PROGRESS ON SEISMIC MARGIN ASSESSMENT AND FIRE PSA FOR LEPREAU REFURBISHMENT

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ABSTRACT

On July 29, 2005 it was announced that the Province of New Brunswick would proceed with the $1.4 Billion project to refurbish the Point Lepreau Generating Station with Atomic Energy of Canada Limited (AECL) as the general contractor. The major activity during the outage would be the replacement of all 380 Fuel Channels, Calandria Tubes and connecting feeder pipes. This activity is referred to as ***Retube***. NB Power Nuclear would also take advantage of this outage to conduct a number of additional repairs, replacements, inspections and upgrades. These collective activities are referred to as ***Refurbishment***. This would allow the station to operate for an additional 25 to 30 years.

As part of Refurbishment, NB Power is performing a Level II PSA (Probabilistic Safety Assessment) to complement the current safety analysis, which supports the existing Operating License for the station. The events considered in the PSA include internal fires and seismic events. The approach adopted for fire events is the conventional fire PSA [1] and that for the seismic events analyses is a PSA-based Seismic Margin Assessment (SMA). The PSA is scheduled to be completed at the end of 2007.

This paper discusses the progress on the fire PSA and the PSA-based SMA.

Keywords: Pt. Lepreau Refurbishment Project, NB Power Nuclear (NBPNC), Level II PSA, Fire PSA, PSA-based Seismic Margin Assessment (SMA), Seismic Safety Equipment List (SSEL), Seismic Review Team (SRT)

1 INTRODUCTION

The main objective of this Level II PSA is to provide insights into plant safety design and performance, including the identification of dominant risk contributors and a comparison of options for reducing risk. The scope of work includes internal events, internal fires, internal floods, as well as external events involving seismic for the plant operating state. A shutdown state PSA for the internal events is also being produced.

A significant number of upfront activities have already been completed. These activities include completion of all PSA methodology documents, identification of internal initiating events, event tree analysis, system dependency matrices, 50% of the fault tree analyses, and plant walkdowns in support of PSA-based Seismic Margin Assessment (SMA), fire and flood PSA.

Regulatory acceptance of key PSA methodology documents has already been received. Performance of the PSA activities is a team effort involving both the Atomic Energy of Canada Ltd (AECL) and NB Power Nuclear (NBPNC) PSA analysts. For example, for a number of systems, NBPNC prepared the initial fault tree analysis models, which AECL reviewed and revised to incorporate common cause failures. Furthermore, all PSA methodology documents prepared by AECL were reviewed by NBPNC and operational data for input to the fault tree and event tree analysis was provided by NBPNC. Plant walkdowns in support of the fire and flood PSA, and PSA-based seismic margin assessment were a joint exercise between the two groups.

This paper discusses the progress on the PSA-based SMA and fire PSA that are being performed in support of Point Lepreau Refurbishment.

2 OVERVIEW OF APPROACH

2.1 Fire PSA

The major fire PSA tasks currently in progress and/or planned are summarized below.

- Develop a fire PSA database of information to be used in the PSA.
- Perform plant walkdowns to confirm the information contained in the fire PSA database and collect additional information required for fire PSA.
- Partition the plant area into fire compartments.
- Develop a Point Lepreau specific fire frequency ignition database.
- Perform cable route analysis to trace and establish electrical/control cable routes for the devices in the PSA-credited systems.
- Perform qualitative and quantitative screening analysis to screen out the areas that are not safety significant in view of fire.
- Perform fire growth and progression modelling using the computer code COMPBRN IIIe [2] and/or CFAST [3] or manual calculations.
- Develop fire scenario event trees and fault trees for fire protection systems.
- Perform Accident Sequence Quantification in order to obtain the severe core damage frequency from fire.
- Perform sensitivity analysis for key assumptions.

2.2 PSA-based SMA

A PSA-based Seismic Margin Assessment is adopted for Point Lepreau seismic events analyses. The US NRC recommended this approach in SECY 93-87 [4], to avoid the problems encountered with seismic PSAs where the results of severe core damage frequency (SCDF) may be dominated by the uncertainties in the hazard curve. The PSA-based SMA essentially performs all the steps of a seismic PSA except convolution of fragilities with the hazard input. It thus provides all the design insights expected of a seismic PSA without making the results vulnerable to the large uncertainties typically encountered in site hazard input.

The major PSA-based SMA tasks in progress and/or planned are summarized below.

