REFURBISHMENT OF POINT LEPREAU GENERATING STATION

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ABSTRACT

NB Power is conducting an 18-month maintenance outage of the Point Lepreau Generating Station (PLGS) which began on March 28, 2008 (Reference-1). The major activity is the replacement of all 380 Fuel Channel & Calandria Tube Assemblies and the connecting feeder pipes. This activity is referred to as ***Retube*** (Reference 2). NB Power is also taking advantage of this outage to conduct a number of repairs, replacements, inspections & upgrades (such as rewinding or replacing the generator, replacement of shutdown system trip computers, replacement of certain valves & expansion joints, inspection of systems not normally accessible, etc). These collective activities, as well other regular maintenance and inspection activities are referred to as ***Refurbishment***. This would allow the station to operate for an additional 25 to 30 years.

Work on the project began in early 2000. The scope was determined from the outcome of a two-year study involving a detailed condition assessment of the station that examined issues relating to ageing and obsolescence (Reference-3). The majority of the plant components were found to be capable of supporting extended operation without needing replacement or changes. In addition to the condition assessment, a detailed review of Safety & Licensing issues associated with extended operation was performed. This included a review of known regulatory and safety issues, comparison of the station against current codes and standards, and comparison of the station against safety related modifications made to more recent CANDU 6 units.

Benefit cost analyses (BCA) (Reference-4) were performed to assist the utility in determining which changes were appropriate to include in the project scope. As a Probabilistic Safety Assessment (PSA) for PLGS did not exist at the time, a risk baseline for the station had to be determined (Reference-5) for use in the BCA. Extensive dialogue with the Canadian Nuclear Safety Commission staff was also undertaken during this phase. A comprehensive Licensing Framework was produced upon which the CNSC provided feedback to NB Power. This feedback was important in terms of achieving clarity of the regulatory position and thus to minimize the financial risk associated with regulatory uncertainty.

The Refurbishment outage was preceded by a detailed Engineering Project Phase that included:

- Finalizing details of the Retube process including modeling, tooling development, site facilities and training of personnel
- Performing Engineering activities related to design modifications, safety analysis and level II PSA
- Construction of new waste storage structures to house Retube Waste and other additional waste storage structures for the extended life of the station[1]
- Setup necessary construction facilities (offices, storage areas, change rooms, decontamination and maintenance areas)
- Performing detailed outage planning
- Development of detailed layup, commissioning and return to service procedures
- Procurement of equipment & components

Prior to receiving final project approval in August 2005, NB Power conducted a limited scope of activities that were important in reducing overall project financial risk. A number of these up-front activities related to safety analysis and licensing issues related to life extension. In particular, a level II PSA along with additional safety analyses were performed to complement that which supported the Operating Licence for the station.

This paper discusses the Safety & Licensing activities that were involved in defining the project scope and outlines the safety analysis related activities that were performed in support of the Refurbishment project and extended operation.

Background:

The Point Lepreau Generating Station is a 680 MW(e) CANDU- 6 reactor located in the province of New Brunswick on the Atlantic coast of Canada. It is owned and operated by NB Power which is the provincial utility[2]. PLGS was constructed between May 1975 and the summer of 1982. The initial Operating Licence was granted in July of 1982, with commercial operation beginning on February 1, 1983. The Station has proven to be an economic and environmentally sound source of electricity generation providing about a third of the power consumed in the province of New Brunswick. It has a significant positive economic impact in the southern part of the province, employing over 600 people and having an annual operating budget of over 100 Million dollars. In addition the station is an important element in achieving provincial environmental emission targets.

The original design of the station anticipated an operating period of about 30 years. This set the basis for the assumed number of transient cycles and the number of full power running days that equipment could either experience or be exposed to, and thus had to be designed to withstand[3]. This implies that PLGS would reach the end of the planed period of operation around 2013. Although pressure tubes and feeders can be replaced on an individual basis, the number of tubes requiring replacement increases significantly starting about 2008 –2010, making the economics of continued operation during this time less and less favorable. For this reason it was decided that the refurbishment outage would commence in late March of 2008.

