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November 8, 2016
GO2-16-119

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 LICENSE AMENDMENT REQUEST FOR ONE-TIME 7 DAY EXTENSION OF COMPLETION TIME FOR TS CONDITION 3.5.1.A, 3.6.1.5.A, AND 3.6.2.3.A**

References:
1. Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk- Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, dated May 2011
2. RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," Revision 1, dated May 2011
3. RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009

Dear Sir or Madam:

In accordance with the provisions of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Energy Northwest requests an amendment to the Technical Specifications (TS) for Columbia Generating Station (Columbia). The proposed amendment would on a one-time basis, extend the completion time by 7 days for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A. This one-time extension will be used to support preventive maintenance (PM) which replaces the Residual Heat Removal (RHR) train A subsystem's pump and motor.

The proposed changes have been evaluated using the risk-informed processes described in RG 1.174 (Reference 1), and RG 1.177 (Reference 2). The risk associated with the proposed amendment for a one-time 7 day completion time extension for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A was found to be acceptable.

The attached request is subdivided as shown below.

Attachment 1 provides an evaluation of the proposed changes.

Attachment 2 provides the proposed TS markup pages.

Attachment 3 provides the clean pages of the proposed TS changes.

Attachment 4 provides a summary of regulatory commitments contained in this letter.

Attachment 5 provides the supporting risk-informed Probabilistic Risk Assessment (PRA).

Attachment 6 provides the evaluation of the Technical Adequacy of Columbia PRA in accordance with RG 1.200 (Reference 3).

Energy Northwest requests approval of the proposed license by November 10, 2017 to support the preventive maintenance under Engineering Change 14635 currently scheduled for December 2017 following NRC approval. Energy Northwest will implement the amendment within 60 days of the NRC approval date.

If there are any questions or if additional information is needed, please contact Mr. R.M Garcia, Licensing Supervisor, at 509-377-8463.

In accordance with 10 CFR 50.91, Energy Northwest is notifying the State of Washington of this amendment request by transmitting a copy of this letter and attachments to the designated State Official.

I declare under penalty of perjury that the foregoing is true and correct. Executed this 8th day of November, 2016.

Respectfully,

A. L. Javorik
Vice President, Engineering

Attachments: As stated

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Sr. Resident Inspector - 988C
CD Sonoda - BPA1399 (email)
WA Horin - Winston & Strawn
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Evaluation of Proposed Change

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1.0 SUMMARY DESCRIPTION

This evaluation supports a revision to the completion time (CT) specified in Columbia Generating Station (Columbia) Technical Specification (TS) conditions 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A, by adding a footnote to each of the required actions to allow a one-time 7 day extension (14 day completion time) for restoring Residual Heat Removal (RHR) train A. The extended CT will allow sufficient time to complete the preventive maintenance to install a new pump and motor in the RHR train A subsystem. This change supports the implementation of the Columbia Equipment Reliability program associated with the Emergency Core Cooling System (ECCS). The proposed change also deletes a footnote associated with TS conditions 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A which expired at 05:00 PST on February 9, 2015. The proposed change is similar to a license amendment granted February 1, 2015 (ML15030A501).

2.0 DETAILED DESCRIPTION

2.1 RHR MODES OF OPERATION

The RHR system is comprised of three independent trains. Each train contains its own motor-driven pump, piping, valves, instrumentation, and controls. In addition, the A and B trains have heat exchangers which are cooled by standby service water (SW).

<u>Low-pressure coolant injection (LPCI) mode</u>

There are 3 redundant subsystems in the LPCI mode of RHR: train A, train B, and train C. The RHR system's LPCI mode automatically pumps suppression pool water into separate lines and core nozzles for injection into the core region of the reactor pressure vessel (RPV) following a loss of coolant accident (LOCA). The RHR system's LPCI mode operates in conjunction with the other ECCS subsystems to provide adequate core cooling for all design basis LOCA conditions. The LPCI mode of RHR is part of the ECCS along with the high pressure core spray (HPCS) subsystem, the low pressure core spray (LPCS) subsystem and the automatic depressurization system (ADS).

