal events, fire and flood, the results were integrated to provide an overall estimation of the SCDF and LRF. The combined result was within the proposed risk limit of 1E-04/yr and 1E-05/yr for SCDF and LRF respectively (Reference [1]), which is in line with the international results for the refurbished plants.

[1] CANDU is a registered trade-mark of Atomic Energy of Canada Limited (AECL).

In the Level 1 flood PSA, the flood-induced SCDF is estimated for full power operation only and includes an analysis of plant design and operation following internal floods with emphasis on the accident sequences that lead to core damage, their sources and their frequencies. The frequency or mode of containment failure and the consequences of radionuclide releases are a part of the Level 2 analysis and are not presented in this paper.

The internal flood PSA considered potential floods in the Reactor Building (RB), Turbine Building (TB), Service Building (SB), Secondary Control Building, Emergency Core Cooling (ECC) building, Condenser Cooling Water (CCW) building, and Freshwater Pumphouse (FWPH). Events such as a Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), Loss of Service Water, Loss of End Shield Cooling, Loss of Moderator Cooling, and Loss of Feedwater are not considered in the flooding analysis as they are initiating events considered in the full power internal events analysis. Pipe breaks, spray or splashing, including fire sprinkler spray effects and firewater pipe breaks, that cause damage to other systems or components are considered in the flood analysis. Seismic-induced flooding was considered in the seismic-based PSA and found to be not credible. It is assumed that the reactor is tripped and the plant is shutdown in the event of a flood. The plausible degradation of mitigation functions and consequential impact in the unlikely event that the plant remains at power in the event of a flood is not considered.

While the frequency of floods is significantly lower than that of full power internal events, floods impact plant risk by acting as common cause initiators. The Level 1 flood PSA was performed in two stages: flood analysis and ASQ. In the first stage, flood analysis, a plant walkdown was performed to identify significant flooding sources as well as any key safety equipment that may be submerged or sprayed within its proximity. Subsequently, qualitative screening was performed to eliminate flood scenarios from further analysis based on the location of safety related systems and equipment. For the flood scenarios that were not screened out qualitatively, the flood frequency was estimated. Quantitative screening was performed to eliminate flood scenarios from further analysis by conservatively evaluating the SCDF of the bounding flood scenario. The flood scenarios that are not screened out quantitatively proceeded to ASQ.

For each flood scenario, the flood analysis provided the ASQ with the flood scenario frequency, mitigation and support systems that are impaired by the flood, and the time available for operator actions to stop or isolate the source of flood. In ASQ, an event tree representing the plant response following the flood was developed and quantified in order to estimate the SCDF for each flood scenario.

2. Flood Analysis

The list of components credited in the internal event models for full power was the basis of the analysis of the fire, flood and seismic events. The component room locations were identified and included in the walkdown areas for fire, flood and seismic events. Three walkdowns were performed for the fire, flood and seismic events.

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The flood and fire walkdowns were performed in all the areas that have PSA credited components and related cables. Since the complete list of the basic events considered in the internal events PSA was not available when the walkdown was performed, rooms were selected for walkdown if they contain components associated with systems considered in the fire hazard analysis; cable trays associated with the components included in the fire hazard analysis; or contain components or cables related to systems considered for fault tree analysis.

As a result of the flood walkdown, the following information was established: definition of the flood zones; flood sources within each flood zone; PSA credited devices within each flood zone; floodable volumes; flooding barriers, sump drains and sump pumps; and the effect of spray. Using this information, qualitative and quantitative screening was performed. In the qualitative screening, flood areas are screened out if they do not contain equipment susceptible to damage and necessary for safe shutdown or lead to an internal event. Also, flooding sources that do not have enough capacity to damage safe shutdown equipment or to lead to an internal event are screened out in this stage of the analysis.