Because of the significant cost associated with retubing the reactor, there was a need for the refurbishment outage to extend the period of operation out another 25 to 30 years. In addition, a number of new regulatory requirements have been issued since the station was granted its first Operating Licence. These issues drove the need to perform the comprehensive condition assessment and the safety & licensing reviews to determine the scope and cost of

[1] An environmental assessment for these additional structures was conducted and approved as part of the projects early start activities
[2] On Oct 1, 2004 NBP was restructured into a holding Company (NB Power Holding) and four subsidiary companies, one of which is NB Power Nuclear company.
[3] Based on the condition assessment review, it was identified that the number of actual plant transients and cycles experienced were far less than the number assumed in the original design and in most cases sufficient to allow an additional 25 to 30 years of post refurbishment operation.

refurbishment, to ensure the station could be operated over the extended period, and to understand the regulatory aspects.

Regulatory Climate

The station Operating Licence is renewed on a periodic basis. This drives an on-going process at the station related to updating the design, safety analysis and station programs. As a result, in response to emerging issues, a number of safety-related design changes have been introduced over the years. Similarly, a number of station programs and processes have likewise been introduced and/or improved. Safety Analysis is performed on an as needed basis in response to proposed design changes, internal and external operating experience (such as response to plant ageing, unanticipated plant response to a transient, unexpected research and development or analysis findings, plant events, improving the definition of safe operating envelope), and safety related issues raised by the regulator under an "Action Item" process. When new analysis is performed, it is conducted with current methods. The analysis is summarized in the station Safety Report that is reviewed and updated on a 3-year basis. Thus over time, much of the analysis in the Safety Report has been redone and updated with modern methods.

The periodic licence renewal and continuous safety improvement process used in Canada easily accommodates extending the planned period of operations (sometimes referred to as "life extension" in other jurisdictions). The central safety and licensing items performed in support of refurbishment were as follows:

- Identify additional safety related design changes through review[4] of:
  - Comparison of station against current codes & standards
  - Safety related changes introduced at newer CANDU 6 stations
  - Outstanding safety issues where the optimum solution would be a design change that is not likely to be introduced unless there was an extended station outage followed by prolonged period of station operation to allow
  - Results of Level II PSA against set of established goals and targets

- Identify additional Safety Analysis to be conducted through consideration of:
  - Events not included in the Safety Report but required for stations licenced to more recent regulatory documents
  - Conditions expected with refurbished and fresh core
  - Design changes to be introduced during refurbishment
  - Support to the level II PSA

- Assess plant condition via a comprehensive Plant Condition Assessment
- Assessment of equipment and cable qualification over the extended operating period
- Determine the changes to Operating Policies & Principles to cater to the defuelled core state
- Development of a benefit cost analysis process to assist in decision making (back fit issues)
- Conduct a level II PSA to replace the earlier Safety Design Matrix studies (Reference 6)
- Perform Environmental Assessment for additional structures to be built and operated at the on-site Solid Radioactive waste facility to support the refurbishment and extended operations (References 7 and 8)

Guiding principles related to safety & licensing were established at the onset of the project and presented early to the regulatory authority (the Canadian Nuclear Safety Commission - CNSC). These guiding principles evolved into a detailed licensing framework for the project. To achieve regulatory clarity, the regulator was asked to

[4] Safety significant issues raised during the comparison reviews then under went a benefit cost analysis to determine whether or not a change was warranted.

provide comments on the licensing framework. This then allowed for focused discussions. The central principles are provided in Reference 1 and 15.

Key steps in the regulatory interaction were as follows:

- Understanding of the importance of achieving regulatory clarity related to the project and extended operations
- Establishment of primary contacts
- Communication of guiding principles
- Outlining overview of project execution plan (scope, timeframe, interfaces etc. relating to project management, QA, design, assessments, outage activities, fuel loading, commissioning and restart)
- Discussions on safety reviews to determine design changes and analysis
- Discussion on key design and safety analysis issues such as shutdown system modifications, fuel channel design, PSA methods goals and targets, etc.