<u>Suppression pool cooling (SPC) and containment spray cooling (CSC) modes</u>

There are 2 redundant subsystems in the SPC and CSC modes of RHR: train A and train B. The RHR system's SPC and CSC modes provide heat removal from the suppression pool and containment by pumping suppression pool water through the system's heat exchangers and discharging the water either directly back to the suppression pool (i.e., in the SPC mode) or discharging the water to the wetwell and drywell spray spargers (i.e., in the CSC mode) where the water is then returned, by drainage, back to the suppression pool. These modes of operation are designed to provide cooling to maintain containment and suppression pool temperatures and

pressures following major transients. For the CSS mode, only drywell spray mode is included in the Columbia TS.

Shutdown Cooling Mode

The RHR system's normal shutdown cooling mode removes reactor core decay and sensible heat from the primary reactor system to permit refueling and servicing. This heat removal function is initiated manually after the reactor pressure has been reduced to less than 48 psig (295°F) by discharge of steam to the main condenser. This mode of operation provides the capability to cool down the reactor under controlled conditions.

2.2 APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of safety analysis break sizes for a postulated LOCA. The limiting single failures are discussed in the Columbia final safety analysis report (FSAR) Chapter 6. For a large break LOCA, failure of ECCS subsystems in electrical division 1 which consists of LPCS and LPCI train A, or electrical division 2 which consist of LPCI train B and LPCI train C, due to failure of its associated diesel generator is, in general, the most severe failure. For a small break LOCA, HPCS system failure is the most severe failure. The small break analysis also assumes two ADS valves are inoperable at the time of the accident. The remaining operable ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

While RHR train A is out of service, the remaining required operable ECCS subsystems ensure the 10 CFR 50.46 limits are not exceeded.

The RHR CSC mode for drywell spray is credited for two functions in the LOCA analysis. The RHR CSC for drywell spray is credited for scrubbing inorganic iodines and particulates from the primary containment atmosphere. This function reduces the amount of airborne activity available for leakage from primary containment. The RHR drywell spray is also credited for primary containment pressure reduction. This function reduces the leak rate of airborne activity from primary containment.

2.3 CURRENT TS REQUIREMENTS

TS 3.5.1 limiting condition for operation (LCO) requires each ECCS injection/spray subsystem and six ADS valves to be OPERABLE. The TS applies in MODE 1, MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig. The ECCS injection/spray subsystems are defined as the three LPCI subsystems, the LPCS System, and the HPCS System. The low pressure ECCS injection/spray subsystems are defined as the LPCS System and the three LPCI subsystems.

The LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below 48 psig reactor steam dome pressure in MODE 3, if capable of being manually realigned either remotely or locally to the LPCI mode and not otherwise inoperable. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is being realigned from or to the RHR shutdown cooling mode. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.

TS 3.6.1.5 LCO requires two RHR drywell spray subsystems to be OPERABLE. The TS applies in MODES 1, 2, and 3. In the event of a Design Basis Accident (DBA), a minimum of one RHR drywell spray subsystem is required to mitigate the effects of potential bypass leakage paths and maintain the primary containment peak pressure below design limits.

TS 3.6.2.3 requires two RHR suppression pool cooling subsystems to be OPERABLE. The TS applies in MODES 1, 2, and 3. In the event of a DBA, a minimum of one RHR drywell spray subsystem is required to mitigate the effects of potential bypass leakage paths and maintain the primary containment peak pressure below the design limits in Columbia FSAR Section 6.2.2.3.

2.4 REASON FOR THE PROPOSED CHANGE

The proposed change would allow preventive maintenance (PM) in support of Engineering Change (EC) 14635 to replace RHR train A pump and motor. This EC is part of the Energy Northwest long range plan for improving reliability of the RHR system pumps and motors. The pump replacement also satisfies the corrective action for a 10 CFR 21 notification for this model pump.

The work to implement the replacement of RHR train A pump and motor is being scheduled to minimize unavailability time for the RHR system. Previous experience from replacing just the RHR train B pump in May 2015 required approximately 5 days out of the 7 day completion time for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A. Since just replacing the RHR train B pump took most of the current allowed outage time, it is expected that the work needed to replace both the RHR train A pump and motor under engineering change (EC) 14635 will not be able to be completed within the current 7 day CT and would necessitate a plant shutdown. The proposed addition to the CT provides margin to ensure adequate time is provided for the pump and motor replacement.

In addition, the proposed change to delete a footnote from TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A is due to this note being no longer applicable.