- Establish the Safety Objective in terms of plant High Confidence Low Probability of Failure (HCLPF) capacity.
- Collect and review seismic design documentation, design criteria, seismic analysis reports, flow sheets and general arrangement drawings.
- Identify structures/components for seismic capacity analysis – Seismic Safety Equipment List (SSEL).
- Perform plant walkdown.
- Perform seismic capacity analysis for the structures/components identified from the plant walkdown.
- Develop seismic initiating hierarchy event trees, which determine seismic-induced initiating events, and secondary seismic event trees, which delineate the plant responses on the seismic-induced initiating events.
- Develop seismic fault trees by modifying internal event fault trees.
- Generate minimal cutsets by quantifying the seismic event trees.
- Calculate HCLPF value and random failure probabilities for each minimal cutset.
- Derive the HCLPF value for each seismic core damage sequences in seismic event trees.
- Derive plant HCLPF value.

3 PROGRESS ON FIRE PSA

The first step of the fire PSA was to collect fire hazard information regarding combustible materials, ignition sources, fire barriers, fire protection measures, etc. A large part of such information was available from the station Fire Hazard Assessment (FHA). However, from the PSA viewpoint, the information in the FHA was not complete since the main purpose of the FHA was to assess the potential for fires that impact both Group 1 and Group 2 safety significant systems. The fire PSA requires additional information about PSA-credited structures, systems, and components (SSCs). A plant walkdown was conducted to collect additional information required for fire PSA. The main purpose of the fire walkdown was to confirm the information already collected from documents/drawings and to collect additional information that could not be easily obtained from documents/drawings.

The areas covered in the fire walkdown include all those areas in which PSA-credited components and their corresponding cables are located. Determining these areas required identification of all the equipment considered in the analysis and tracing of the cables related to the equipment. However, the complete internal events PSA models were not available at the time of the walkdown and thus the equipment list considered in the internal events PSA was not available. Given this limitation, the walkdown areas were selected as follows:

- The system Basic Subject Index (BSI) numbers listed in the FHA were reviewed.
- The rooms that contain any cable trays or components included in the BSI numbers considered in the FHA were identified.

- Based on the list of systems considered for fault tree analysis, the BSIs not considered in the FHA were identified and added. The list of BSIs for fault tree analysis was based on the system dependency matrix.
- Additional rooms that contain the equipment or cables related to the additional BSIs were reviewed.

All the rooms determined from the above process were identified. Once the area of the walkdown was determined, as much information as possible was obtained from documents and drawings. Then the walkdown datasheets were prepared noting the existing information and the additional information required to be collected during the walkdown. During the walkdown, the information was confirmed or added using the walkdown datasheets. Typically for each room, the following data was recorded in separate tables.

- Physical Room Data: elevation, dimensions, floor area, room boundaries, etc.
- Ignition sources in the room.
- Combustible materials including the transient materials.
- Nuclear Safety System Devices: all the PSA-credited devices located in the room excluding the cables.
- Fire Protection Data: manual and automatic fire detection/suppression systems installed in the room.
- Fire Barrier Analysis: information on fire barriers in the room.
- Adjacent Space Evaluation: results of fire resistance rating evaluation for the barriers.
- Openings in each room.
- Rooms, Trays and Cables: tray number in the room, and cable number in each cable tray.

It was observed that the plant has good house keeping, adequate procedures for handling the transient combustibles and hot works, and strict safety practices. During the walkdown it was observed that all potential significant combustibles such as oil were kept in a designated place and marked appropriately. In view of fire suppression, the plant mainly relies on manual fire suppression for almost all areas except those containing large amounts of oil and cable access areas. The fire hose stations and the portable fire extinguishers are well maintained. The fire brigade is well organized and the training of the fire brigade is performed regularly. Security staff and fire watch for hot work (maintenance personnel) are also regularly trained. Certain areas in the plant are completely enclosed by fire barriers. In all the other areas, potential for fire propagation will be dependent on the presence of combustibles and most of the fires if ignited would be fuel-controlled fires. Some concerns of Seismic/Fire interaction were also identified during the walkdown. Cable trays, which cannot be easily accessible if ignited, and vertical cable trays, in which fire propagates rapidly if ignited, were also noted.

The information and the insights collected during the walk down will be used as inputs for the subsequent fire PSA work (screening analysis, fire propagation analysis, fire scenario development, etc).