Although the station condition and plant processes and programs are subject to on-going reviews and inspections from the CNSC as well as from WANO and the insurance brokers, CNSC staff determined that a review along the lines of the IAEA periodic Safety Review should also be performed. Because a good portion of such a review had already been completed by the project through the condition assessment and the comparison of the plant to current codes & Standards, NB Power choose to perform an Integrated Safety Review (References 9 and 10). The overall process is now indoctrinated into regulatory document RD-360.

The remaining portion of the paper outlines the PSA and Safety Analysis scope related to the refurbishment and life extension:

PSA

The plan for producing the PSA's called for the production of a level II PSA for internal events and *External* events involving station fires, station floods and a PSA based seismic margin assessment. In addition a shut-down state PSA for internal events was produced.

The main objective of the PSA was to provide insights into plant design and performance, including the identification of dominant risk contributors and the comparison of options for reducing risk to verify that the Point Lepreau refurbished station will meet current internationally accepted safety goals. The limits for SCDF and LRF for internal and external events are 1E-04/yr and 1E-05/yr, respectively, with a goal that is ten time lower. These are in line with the international targets for the refurbished plants. The safety goals for the PSA based SMA are 0.3g for SCDF and 0.4g for LRF. An important aspect of this PSA project was an upfront regulatory review of the various PSA methodology documents by the regulator, CNSC. This provided an early opportunity to resolve any issues with the regulatory body in view of compliance with the new regulatory standard S-294.

The main tasks associated with the level II PSA were:

Level 2 PSA – Internal Events:
The internal events PSA starts off by systematically identifying a number of initiating events that cause a plant disturbance and may potentially lead to core damage. The initiating events are grouped into events involving similar plant response. For each group, an event tree is drawn indicating the required mitigating systems and operator actions to bring the plant to a safe stable state, or which otherwise lead to core damage. The probability of mitigating systems being unavailable or failing during the mission period is estimated by fault tree analysis. Fault trees are developed to identify combinations of individual component failures that can cause the system failures modelled in the event trees. All of this information is then synthesized to perform a quantification of the frequencies of the accident sequences shown in the event trees. Main elements of the Level I internal events PSA are noted below:

- Initiating event analysis to establish a comprehensive listing of internal initiating events for on power as well as the shutdown state.
- Develop a plant-specific dependency matrix to gain a full understanding of the dependencies, which exist between plant systems, as well as initiating events and plant systems. Two dependency matrices will be developed, one system-system matrix, and one initiator-system matrix.
- Develop accident sequence event trees (adopt intermediate size event tree / fault tree linking approach to defining accident sequences)
- Develop fault trees for the plant design.
- Incorporate Common Cause Failures (CCFs) in the fault tree analysis using the Unified Partial Method (UPM).
- Perform Human Reliability Analysis (HRA) related to pre accident as well as post accident operator actions. This analysis methodology is consistent with ASEP, which is a simplified version of the more analysis-intensive THERP method, developed by the US NRC.
- Perform Accident Sequence Quantification (ASQ) to evaluate frequency of the core damage related end states in each event tree.

From the Level 1 results, 184 sequences were grouped according to five representative severe core damage accidents. Five primary groups were identified: Containment Bypass, Station Blackout, Small Loss Of Coolant Accident, In-core LOCA, and Shutdown State LOCA.

The pressure capacities of the airlocks, containment penetrations and internal structures were assessed to identify the pressure capacity and location at which containment would leak. Given the containment pressure capacity, severe accident progression and containment performance analysis using MAAP4 CANDU v4.0.5A was performed for each of the five primary groups. The sequences within each of the five primary groups were further grouped according to similarity of plant configuration with respect to the status of mitigating systems, the impact on the containment system availability, and the containment status. The Accident Sequence Quantification was performed for all sequences for both Level 1 and Level 2 using a cut off truncation limit that is three orders of magnitude lower than the sequence frequency, in accordance with IAEA recommendation.

For both Level 1 and Level 2 ASQ, recovery was applied at the sequence level. Dominant CCFs were re-evaluated using the alpha factor method and the USNRC database. HRA was performed using Accident Sequence Evaluation Program methodology for the Level 1 PSA and a simplified ASEP methodology for the Level 2 PSA. The dependency between operator actions in the same accident sequence was evaluated for Level 1 through recovery after Accident Sequence Quantification using the Standardized Plant Analysis Risk HRA method and for Level 2 directly in the containment event trees. The dominant HRA contributors were re-evaluated using the Technique for Human Error Rate Prediction (THERP) methodology.