2.5 PROPOSED CHANGE

The proposed change will revise the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A by adding a footnote for restoring RHR train A to each of the required actions to allow a one-time 7 day extension (14 day completion time). The footnote will state:

> (1) The Completion Time that one train of RHR (RHR-A) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-A following pump and motor replacement. This footnote will expire at 23:59 PST February 28, 2019.

The proposed change will also delete the following footnote from TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A which is no longer applicable:

> (1) The Completion Time that one train of RHR (RHR-B) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-B from the modification activity. Upon successful restoration of RHR-B, this footnote is no longer applicable and will expire at 05:00 PST on February 9, 2015.

The markup of the proposed changes to TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A is provided in Attachment 2 of this letter.

3.0 TECHNICAL EVALUATION

The proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A has been evaluated using the risk-informed approach for assessing changes to TS allowed outage times (AOT) or CT described in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," (References 1, and 2 respectively).

RG 1.200 "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment [PRA] Results for Risk-Informed Activities," provides guidance for determining the technical adequacy of an internal events Probabilistic Safety Assessment (PSA) (Reference 3).

This section provides the risk-informed evaluation of the proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A, against the five key principles involved in implementing risk informed decision-making for assessing TS changes.

As provided in RG 1.177 the five key principles are:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

3.1 KEY PRINCIPLE #1: COMPLIANCE WITH CURRENT REGULATIONS AND COMMITMENTS

The LPCI operational mode of the RHR system complies with the applicable Nuclear Regulatory Commission (NRC) Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50), Appendix A, General Design Criteria (GDC). The specific GDC applicable to the proposed license amendment are GDC 35, “Emergency Core Cooling,” GDC 36, “Inspection of Emergency Core Cooling System,” and GDC 37, “Testing of Emergency Core Cooling System.” The proposed change only affects the TS CT and does not involve a change to the design or method of operation of the ECCS. Conformance to the GDC criteria is not altered by the proposed change.

The SPC and CSC operational modes of RHR system comply with GDC 38, “Containment Heat Removal." The containment heat removal function is accomplished by the SPC and CSC operational modes of RHR train A and train B. Following a LOCA, one or both of the SPC and CSC modes of RHR would be initiated. The two cooling trains, A and B, are each mechanically and electrically separate from the other to achieve redundancy. The proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A does not alter the redundant design capability of the RHR system to accomplish these functions.

The regulatory requirements related to the contents of TS are contained in 10 CFR 50.36. Pursuant to 10 CFR 50.36, TS are required to contain (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation (LCO), (3) surveillance requirements, (4) design features, and (5) administrative controls.

The regulations in 10 CFR 50.36(c)(2) state that the LCOs are the lowest functional capability or performance level of equipment required for safe operation of the facility

and when LCOs are not met, the licensee shall shut down the reactor or follow any remedial actions permitted by the TS until the LCO can be met.

This proposed revision to the Technical Specifications (TS) provides the plant-specific review that supports a finding of continued adequate protection of public health and safety. Although it is less restrictive than the current TS requirement, it still affords adequate assurance of safety when judged against the current regulatory standards. The proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A associated with limiting conditions for operation is proposed in accordance with 10 CFR 50.90. The detailed technical evaluation is provided in Sections 3.2 through 3.10.

10 CFR 50.46 provides acceptance criteria for ECCS. A summary of the 10CFR 50.46(b) acceptance criteria is provided below:

- Peak cladding temperature – the calculated maximum fuel element cladding temperature shall not exceed 2200 °F.

- Maximum cladding oxidation – the calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

- Maximum hydrogen generation – the calculated total amount of hydrogen generation from chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

- Coolable geometry – calculated changes in core geometry shall be such that the core remains amenable to cooling.

- Long-term cooling - after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A does not modify any ECCS component or the ECCS system’s ability to perform its safety related function. Thus, compliance with 10 CFR 50.46 is not altered by the proposed change.

There are no other license conditions or orders applicable to this proposed change.

3.2 KEY PRINCIPLE #2: DEFENSE IN DEPTH

The impact of the proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A was evaluated and is consistent with the defense-in-depth philosophy and also ensures the protection of the public health and safety.