The fire ignition frequency database has been developed, combining the generic CANDU fire ignition frequency [5] and the Point Lepreau specific fire experience using the Bayesian technique. The Point Lepreau fire ignition frequency database developed from the analysis is presented in the Table 1.

The cable routing analysis was conducted to identify the location of cables related to the PSA-credited cables. Since internal event fault trees are not developed for all systems considered, the devices selected for cable routing are determined based on the systems, rather than the individual device function. The steps taken for the cable routing analysis are as follows:

- List all the system BSI numbers credited in the internal events PSA.
- Add system BSIs that are required for fire analysis such as HVAC system, emergency lighting system, etc.
- Determine all the devices in the listed BSIs.
- Identify cable numbers/route numbers for the devices using the Integrated Electrical and Control (IntEC) database.
- Identify cable trays that the cable/route is passing through using the IntEC database.
- Identify room numbers using the cable/conduit location information in the IntEC and FHA databases.
- Present all the rooms that the cables are passing through from device to device.

The number of devices selected for cable routing from the above steps are about 38,000 and the routes required for tracing are about 57,000. Since it is not practicable to trace all those routes manually, a computer program was developed to extract cable information from IntEC and the existing FHA database. The analysis was completed using this program.

The results of the computer run showed that about 13,000 routes or 23% of the total routes were “Incomplete”, as the route ended at a non-functional device such as a distribution frame, junction boxes, and panels. Also, it showed that there are about 800 cable trays/conduits for which the locations could not be determined from the IntEC and FHA databases.

The partitioning of the plant areas into fire compartments is currently in progress. The existing fire zones at Point Lepreau were based on fire protection/fire suppression, and not on the fire barriers or potential fire propagation. In the fire PSA analysis a fire compartment is defined as a space enclosed by, not necessarily fire barriers, but by non-combustible barriers or spaces that could limit fire propagation to other fire compartments. The partitioning includes arranging all available information per compartment so that the fire analysis can be performed based on the fire compartment.

When the cable routing analysis and partitioning of the plant areas are completed, the screening analysis for each fire compartment will be started.

4 PROGRESS ON PSA-BASED SMA

The first step of the PSA-based SMA was to determine safety objective for the seismic event in view of HCLPF capacity. The Design Basis Earthquake (DBE) of Point Lepreau is 0.2g and the safety objective was established as 0.3g HCLPF capacity.

The 0.3g has less than 1.0E-4/yr mean return frequency. The safety objective proposed corresponds to 1.0E-6/yr frequency for severe core damage when the HCLPF capacity is convolved with the Point Lepreau seismic hazard information.

Considering this safety objective, two screening levels were established:

- 0.5g HCLPF capacity for seismically qualified structures and components and
- 0.3g for non-seismically qualified (NSQ) structures and components.

The impact of the screened out components and structures on the plant safety would be negligible.

The SSCs (Structures, Systems, and Components) requiring seismic capacity analysis were then determined. As in the case of the fire PSA, the SSCs were mainly determined based on the system BSIs rather than the components, since internal events PSA models were not completed at the time. The systems and components requiring Class IV power were excluded at the initial stage considering that Class IV power has generally the lowest seismic capacity. The total number of components potentially requiring seismic capacity evaluation were determined to be about 1,400.

For the SSCs identified for the seismic capacity evaluation, a plant seismic walkdown was performed. The purpose of the seismic walkdown was to:

- Screen from the Safe Shutdown Equipment List (SSEL) those equipment items that have high HCLPF capacity at or above the screening levels specified (0.3g for NSQ and 0.5g for DBE qualified components)
- Identify equipment or structures that are not included in the SSEL but the structural failure of which may impact the nearby SSEL items (i.e., seismic interaction concerns).
- Define failure modes (e.g., functionality, structural integrity, or anchorage failure) of the SSEL items that are not screened and the type of further evaluation required.
- Address issues of seismic induced fire, seismic induced flooding, and actuation of fire suppression systems. The SSEL includes equipment items for seismic/fire interaction and seismic/flooding interaction.

The seismic walkdown was conducted following the guidelines in EPRI NP-6041 [6], and screened out the equipment according to the criteria presented in the document. For screened-in equipment, recommendations were provided so that HCLPF capacity based on the as-installed conditions, equipment qualification documents, and engineering drawings can be evaluated.