For flooding areas that have not been screened out from qualitative screening, the flood frequencies were evaluated, the flood-induced initiating events were identified, and an initial quantification of flood-induced accident sequences was performed. Based on this quantification of severe core damage frequencies, the flood scenarios were either quantitatively screened out from further analysis or selected for detailed analysis. For the flood scenarios proceeding to detailed analysis, the frequency of the initiating event has been calculated based on operating experience. Table 1 provides a summary of the analysis performed for flood events.

Table 1 – Summary of Flood Zones Analysed

Area	Total Flood Zones Identified	Flood Zones Screened Out in Qualitative Screening	Flood Zones Screened Out in Quantitative Screening	Flood Zones Requiring Detailed Analysis
Reactor Building	15	14	1	0
Service Building	38	32	6	0
Turbine Building	15	9	5	1
Secondary Control Building	2	2	0	0
ECC Building	1	1	0	0
CCW Building	3	2	1	0
Freshwater Pumphouse	1	0	1	0

3. Turbine Building General Flooding Zone

As a result of the qualitative and quantitative screening analyses, all plant areas except the TB general flooding zone were screened out. The TB general flooding zone includes six floor elevations: -16' (basement), -5', 10', 22', 35', and 45' or higher. Since these areas are connected to one another vertically, any flooding originating in this zone will eventually flow to the basement. There are two significant flooding sources of concern in the TB general flooding zone: CCW and Raw Service Water (RSW), which supply the plant with sea water from the Bay of Fundy. A CCW or RSW pipe break can result in significant flooding because of the high tidal variation in the Bay of Fundy. When the CCW or RSW line break occurs, sea water would flow by gravity until the water level in the TB reaches the tidal level. In order to prevent catastrophic flooding, the TB is equipped with a flooding protection system which detects high flood levels in the TB, sends an alarm to the Main Control Room and commands the automatic closure of the condenser discharge valves, the CCW pump discharge valves, and trips the CCW pumps.

Flood events in the TB general flooding zone include breaks in the 60" CCW piping, 24" RSW piping, and 12" RSW piping. In the analysis, the flood events were defined by the location of the break. For CCW flood events, inlet breaks are those which occur at the inlet side of the condenser; outlet breaks are those which occur at the discharge side of the condenser; and non-isolable breaks are those which occur after the condenser discharge isolation valve including the isolation valves. For RSW flood events, non-isolable breaks are those which occur at or downstream of the temperature control valves. Since flooding is caused by sources such as piping breaks, valve ruptures, expansion joint ruptures, and tank ruptures, all such equipment was included in the calculation of the flood event frequencies for each of the seven break locations (three for CCW and two for each RSW piping). Thus, the frequency of each flood event, provided in Table 2, was a tally of international failure rate data for piping, valve, expansion joint and tank rupture frequencies.

Table 2 – Turbine Building General Flood Zone Event Frequencies

Flood Event	Frequency (/year)
CCW 60" Inlet Pipe Break	1.66E-03
CCW 60" Outlet Pipe Break	2.49E-03
CCW 60" Non-isolable Pipe Break	5.39E-04
RSW 24" Isolable Pipe Break	4.14E-04
RSW 24" Non-isolable Pipe Break	6.84E-05
RSW 12" Isolable Pipe Break	2.64E-03
RSW 12" Non-Isolable Pipe Break	2.72E-05

For each pipe break, the factors affecting the flooding flow rate include the mechanism causing flow and the size of break. For CCW pipe breaks, the flooding may be caused by the CCW pumps or the tide from the Bay of Fundy. The flooding flow rate due to the tide is dependent on the level of tide in the Bay of Fundy, which continuously varies between –16' to 16' elevation. The tide changes twice per day and assuming that the tide changes with a constant rate, it takes six hours to change from the lowest elevation to the highest elevation. Therefore, for CCW flood events the flooding flow rate from high tide occurs half of the time. Similarly, for RSW pipe breaks, the flooding may be caused by the RSW pumps or the tide. However, since the RSW piping is connected at –5' elevation to the CCW concrete duct, the flooding flow rate from high tide occurs when the tide level is between –16' to –5' elevation, which is two-thirds of the time. Each pipe break can be classified as small, medium, or large. The flooding flow rate is calculated for each size of pipe break. As such, 21 flood scenarios were established based on the seven break locations and three break sizes.