The defense in depth is primarily maintained by the TS requirements for operability of the remaining RHR trains. The limited increase in unavailability for the proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A for the LPCI mode of operation subsystem, SPC mode of operation subsystem, and CSC mode of operation subsystem does not significantly change the balance among the defense-in-depth principles of prevention of core damage, prevention of containment failure, and consequence mitigation. The proposed change does not impact the capability of the remaining ECCS subsystems to provide adequate core cooling, suppression pool cooling, or containment spray. The small increase in unavailability for a single ECCS subsystem does not significantly change the balance among the defense-in-depth principles. Additionally while on-line, more ECCS systems and or subsystems are available to mitigate a design basis accident such as HPCS, LPCS, LPCI mode of RHR and the ADS system.

Over-reliance on programmatic activities to compensate for weakness in plant design is avoided. Columbia's normal administrative controls will be used during the extended completion time which includes protecting opposite train systems and support systems. Compensatory measures above and beyond those required by plant procedure will be employed as further risk management actions. However, no credit for these measures were taken in the probabilistic risk assessment (PRA) performed for the proposed CT extension. This programmatic activity does not constitute over-reliance on programmatic activities to compensate for weakness in the Columbia plant design. Administrative controls ensure that redundant ECCS subsystems are not removed from service thereby maintaining system redundancy.

There are two redundant SPC and CSC subsystems for containment heat removal at Columbia. There are three redundant LPCI subsystems for ECCS at Columbia. Both the Columbia ECCS and containment heat removal systems are robust and diverse. In addition to the LPCI subsystems, the ECCS is also composed of diverse HPCS and LPCS subsystems. There is no change in the automatic operational responses to accident signals or are any abnormal lineups proposed for the ECCS, SPC or CSC subsystems. Additionally, independence, redundancy, and diversity are maintained during the increased CT period. Restrictions are placed on simultaneous equipment outages that would erode the principles of redundancy and diversity.

Planning for preventative maintenance activities during the proposed extension will assess overall plant risk and establish compensatory measures in accordance with

Columbia's risk management program. Additional compensatory measures for avoidance of risk significant configurations are discussed in Section 3.6.

Voluntary removal of redundant equipment from service as allowed by TS and risk assessment is reassessed when severe weather conditions are predicted or at a time when the plant may be subjected to other abnormal conditions.

Columbia's risk management configuration control program protects redundant equipment and systems to assure redundancy and diversity are maintained. Additionally, Columbia uses procedural controlled risk informed approaches for scheduling maintenance for all modes of plant operation, which limits removal of risk sensitive equipment combinations from service.

For the RHR train A subsystem outage during the extended CT, the redundant subsystems of RHR LPCI mode of operation and the ECCS diverse subsystems HPCS and LPCS as well as the redundant SPC and CSC subsystems will continue to be capable of performing the safety functions assumed in Columbia's FSAR Chapter 15 accident analyses.

The potential for a common cause failure is not increased. No change in system alignment or operational modes is proposed. The proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A for the LPCI mode of operation subsystem, SPC and CSC mode of operation subsystems, will not significantly increase the unavailability such that new common cause failures not previously considered would occur.

Additionally, movement of the RHR train A pump and motor has been planned to ensure the load path does not affect other ECCS equipment. The floor plugs remain in place the RHR train B pump room and the reactor core isolation cooling (RCIC) pump room during load travel. The removed floor plugs from the RHR train A pump room will be transported on rollers and staged above the RCIC pump room floor plugs in accordance with plant procedure. The removed pump and motor will be set into the staging area setup at the 471 elevation of the reactor building and then lowered to the truck bay for shipping. The load paths as well as floor loading of the staging areas has been evaluated to ensure these locations contain margins of safety for the additional floor loads during the pump and motor replacement activity. This ensures that no new common cause failures will occur during this replacement.

The independence of physical barriers is not degraded. For the proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A LPCI mode of operation subsystem, SPC mode of operation subsystem, and CSC mode of operation, the redundant low pressure ECCS subsystems and the diverse HPCS system, as well as the redundant SPC mode of operation and CSC mode of operation subsystems will be protected, providing sufficient independent, redundant and diverse

means to prevent any undue challenges to the fuel cladding, reactor coolant pressure boundary, and the containment from occurring.

Defenses against human errors are maintained. The proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A LPCI mode of operation subsystem, SPC mode of operation subsystem, and CSC mode of operation will not change the mode of operation, introduce additional operator challenges, or cause any new operator actions in response to normal, abnormal or postulated accident conditions. The risk management categorization for this evolution is a “high” risk classification in accordance with Columbia procedures. High risk work activities require a documented action plan and Plant Operations Committee (POC) approval. This provides heightened awareness to reduce human errors.