Seismic Evaluation Walkdown Sheets (SEWS) were prepared for each of the components in the SSEL. For the SSEL lead items, the SEWS developed for the Generic Implementation Procedure under the USI A-46 program [7] was used instead of those in the Reference [6] since it provides a detailed checklist for components. For the other SSEL items, only the major questions like capacity/demand, anchorage, and interactions were addressed following the EPRI guidance.

Screening of the SSEL items was performed after the seismic walkdown. Some calculations for anchorage and masonry walls were performed to support the screening. The screening of components followed the guidance in EPRI NP-6041, which provides the screening tables to decide the generic seismic capacity for the SSEL items. The horizontal ground response spectrum used for screening purposes is defined by NUREG/CR-0098 [8] as 5% damped median ground response spectrum for rock site anchored to 0.3g peak ground acceleration (pga). The 5% damped median peak spectral acceleration is 0.64g. If the ground response spectrum is anchored to 0.5g pga, the peak spectral acceleration is 1.06g.

Three ranges of seismic capacities of nuclear power plant structures and equipment in terms of 5% damped peak spectral ground acceleration (S_a) are provided in the EPRI tables: < 0.8g, 0.8g to 1.2g, and > 1.2g. In terms of peak ground acceleration (pga), these ranges correspond to 0.3g, 0.5g, and > 0.5g based on the median spectral accelerations of 0.64g and 1.06g discussed above.

The major findings from the seismic walkdown and subsequent screening calculations are as follows:

- The NSQ electrical equipment located in Rooms T1-301, 601, and 701 of the turbine building and the control panels in Rooms S1-327 and 328 of the service building could not be screened due to the following reasons:
  - The median floor response spectra exceed the generic seismic capacity of the equipment.
  - Inadequate or marginal anchorage.
  - Concern of seismic interaction from the nearby tall masonry walls.
  - Potential seismic-induced pounding effects to the control equipment in Rooms S1-327 and 328.

  Further evaluations (e.g., better estimate of frequency and/or modifications of narrow peak of the floor response spectra) are recommended for calculating fragility of these items as well as those masonry walls that were not screened.
- The majority of the motor-operated valves and pneumatic valves in the SSEL were screened at 0.3g for NSQ systems and 0.5g for seismically qualified (SQ) systems. The actuators' height and weight are typically within the limits determined from the earthquake experience database. In the case of a valve with long actuator extended to the next higher floor elevation, the shaft was observed braced laterally at mid-height. Many valves inside the reactor building are located at a high elevation in the internal structure (i.e., elevations 86 feet and higher) or in the containment structure (i.e., R1-601). At this high elevation, the 5% damped peak spectral accelerations of the 0.3g SME (Seismic Margin Earthquake) floor response spectra exceed the 2g spectral acceleration capacity given in the EPRI NP-6041 screening table. As such these valves could not be screened. The seismic qualification documents of these valves will need to be reviewed to verify if the valves should be screened. Alternatively, a generic evaluation can be performed to quantitatively identify conservatism in the valve seismic qualification process.
- Many masonry block walls were observed throughout the safety-related buildings. Those walls whose failure may impact nearby SSEL items were identified. Both DBE qualified and non-DBE qualified walls were confirmed to have reinforcing steel, both vertical and horizontal, from the review of drawings. This reflects good seismic design practice at the station. Bounding evaluations of selected block walls have confirmed that DBE-qualified walls could be screened at 0.5g and that the NSQ masonry walls above elevation 75 feet of the turbine building and the tall masonry walls in the standby diesel generator rooms cannot be screened. This bounding evaluation for the selected block walls resolved masonry wall seismic interaction concerns for some of the SSEL items. Seismic fragility evaluations are required for the non-screened masonry walls considering realistic

behaviour of the walls (i.e., nonlinear frequency estimate and realistic lateral displacement limit of the walls).