The time from the initiation of the flooding to damage of PSA-credited equipment is dependent on the flooding flow rate and the floodable volume. The floodable volume is determined by the floor area, vacancy factor, and critical height. For each elevation in the turbine building general flooding zone, the PSA-credited equipment was identified.

4. Accident Sequence Quantification for Turbine Building Flood Events

For each of the turbine building flood scenarios, an event tree representing the plant response following the flood event was developed. These event trees consist of the flood initiating event, flood mitigating systems and operator actions, and internal events mitigation systems and operator actions. While the termination points of the event tree are classified as plant damage states, the nodes between the flood mitigation systems and the internal events mitigation systems are flood damage states. The flood mitigating systems and operator actions stop the escalation of the flood by detecting the flood, isolating the break through closure of valves, or stopping the pumps. Human reliability analysis was performed for the operator actions to mitigate the flood using the ASEP methodology (Reference [2]). The time available for the operator action is dependent on the flooding flow rate and the critical flood volume associated with the flood damage state that occurs if the operator action was not performed successfully.

The accident sequence following the flood damage state is dictated by which systems remain available and are not impaired by the flood. This accident sequence logic is similar to that of the Level 1 full power internal events. For floods due to a break in the CCW pipes, the accident sequence logic is similar to the event tree for a loss of condenser vacuum with modifications for impaired equipment depending on the flood damage state. For floods due to a break in the RSW pipes, the accident sequence logic is similar to the event tree for a loss of service water with modifications for impaired equipment depending on the flood damage state. The accident sequence termination is classified as plant damage states which are provided in Table 3.

Table 3 – Plant Damage States

Plant Damage State	Definition	Type of Accident
PDS0	Early (Rapid) Loss of Core Structural Integrity	Severe Core Damage
PDS1	Late Loss of Core Structural Integrity with High PHT Pressure	Severe Core Damage
PDS2	Late Loss of Core Structural Integrity with Low PHT Pressure	Severe Core Damage
PDS3	LOCA + LOECC with Moderator Required within Fifteen (15) Minutes	Widespread Fuel Damage
PDS4	LOCA + LOECC with Moderator Required after Fifteen (15) Minutes	Widespread Fuel Damage
PDS5	Large LOCA with Early Flow Stagnation	Limited Fuel Damage
PDS6	Single Channel LOCA with Containment Overpressure	Limited Fuel Damage
PDS7	Single Channel LOCA with No Containment Overpressure (In-Core LOCA)	Limited Fuel Damage
PDS8	Loss of Cooling to Fuelling Machine	Limited Fuel Damage
PDS9	LOCA with No Significant Fuel Failures	Limited Fuel Damage
PDS10	Deuterium Deflagration (D2 > 4%) in Cover Gas and/or Release of Moderator into Containment (Fuel Cooling is Maintained)	No Significant Fuel Damage

The fault tree used for flood ASQ is the multi-top master fault tree from Level 1 internal events including additional fault tree tops for the flood mitigating systems. This fault tree is the only one used during the ASQ phase and was associated with every flood event tree during quantification. The computer code CAFTA 5.2 (Reference [3]) is used as the fault tree, event tree, and cutset editor. PRAQuant 4.0a (Reference [4]) is used as the interface to evaluate the accident sequences while FTREX 1.2 (Reference [5]) was used as the quantification engine. Mutually exclusive events were removed from the resulting cutsets, followed by the application of recovery factors using Qrecover. Recovery is used to replace the dominant CCFs calculated using Unified Partial Method (Reference [6]) with CCFs calculated using the alpha method and the alpha factor database (Reference [7]). Recovery is also used to replace the dominant human error probabilities that were evaluated using the ASEP methodology with human error probabilities using the Technique for Human Error Rate Prediction methodology (Reference [8]). The basic event data used for the flood ASQ is based on the PLGS site-specific reliability database of safety-related systems.