The intent of the plant design criteria is still met. There are no design changes to the RHR system associated with the proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A LPCI mode of operation subsystem, SPC mode of operation subsystem, and CSC mode of operation. Therefore the criteria in NRC Standard Review Plan (SRP) 5.4.7 are still met. There are no design changes to the Columbia containment for the proposed change. Therefore the criteria in NRC SRP 6.2.1.1.C are still met. There are no changes to the design of the ECCS system associated with the proposed change. Therefore, the criteria in NRC SRP 6.3 are still met.

3.3 KEY PRINCIPLE #3: SUFFICIENT SAFETY MARGINS

The proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A for RHR train A LPCI mode of operation subsystem, SPC mode of operation subsystem, and CSC mode of operation does not conflict with existing codes and standards applicable to Columbia. There are no codes or standards that place a limit on the CT for ECCS subsystems or containment heat removal systems. There is no change in plant configuration associated with the proposed changes. The same configuration conditions exist for the extended CT as is currently allowed for a CT of 7 days. The safety analysis acceptance criteria would continue to be met during the extended CT. Specific limitations exist within Columbia’s TS that identify conditions where more restrictive CT limits would be required should other ECCS subsystems or if other containment heat removal SPC or CSC subsystems become unavailable at the same time. Required Actions associated with these more restrictive conditions assure adequate safety margins are preserved for maintaining the intended ECCS safety function for LPCI as well as SPC and CSC.

3.4 KEY PRINCIPLE #4: RISK IMPACT

The NRC identified a three tiered approach for licensees to evaluate the risk in RG 1.177 associated with proposed CT changes.

> Tier 1 is an evaluation of the impact on plant risk of the proposed TS change as expressed by the change in core damage frequency (CDF), the incremental conditional core damage probability (ICCDP), and when appropriate, the change in large early release frequency (LERF) and incremental conditional large early release probability (ICLERP).
>
> Tier 2 is an identification of potentially high-risk configurations that could exist if equipment in addition to that associated with the change were to be taken out of service simultaneously, or other risk-significant operational factors such as concurrent system or equipment testing were also involved. The objective of this part of the evaluation is to ensure that appropriate restrictions on dominant risk-significant configurations associated with the change are in place.
>
> Tier 3 is the establishment of an overall configuration risk management program to ensure that other potentially lower probability, but nonetheless risk-significant configurations resulting from maintenance and other operational activities, are identified and appropriate compensation taken.
>
> If the Tier 2 assessment demonstrates, with reasonable assurance, that there are no risk-significant configurations involving the subject equipment, the application of Tier 3 to the proposed TS CT change may not be necessary. Although defense in depth is protected to some degree by the current TS, application of the three tiered approach to risk-informed TS CT changes discussed below provides additional assurance that defense-in-depth will not be significantly impacted by such changes to the licensing basis. Energy Northwest has evaluated the proposed extension of the TS CT using the guidance of RG 1.177 and the results are provided.

Sections 3.5 through 3.8 address the risk informed impact.

3.5 TIER 1: PSA EVALUATION OF RISK IMPACT

Tier 1 is an evaluation of the impact on plant risk of the proposed TS change as expressed by the change in CDF, the ICCDP, the change in LERF and ICLERP. The results of this evaluation, the PRA insights, a discussion of the uncertainty associated with these results, and the capability of the PRA model of record assessed to RG 1.200 (Reference 3) for the proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A are presented in Attachment 5.

The risk evaluation considers the impact of the proposed change with respect to the risks due to internal events, internal fire, seismic, and other external hazards. A quantitative evaluation was performed for internal event risks. Qualitative and quantitative evaluations were performed for fire and seismic risks. A qualitative evaluation was performed for other external hazards.

Table 1 provides a summary of the approach and results of the evaluation of each of the potential risk contributors. These analyses demonstrate that the risk impact of the proposed one-time change in the CT for TS conditions 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A are small and below the acceptance guidelines.