The basic event data used for the Internal Events PSA is based on the PLGS site-specific reliability database of safety-related systems.

Uncertainty analyses were performed on parameters such as: failure rates, component unavailabilities, initiating event frequencies, and human error probabilities. The uncertainties for each of these quantities were expressed in terms of error factor, which is defined as the ratio between the 95% confidence value and the 50% confidence value. However, whenever the PLGS failure rate is calculated as the average value, the Error Factor is taken as the ratio between the 95% confidence value and the mean.

Risk Importance measures were used to identify the risk significant components that were selected for the sensitivity analyses. Sensitivity analyses were performed to test the impact of certain changes in key input values (different maintenance practices, testing intervals, design changes, HRA, etc.) to PSA results.

Level 2 PSA – External Events and Seismic Margin Assessment:
Based on previous CANDU experience with PSA for external events, the events selected for Lepreau site are seismic, internal fires and internal floods due to failure of the large piping/expansion joints in the RSW and/or CCW systems. This list of external events to be explicitly assessed was subsequently verified through a review of common cause events. The main elements of the Level II external events PSA are noted below:

Plant walk-downs were carried out to both verify and supplement the information contained in the fire and flooding database. The walk-downs also provided a greater understanding of the failure modes of structures and equipment due to spatial interaction during a seismic event.

A fire database was developed to define “fire zones” and “fire areas”, typically based upon the location of barriers to fire propagation and develop hazard scenarios for the various fire zones. Only fires occurring at-power have been considered. Using the available information, the qualitative and quantitative screening was performed.

During the qualitative screening, fire areas were screened out if they do not contain any susceptible equipment for safe shutdown, or if they do not contain any equipment that, if damaged, would lead to an internal event. Also, fire sources that did not have enough capacity to damage the safe shutdown equipment or to lead to an internal event were screened out in this stage of the analysis. For the fire areas that were not screened out, the following tasks were performed as part of the quantitative screening:

- Compute initiating event frequency for all areas not screened out in qualitative screening;
- Assume all equipment and cables are damaged by fire in fire area/scenario;
- Determine fire impact on mitigating system models, and determine which ones cannot be credited for reasons such as environmental qualification;
- Determine which internal event PSA event tree can be used, and modify accordingly, and
- Perform ASQ for the fire scenario; calculate Conditional Core Damage Probability (CCDP) and SCDF for each fire scenario.

If the screening fire scenario SCDF was less than 1E-07 event/year, the fire scenario was screened out. For the fire scenarios screened in, the following tasks were performed as part of the detailed analysis:

- Fire Progression Modelling and Fire Consequence Evaluation using CFAST [10] and NUREG-1805 [11];
- Severity Factor evaluation;

- Automatic fire detection / fire suppression;
- Manual fire detection / fire suppression, and
- Identification of the representative initiating event from internal events.

All operator actions within the ET, credited mitigating system FTs, and recovery actions, were examined to determine if the operator action was in the vicinity of the fire. For Main Control Room or Secondary Control Area fires, if an operator action is on the same panel that the fire takes place, the operator action could not be credited. If the operator action is on a panel adjacent to a fire, a factor was applied to account for the increased stress from the fire. For field operator actions, if the operator action is within the same room that the fire takes place, the operator action could not be credited as it was assumed that there would be no personnel in the room except the fire brigade.

The Accident Sequence Quantification was performed for all sequences for both Level 1 and Level 2 using a cut off truncation limit that is three orders of magnitude lower than the sequence frequency.

From the Level 1 results, 322 sequences were grouped based on three representative severe core damage accidents resulting from fire: Station Black Out, Small LOCA and Incore LOCA. Similar to Internal Events, the sequences within each of the three primary groups were further grouped according to similar plant configuration accounting for the status of mitigating systems, the impact on the containment system availability, and the containment status. Similar to Internal Events, Level 1 and Level 2 ASQ for fire applied the UPM and alpha method CCFs, included ASEP and THERP derived HRA, and accounted for dependency between operator actions.