The total flood-induced SCDF is estimated to be 1.15E-06 events/year and a breakdown of the different flood events are shown in Figure 1. The most dominant flooding source was isolable breaks in RSW 12" pipes, which accounts for 81.5% of the flood-induced SCDF. While the flood-induced SCDF did not significantly contribute to the total risk associated for the plant in comparison with full power internal events and internal fire events, it was suggested that some general instructions be included in the associated operating manuals for CCW and RCW which at this time, do not have specific operating instructions in the event that a flood is not automatically isolated due to a failure of the TB flooding logic. In addition, the need to qualify the ECC low pressure suction valves was identified since they are susceptible to submergence in the event of significant flooding. As such, one of the recommendations that came out of the Level 1 internal flood PSA includes the environmental qualification of ECC low pressure suction valves for submergence.

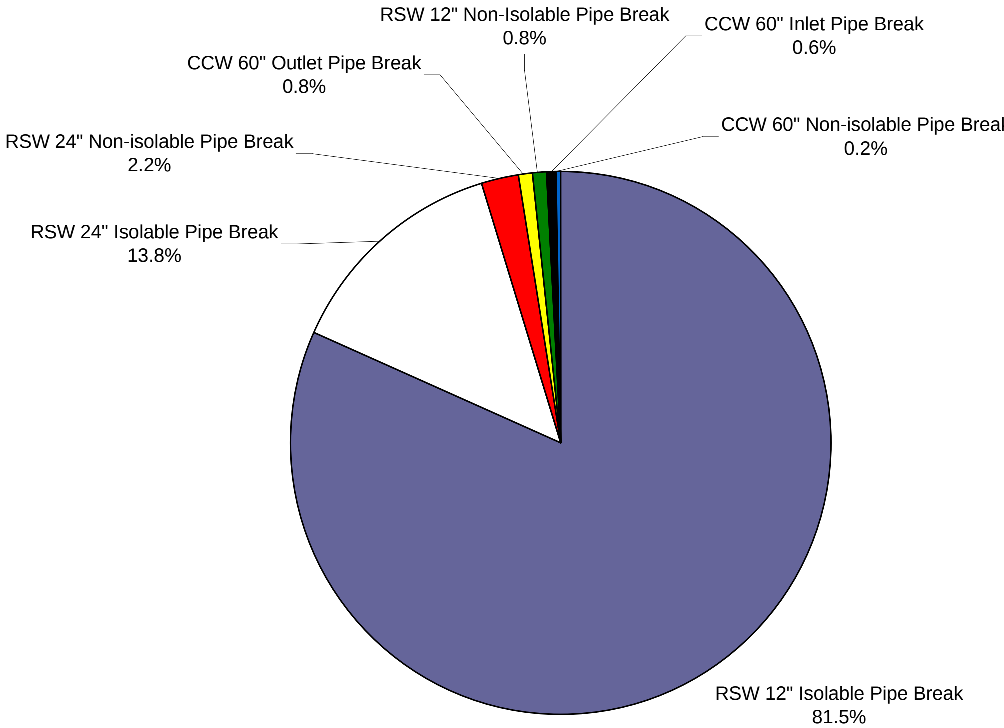

Figure 1 – Breakdown of Flood-induced SCDF

5. Conclusion

As a part of the refurbishment, a Level 1 and Level 2 PSA including internal and external was performed for Point Lepreau Generating Station. This paper presented the main steps of the Level 1 Internal Flood PSA which includes qualitative screening, quantitative screening, and accident sequence quantification. The estimated flood-induced SCDF estimated from this analysis does not dominate the total SCDF for the plant. Two plant improvements related to environmental qualification and operating manuals were recommended.

6. References

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[8] Swain, A.D., and H.E. Guttman, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications", NUREG/CR-1278, Rev. 1, August 1983.