Table-1
SUMMARY OF RISK INSIGHTS

Risk Contributor	Approach	Insights
Internal Events	Quantify ICCDP & ICLERP for planned configuration • ICCDP < 1E-6 • ICLERP < 1E-7	• Base risk within acceptance guidelines • Compensatory measures keep risk well within the acceptance guidelines
Internal Fire	Qualitatively and quantitatively evaluated: • Identify scenarios impacted by configuration • Estimate risk impacts due to configuration and quantify ICCDP and ICLERP • Identify compensatory measures	• Base risk within acceptance guidelines • Compensatory measures keep risk well within the acceptance guidelines • New fire-related compensatory measures identified

Table-1
SUMMARY OF RISK INSIGHTS

Risk Contributor	Approach	Insights
Seismic	Qualitatively and quantitatively evaluated: • Identify scenarios impacted by configuration • Estimate risk impacts due to configuration and quantify ICCDP and ICLERP • Identify compensatory measures	• Seismic risk impacts negligible • Seismic risk reduced with compensatory measures for internal events and fire
Other External Hazards	Qualitatively evaluate based on the CGS IPEEE	• Other external event risks are negligible contributors
Overall At-Power Risks	Quantify ICCDP & ICLERP for planned configuration • ICCDP < 1E-6 • ICLERP < 1E-7	• Base risk within acceptance guidelines • Compensatory measures keep risk well within the acceptance guidelines

The numeric results are summarized in Table 2 for RHR Train A out of service (OOS) for the extended CT. Since this is a one-time change and there is no permanent change to the plant CDF or LERF, delta CDF and delta LERF values are not included as would normally if requesting a permanent change request. The results are compared with the risk acceptance guidelines described in Reference 2. The values for the Incremental ICCDP and the ICLERP demonstrate that the proposed one-time change in the CT for TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A is small and below the acceptance guidelines.

Table-2 14-DAY AOT PREVENTIVE MAINTENANCE Average Maintenance Model with protected Trains per plant procedures		
Risk Metric	**Acceptance Guideline**	**PRA Results**
RHR-A_OOS - ICCDP (Total)	< 1.0E-6	6.25E-7
RHR-A_OOS- ICCDP (Internal Events)		2.10E-7
RHR-A_OOS- ICCDP (Fire)		2.73E-7
RHR-A_OOS- ICCDP (Seismic)		1.41E-7
RHR-A_OOS - ICLERP (Total)	< 1.0E-7	6.90E-10
RHR-A_OOS - ICLERP (Internal Events)		7.67E-11
RHR-A_OOS - ICLERP (Fire)		6.14E-10
RHR-A_OOS - ICLERP (Seismic)		Negligible

The technical adequacy of Columbia's PSA has been evaluated and contains sufficient detail and features to perform this evaluation. The technical adequacy of Columbia's PSA Revision 7.2 was reviewed for TSTF-425 and is discussed in detail in Attachment 6 of this submittal.

In PSA minor model Revision 7.2.1 selected dependent Human Error Probabilities (HEP) values were modified for use in the non-LERF Level 2 release term quantification and separate recovery files were constituted to implement these recoveries. The revisions associated with this model release had no impact on the CDF or LERF modeling or quantitative solutions; therefore, the release of this model version had no impact on the assessed technical adequacy of the PSA with respect to the ASME Standard.

The Columbia PRA models are technically adequate to support this risk assessment as described in Attachments 5 and 6.

3.6 TIER 2: AVOIDANCE OF RISK SIGNIFICANT PLANT CONFIGURATIONS

High-risk plant configurations were identified and the following compensatory measures will be established to lessen the calculated increase in core damage risk when RHR train A is OOS. The compensatory measures will provide additional risk reduction not included in the ICCDP and ICLERP calculated for the procedurally required protected equipment.

Based on the risk-significant contributors to the ICCDP, the following compensatory measures will be taken during the 14 day CT for the RHR train A OOS to address the risk-significant scenarios discussed in Attachment 5.

Compensatory Measures

1. In addition to the protected equipment required by plant procedures when RHR train A is OOS, which are RHR train B, Diesel Generator (DG2) and SW train B; the following equipment will be protected HPCS, high pressure core spray service water (HPCS-SW), startup transformer (TR-S), DG3, RCIC, LPCS and RHR train C.

2. While no credit is taken in the PRA evaluation in Attachment 5, the Control Air System (CAS), Standby Gas Treatment system (SGT), and the division 2 DC battery chargers will be protected.

3. A flood watch tour of the Turbine Building (TB) to provide early detection of internal floods will be established.

4. A fire watch tour of the TB corridor (TG-12), Reactor Building elevations 471' and 522', cable chase, and the Division 2 electrical switchgear rooms will be established.