Flooding PSA:

A database was developed to define flood areas typically based upon the location of flood sources. A flood hazard scenario was then development for various flood zones. This was then followed by flood Progression Modeling which included calculations of probability of water spray interactions and time to submergence calculations. If the quantitative screening result was less than 10^{-6} events/year, then the flood scenario was screened out from further analysis.

For the flood scenarios selected for detailed analysis, the frequency of the initiating event was calculated based on operating experience. Also event trees were developed for all flood sequences. For flood mitigating systems, new fault trees were developed or the existing ones have been adapted as required. Similar to internal events, CCFs and HRAs were evaluated and ASQ was performed. The Accident Sequence Quantification was performed for all sequences for both Level 1 and Level 2 using a cut off truncation limit that is three orders of magnitude lower than the sequence frequency.

From the Level 1 results, 92 sequences were grouped according to three representative severe core damage accidents resulting from flooding events: Station Black Out, Small LOCA and Incore LOCA. Similar to Internal Events, the sequences within each of the three primary groups were further grouped according to similar plant configuration accounting for the status of mitigating systems, the impact on the containment system availability, and the containment status. Similar to Internal Events, Level 1 and Level 2 ASQ for flood applied the UPM and alpha method CCFs, included ASEP and THERP derived HRA, and accounted for dependency between operator actions.

Uncertainty and sensitivity analyses were performed for both fire and flood PSA together with the Level 2 internal events. The Level 2 PSA models for internal events, flood and fire were integrated in a master FT that was used for the uncertainty and sensitivity analyses.

Seismic based margin assessment:

A PSA based Seismic Margin Assessment was performed as the US NRC recommends this approach in SECY 93-87, to avoid the problems encountered with seismic PSA's where the results of severe core damage frequency (SCDF) may be dominated by the uncertainties in the hazard curve. The Seismic Margin Assessment involves essentially performing all the steps of a seismic PSA (fragility analysis, event trees & fault trees) 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. Main tasks involved in performance of the Seismic Margin Assessment include:

- Establish seismic safety target defined in terms of plant HCLPF (High Confidence of Low Probability of Failure)

- Seismic fragility evaluation for structures and equipment which affect consequences or mitigation of the seismic event

- Perform Failure Mode and Effect Analysis for Seismic Failures

- Develop Plant Models, in association with the components' fragility data, the seismic event trees were developed. The seismic event trees consist of a primary Seismic Event Tree (SET) and secondary SETs. The primary SET presents seismic events that lead to SCD or other seismic-induced initiating events and were used to determine seismic-induced initiating events. The secondary SETs contain random failures of systems and/or seismic-induced failures and delineate the plant behaviour after the seismic initiators.

  It was assumed that the operator faces a complex situation due to the supplementary stress of a seismic event. Performance shaping factors account for the intensity of the earthquake and the elapsed time from the earthquake separately for operator actions in the control room and for the actions on the field. The Human Error Probability (HEP) for execution of actions during seismic events was obtained by multiplying the HEPs calculated for internal events PSA with the performance shaping factors. It was assumed that for higher earthquakes, where the integrity of the MCR and the MCR functionality is not maintained, the plant operation is performed from the seismically qualified SCA.

  All Level 1 sequences were analysed in the Level 2 report. The PSA-based SMA ASQ process yields the results in two types of cutsets leading to SCD: seismic cutsets and mixed cutsets. Seismic cutsets include only seismic-induced failures. Using the MIN-MAX method, the HCLPF value of a seismic cutset is the maximum of the seismic failures in the cutset. Mixed cutsets include seismic-induced and random component failures or operator errors. From the Level 1 results, the sequences were grouped according to two representative severe core damage accidents: Station Black Out and Small LOCA. Similar to Internal Events, the sequences within each of the two primary groups were further grouped according to similar plant configuration accounting for the status of mitigating systems, the impact on the containment system availability, and the containment status. Similar to Internal Events, PSA based SMA applied the UPM and alpha method CCFs, included ASEP and THERP derived HRA, and accounted for dependency between operator actions.