- The shell and tube heat exchangers on the SSEL were well designed and anchored as observed in the plant and could be screened at 0.3g pga or higher. NSQ vessels and heat exchangers that are located high in the internal structure require further evaluations to verify screening because of the high floor spectral accelerations. A detailed evaluation may show that the vessel's fundamental frequency is far apart from the peak spectral acceleration frequency of the floor response spectra where the vessel is supported. The NSQ deaerator and its storage tank and the reserve feedwater tank are supported high in the turbine building where high design floor spectral accelerations are observed. Further evaluation of the tank support saddles, anchorage, and the steel framing (both horizontal and vertical) of the turbine building are required.
- The Emergency Power Supply (EPS) diesel generators were seismically qualified by testing and were screened at 0.3g based on review of the qualification document. Additional evaluations are required to screen the EPS diesel generator at 0.5g. The NSQ standby diesel generators could have been screened at 0.3g if it was not for the concerns for seismic-induced failure of the nearby tall masonry walls. The weak links of both the EPS and the standby diesel generators appeared to be the peripheral components such as the inadequately anchored fuel oil day tanks.
- Based on the walkdown conducted on the main steam lines, ECC piping, and the RCW/RSW piping, it was concluded that the seismically qualified piping can generally be screened at 0.5g pga. The main steam lines in the turbine building were not seismically qualified and were screened at 0.3g pga. The Fibreglass Reinforced Piping (FRP) portions of the RCW and RSW lines were not screened and require further evaluations of the FRP piping seismic capacity or the consequence of FRP piping failure. Instrumentation lines and their supports in the transmitter room and near the inlet header were reviewed and screened at 0.3g or 0.5g.
- The seismic walkdown conducted for cable trays in several rooms of the service building and turbine building covered both SQ and NSQ cables. There are noted differences in the construction of the SQ and NSQ cable trays. Some level of seismic loads was considered in the design as observed from lateral bracing of NSQ cable tray supports. The cable trays are lightly loaded with cables. The Seismic Review Teams (SRTs) concluded that the NSQ cable trays may be screened at 0.3g pga and the SQ cable trays may be screened at 0.5g pga. One exception is the 100-foot long vertical cable trays in Room S1-004. Further evaluation of seismic adequacy of the vertical space frame that supports these trays is recommended.

Out of a total of about 1400 SSEL items, about 850 were screened out at either 0.3g or 0.5g for SQ items and 0.3g for NSQ items. The screened-in SQ items (labelled as further evaluation required in the SEWS) may be screened out upon reviewing their seismic qualification documents.

The seismic capacity analysis for the components and structures following the screening is now in progress. When the seismic capacity analysis is completed, the plant modelling for the seismic events will be started.

5 SUMMARY

For the first time in Canada, fire PSA and PSA-based seismic margin assessment are being performed for an operating CANDU plant. An important aspect of this PSA project was an upfront regulatory review of the PSA methodology documents by the Canadian regulator (CNSC – Canadian Nuclear Safety Commission), thus providing an early opportunity to resolve any issues with the CNSC and obtaining agreement on the PSA development approach. Regulatory acceptance of fire PSA methodology documents has already been received. Most of the regulatory comments related to the PSA-based seismic margin assessment have also been resolved.

The PSA analysis is in progress and scheduled to be completed at the end of 2007. Problems such as incomplete cable routes, larger than expected number of screened in components for seismic capacity evaluation are being addressed through deployment of additional resources to minimize impact on the schedule.

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Table 1: Point Lepreau Specific Fire Ignition Frequency Vs. Generic CANDU Fire Ignition Frequency

Category ID	Category Name	Generic Frequency/ plant-year	Point Lepreau Frequency/ plant-year
1	Battery	1.8E-3	2.3E-3
2	Battery charger	2.7E-3	3.3E-03
3	Inverters	9.0E-04	1.4E-03
4	Main control room	4.5E-3	4.2E-3
5	Digital control computers	3.1E-3	3.3E-3
6	Diesel generator sets	2.23E-2	2.26E-2
7	HVAC equipment	4.5E-3	6.1E-3
8	Dryers	3.1E-3	3.3E-3
9	Hydrogen fires	8.9E-3	9.0E-03
10	Logic and protection cabinets	2.1E-02	2.1E-02
11	PHTS pumps	4.5E-3	4.2E-3
12	Pumps	1.34E-2	1.29E-2
13	Motor control centre	8.0E-3	8.1E-3
14	Motors	1.43E-2	1.39E-2
15	Motor generator sets	1.8E-3	2.3E-03
16	Power and control cables	1.78E-2	1.78E-2
17	Low voltage switchgear	8.0E-3	8.1E-3
18	High voltage switchgear	1.43E-2	1.39E-2
19	Gas turbine generators	N/A	N/A
20	Turbine-generator	2.5E-2	2.46E-2
21	Main unit transformer	1.25E-2	1.19E-2
22	Transformers	1.43E-2	1.49E-2
23	Human error	2.32E-2	2.26E-2
24	Cable fires caused by welding and cutting	1.8E-3	2.3E-03
25	Transient fires caused by welding and cutting	2.5E-2	2.65E-2
Total		2.57E-1	2.61E-1