5. To increase fire risk awareness, shift briefs or pre-job walk downs will be performed to reduce and manage transient combustibles.

6. Fire risk management actions required per plant procedure which include the following will be established:

Fire Risk	**Zone Description**	**Risk Management Actions**
RW 437 N	Waste Tank Floor Area C106; RW 437 Behind Shield Wall; Offgas Dryer Rooms C130 and C131	Initiate Fire Prevention Evaluation Permit per PPM 1.3.10 to: - implement hourly fire tours for Waste Tank Floor Area C106 per FPP-1.7 - ensure no unnecessary fire hazards and no obstructed fire equipment in Waste Tank Floor Area C106 - confirm area detection, suppression and fire doors available for Waste Tank Floor Area C106, RW 437 Behind Shield wall and Offgas Dryer rooms C130 and C131 - prohibit hot work in Waste Tank Floor Area C106, RW 437 Behind shield wall and Offgas Dryer rooms C130 and C131 - update plant status page with plant fire risk status and affected fire area for station awareness

Fire Risk	**Zone Description**	**Risk Management Actions**
RW 467 RPS 2 CHG 2	Division 2 RPS Room C213; Division 2 Battery Charger Room C224	Initiate Fire Prevention Evaluation Permit per PPM 1.3.10 to: - implement hourly fire tour for Division 2 RPS Room C213 and Division 2 Battery Charger Room C224 per FPP-1.7 - ensure no unnecessary fire hazards and no obstructed fire equipment in Division 2 RPS Room C213 and Division 2 Battery Charger Room C224 - confirm area detection, suppression and fire doors available for Division 2 RPS Room C213 and Division 2 Battery Charger Room C224 - prohibit hot work in Division 2 RPS Room C213 and Division 2 Battery Charger Room C224 - update plant status page with plant fire risk status and affected fire area for station awareness
RW 525 HVAC 1	Division 1 HVAC Equipment Room C507	Initiate Fire Prevention Evaluation Permit per PPM 1.3.10 to: - implement hourly fire tour for Division 1 HVAC Equipment Room C507 per FPP-1.7 - ensure no unnecessary fire hazards and no obstructed fire equipment in Division 1 HVAC Equipment Room C507 - confirm area detection, suppression and fire doors available for Division 1 HVAC Equipment Room C507 - prohibit hot work in Division 1 HVAC Equipment Room C507 - update plant status page with plant fire risk status and affected fire area for station awareness

Fire Risk	Zone Description	Risk Management Actions
RW 525 CHLR	Emergency Chiller Room C502; Communications Room C503; Instrument Shop Room C510	Initiate Fire Prevention Evaluation Permit per PPM 1.3.10 to: - implement hourly fire tour for Emergency Chiller Room C502; Communications Room C503 and Instrument Shop Room C510 per FPP-1.7 - ensure no unnecessary fire hazards and no obstructed fire equipment in Emergency Chiller Room C502; Communications Room C503 and Instrument Shop Room C510 - confirm area detection, suppression and fire doors available for Emergency Chiller Room C502; Communications Room C503 and Instrument Shop Room C510 - prohibit hot work in Emergency Chiller Room C502; Communications Room C503 and Instrument Shop Room C510 - update plant status page with plant fire risk status and affected fire area for station awareness
TG 441 W	Condenser Bay West End T105; H2 Seal Oil Room T117; TG 441 West General Area T106	Initiate Fire Protection Evaluation Permit per PPM 1.3.10 to: - implement hourly fire tour for H2 Seal Oil Room T117 and TG 441 west general area T106 per FPP-1.7 - ensure no unnecessary fire hazards and no obstructed fire equipment in H2 Seal Oil Room T117 and TG 441 west general area T106 - confirm area detection, suppression and fire doors available for Condenser bay west end T105; H2 seal oil room T117; and TG 441 west general area T106 - prohibit hot work in Condenser bay west end T105; H2 seal oil room T117; and TG 441 west general area T106 - update plant status page with plant fire risk status and affected fire area for station awareness

Fire Risk	Zone Description	Risk Management Actions
TG 471 Center	TG 471 Heater Bay	Initiate Fire Prevention Evaluation Permit per PPM 1.3.10 to: - confirm Operations shiftly camera tour of the TG 471 Heater Bay - ensure no unnecessary fire hazards and no obstructed fire equipment in TG 471 Heater Bay (via camera tour) - confirm area detection, suppression and fire doors available for TG 471 Heater Bay - prohibit hot work in TG 471 Heater Bay - update plant status page with plant fire risk status and affected fire area for station awareness