- Calculate the HCLPF value for each seismic core damage sequences

- The plant HCLPF is the lowest sequence HCLPF
- Performed Sensitivity analyses for seismic events. Sensitivity analyses were performed for Level 2 PSA based SMA. For seismic events no uncertainty is required as the uncertainty is directly reflected in the PSA based SMA method

PSA Results

During the numerous analyses performed as part of this PSA study, a number of recommendations were made and accepted with respect to plant design and operation. These addresses issues such as revision of testing and maintenance plans, minor maintenance issues (improvement to anchorage), and design changes. Two design changes that have a significant impact on the Level 2 results are theinstallation of an emergency filtered venting system for containment and a make-up system to the calandria vault.

With the changes, the PSA results for internal events, fire and flood are lower than the 1E-04 events/year and 1E-05 events/year limits for SCDF and LRF respectively. The LRF results are below the plant safety goal of 1E-06 events/year. In addition, the seismic capacity meets the HCLPF safety goal of 0.30g for SCDF and is higher than the HCLPF safety of 0.40g for LRF.

Safety Analysis

The Point Lepreau Refurbishment safety analysis plan was developed based on:

- A systematic review of the initiating events, event combinations and event sequences that are described in the C6 rev. 1 (Reference-11). The events that are not covered by accident analyses currently in the Safety Report and which have a frequency higher than $1x10^{-6}$/year have been included in the safety analysis plan. Moreover, the common cause events relevant to the Point Lepreau site will be documented. The likelihood of various relevant common cause events will be specified, the inherent design features that provide protection against these events, and the contingency procedures to mitigate their consequences will be identified.
- A systematic review of the Safety Report to identify the accident scenarios whose consequence assessments could be affected by the design changes that are planned to be implemented during the refurbishment outage or by the plant conditions that will prevail after the refurbishment outage and which are not currently covered in the Safety Report, namely the use of fresh fuel and new uncrept fuel channels.
- A provision to perform the deterministic safety analyses required to support the PSA is included in the plan.

All the analyses performed in support of the Point Lepreau Refurbishment have followed rigorous quality assurance requirements. They have been performed using the most up-to-date methodologies and computer codes that comply with current standards. They have also been submitted to the CNSC. The station's safe operating limits will be updated to account for the results of these new analyses. The Safety Report is also being revised to reflect the new analysis.

A summary of the main analyses included in the Safety Analysis Plan and performed in support of the Refurbishment project is provided below.

Large LOCA Analysis

The large LOCA analysis was aimed at confirming the adequacy of the reference fuel channel design to accommodate fuel string expansion in the conditions that will prevail after refurbishment. The replacement fuel channel includes a change to the axial location of the shield plugs that will increase the initial axial gap between the fuel and shield plugs compared to the original design. The analysis was performed with updated codes and methodology and has demonstrated that the replacement fuel channel axial gap between the fuel string and the shield plugs is sufficient to preclude any constrained expansion during postulated critical accidents (Reference-12).

The impact of pre-equilibrium core conditions on the large LOCA power pulse was also assessed since the pre-equilibrium core state is the time of maximum poison concentration in the moderator, which enhances the void reactivity. The fresh fuel void reactivity is also greater than for the equilibrium conditions. Conversely, the delayed neutron fraction in a fresh core is about 20% greater than that found in the fuel at the equilibrium core state that acts to significantly reduce the power pulse. Safety analysis for large LOCA in the pre-equilibrium core with the traditional initial core loading was performed with updated codes and methodology as in Reference-12. The analysis demonstrated that the targets for the power pulse calculation are met.

Shutdown System Depth Analysis for Pressure Tube/Calandria Tube Rupture with Loss of ECC with Fresh Fuel and Pressure Tube Rupture Trip Coverage

The objective of the analysis was mainly to assess shutdown system number 1 reactivity depth for a pre-equilibrium core. Although the analysis focused on pre-equilibrium core conditions, some equilibrium core cases were also performed to update the assessment in the Safety Report, using current methodology and validated codes. Analysis was also performed to assess the performance of the high moderator level trip that are being implemented during the refurbishment outage for pressure tube rupture and flow blockage events. The trip coverage analysis was performed for various power levels and accounted for the maximum amount of poison in the moderator during a restart following a long outage.

Options to improve reactivity depth, such as various initial fuel loadings of depleted Uranium fuel were also assessed. In conclusion, it is considered that maintaining the traditional initial core load minimizes the overall risk. Any residual risk associated with the SDS1 depth following an in-core LOCA, can be adequately managed by establishing limits of operating parameters and through awareness training of the control room staff that would be part of the just-in-time training related to fresh-core start-up.

An assessment of the adequacy of the SDS1 reactivity depth following a flow blockage occurring at full power was also performed using a methodology similar to the one referred above for the analysis of the SDS1 reactivity depth following a pressure tube rupture. The analysis assumed that the maximum amount of poison that could be present in the moderator is equal to the amount that would be present during steady-state operation with the reactor at the plutonium peak. The maximum amount of poison that could be present in the moderator during a start-up following a long outage is larger than the value during steady-state operation and therefore would be more limiting. However, since flow verification tests are performed during reactor start-ups, it is considered not credible that a complete flow blockage could occur at high power during a reactor start-up.

Loss of Heat Transport System Flow Events

The objectives of the loss of forced circulation analysis was to assess the trip coverage improvement provided by the additional SDS2 high pressure trip instrumentation on primary heat transport outlet headers 3 and 7, and to finalize the redesign of the trip setpoints and conditioning power levels for the low flow, low pressure differential and high reactor outlet pressure trips.

The analyses focused on the plant physics and thermalhydraulic conditions that will prevail after the refurbishment outage. The analyses covered both pre-equilibrium and equilibrium fuel core configurations. The analyses also included sensitivity cases that covered the range of estimated conditions long after refurbishment, including the aged conditions. The analysis covered all the loss of forced circulation events currently included in Point Lepreau Safety Report.

The preliminary design assist analysis demonstrated trip coverage improvements for the loss of heat transport system flow events in a number of areas and provided insight to finalize the trip setpoint design (Reference-13). The preliminary design assist analysis used the point kinetics methodology in full-circuit simulation.

For the final analysis and for a number of limiting cases, coupled thermalhydraulic and 3-dimensional time dependent neutron kinetics simulations were performed to better assess the impact of neutronics response on trip coverage and to augment the analysis performed using the point kinetics methodology in full-circuit simulation, following loss of heat transport system flow events. A comparison between results obtained using point kinetics and 3-dimensional neutronics has also demonstrated that there is no significant effect of the two different methodologies on the predicted trip times and the timing of dryout.

<u>Regional Over Power Trip Analysis with New Heat Transport System Conditions (to restore set-points)</u>

The analysis of slow loss of reactivity control was updated to determine the ROPT detector trip setpoints that will apply following the refurbishment outage. The analysis included analysis of the pre-equilibrium core (i.e. until equilibrium burnup from fuelling is achieved) and analysis of the equilibrium core. For the equilibrium core, the analysis also addressed the post-refurbishment ageing of the heat transport system.

The critical channel power distributions for the various flux shapes were recalculated using the thermalhydraulic conditions that are predicted to prevail after refurbishment. The channel and bundle power distributions used to calculate the setpoints applicable during the pre-equilibrium phase were based on RFSP simulations. The setpoints for the equilibrium core were based on current ripples. The analysis allowed the removal of plant ageing related penalties which will allow the reactor to achieve full power following refurbishment.

<u>Moderator Events</u>

The review of C6 identified that in addition to the slow moderator drain event currently included in Point Lepreau Safety Report, a number of other failures that can affect the moderator system have to be analysed. Among these other events, the most significant ones are the Loss of Service Water to the Moderator Heat Exchangers (LOSW) and the Loss of Moderator Circulation (LOMC).

As part of the activity scope of the Refurbishment project, NB Power has taken the opportunity to further improve the defence in depth for moderator related events by adding SDS1 and SDS2 trips to the moderator system. Preliminary design proposed high-level trips to provide coverage for loss of moderator heat sink events (Reference-13). However analysis revealed that the proposed design of the high moderator level trips did not meet the specified requirements for the protection from loss of moderator heat sink events (LOSW or LOMC). Accordingly, an alternative design was implemented. The existing SDS1 trip on high temperature was retained, and a new SDS1 trip on low differential pressure between the inlet and outlet header of the moderator pumps was added to provide the necessary trip coverage for the loss of moderator circulation event. For moderator drain, trip coverage is enhanced by the new low moderator level trips on both SDS1 and SDS2.

Assessments of the consequences of the moderator temperature control, moderator heat exchanger tube failure and moderator cover gas system failure were also performed.

<u>Loss of End Shield Coolant, Flow and Heat Sink</u>

The review of C6 identified that more detailed analyses of the failures of the shield cooling system were needed to be included in the Point Lepreau Safety Report. An assessment of the consequences of the various following failures were performed:

- Loss of end shield cooling inventory, including shield cooling single heat exchanger tube failure,
- Loss of end shield coolant flow, and
- Loss of end shield cooling heat sink.

The response of the end shield cooling system following each postulated accident was assessed to determine if the potential exists for differential tubesheet deformation, which can damage fuel channels and/or shutoff rod assemblies. The analysis was performed using the CATHENA code together with a CATHENA model of the End Shield Cooling System for PLGS (Reference-14). In all end shield cooling system failure events, the analysis results have shown that there are sufficient indications available for the operator to recognize that a transient exists in the end shield cooling system. The differential temperature between the inner and outer tube sheets very much depends on the timing of reactor shutdown or the setback and the method of primary heat transport system cooldown. An assessment of the thermal analyses predictions of the temperature profile with the appropriate temperature profiles used in previous stress analyses has also demonstrated that the ASME Code service limits are met.

Shutdown Cooling Events (including Failure of LRV during Entrance in Shutdown Cooling Mode)
The review of C6 revision 1 identified that a number of events affecting the shutdown cooling system have to be analysed. More specifically, the events identified to be analyzed were residual heat removal system failure, including:
- Loss of primary coolant inventory during shutdown cooling mode, including shutdown cooling single heat exchanger tube failure,
- Loss of primary coolant flow during shutdown cooling mode,
- Loss of heat sink during shutdown cooling mode,
- A failure of the primary pressure relief valve in shutdown cooling mode and in particular when the plant is being cooldown, and
- Isolation valve failures.

There are three separate and distinct operating conditions while the shutdown cooling system is in use:
- Primary heat transport system full and vented,
- Primary heat transport system full and not fully vented, and
- Primary heat transport system drained.

All three operating conditions, including operator actions, were considered in developing an accident analysis matrix. Deterministic analyses of events, while in shutdown cooling mode of operation, have been performed. Full circuit simulations were performed for loss of coolant events with the heat transport system full and vented or partially vented and with the heat transport system full but not vented. The CATHENA code was used for thermalhydraulic simulations together with an integrated plant model, which incorporated the shutdown cooling system. The analysis of these Large LOCA cases demonstrates that integrity of the fuel channels and emergency core cooling performance requirements are ensured. For the small LOCA events, fuel integrity is also demonstrated. For the loss of recirculating cooling water events during shutdown cooling mode, bounding calculations also show that there is no risk to fuel and fuel channel integrity.

Multiple Boiler Tube Failure
The review of C6 identified that the consequences of the failure of a large number of boiler tubes is currently not in the Point Lepreau Safety Report. Consistent with the methodology developed for Darlington and used for recent offshore projects, a simultaneous guillotine break of up to 10 steam generator tubes was analysed. The analysis included an assessment of:
- The trip coverage effectiveness,
- The thermalhydraulic response of the primary and secondary heat transport circuits,
- The radionuclide releases to the secondary side and from it to the environment, and
- Population dose.

The integrated break discharge, the transient, and the integrated steam release to the atmosphere for different break sizes and break locations were assessed since plant response is different depending on the break discharge (i.e., number of tubes failed), break location, and whether or not the steam generator level setpoints for turbine trip and condenser steam discharge valve/feedwater motorized valve closure are reached. Dose calculations were performed based on CAN/CSA N288.2-M91 standard methodology, following the guidance from AECB-1059 and C-6 Rev. 1 and have demonstrated compliance.

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