7. To increase the likelihood of successful operator recovery actions in response to initiating events, operator awareness of the following measures will be increased through use of briefing sheets or procedural reviews:

    - The importance of containment venting
    - The importance of RPV depressurization
    - The importance of aligning RHR to suppression pool cooling
    - The importance of connecting fire water to condensate
    - The Importance of connecting SW to RHR train B

8. Weather forecast will be monitored prior to entering the 14 day CT to ensure no hazardous weather conditions are forecasted.

3.7 KEY PRINCIPLE #5 AND TIER 3: CONFIGURATION CONTROL

Plant procedures implement Columbia’s configuration risk management control program. The basis for Columbia’s configuration risk management control program is Maintenance Rule Section 10 CFR 50.65(a)(4). The maintenance rule (MR) requires that licensees perform risk assessments before maintenance activities are performed on SSCs and manage the increase in risk resulting from the planned activities.

"Assessing Risk" means using an appropriate tool to evaluate the temporary and aggregate risk increases generated by the maintenance activities. The assessment tool used at Columbia is PARAGON®. In addition, the Shift Technical Adviser (STA) or on-shift senior reactor operator (SRO) can also perform qualitative risk assessments for unusual external conditions.

"Managing Risk" means using the result of the risk assessment during plant decision-making, this controls overall risk impact. This is accomplished through carefully planning and scheduling maintenance activities, and implementing additional actions beyond routine work controls to address situations where the risk increase is above the established threshold.

PARAGON® software program provides the infrastructure for modeling nuclear plant safety functions, transient functions, as well as PSA and is used to assess plant configuration risk conditions. The at-power model uses a blended method, i.e., both qualitative and quantitative approaches. It has the capability to perform evaluations for combinations of structures, systems, and components (SSC) being out-of-service and human reliability events in an efficient manner.

Qualitative considerations evaluate defense-in-depth in safety functions and potential maintenance activities that may lead to a plant transient or an initiating event. The safety functions and transient functions are listed below.

The Columbia at-power safety functions are:

- Reactivity Control
- RPV Overpressure Control
- High Pressure Inventory Control
- RPV Depressurization
- Low Pressure Inventory Control
- Reactor/Containment Heat Removal
- Primary Containment Control
- Secondary Containment Control
- 4160 AC Power Supply
- Fuel Pool Cooling

The Columbia at-power transient functions (TF) are:

- Plant Transient/Manual Scram
- Loss of Condenser/Loss of Feedwater
- Loss of Off-Site Power
- Anticipated Transient Without Scram (ATWS)

Quantitative considerations use the internal events PSA model to calculate the risk increase for the maintenance configuration.

Plant Risk Level is based on the results of the qualitative and quantitative assessments of the maintenance configuration. The Columbia risk levels are designated as defined below:

- Green – Normal work controls

  - Maximum defense-in-depth
  - No evident plant transient
  - Minimal increase in core damage risk

- Yellow –Increased risk awareness

  - Reduced defense-in-depth
  - Plant transient is evident but a full complement of mitigation capability is available, or plant transient mitigating capability is reduced but no related plant transient is evident.
  - Acceptable increase in core damage risk

- Orange – Risk management actions required

  - Marginal defense-in-depth,
  - Plant transient is evident and plant transient mitigating capability is degraded, or plant transient mitigating capability is significantly degraded but no related plant transient is evident.
  - Significant increase in core damage risk

- Red – Not voluntarily entered

  - No defense-in-depth
  - Plant transient is evident and plant transient mitigating capability is degraded.
  - Unacceptable increase in core damage risk

Columbia also addresses the amended Maintenance Rule Section 10 CFR 50.65(a)(4) for added consideration of plant fire risk. Plant level fire risk is based on qualitative assessments of the maintenance configuration. The plant fire risk level is designated in the colors as defined below:

- Green – Normal work controls

  - Work with no plant fire risk
  - Work with plant fire risk increase with a duration less than 48 hours

- Blue – Risk management actions required

  - Work with plant fire risk increase with a duration greater than 48 hours

3.8 TECHNICAL ANALYSIS CONCLUSION

The five key principles involved in implementing risk-informed decision-making for assessing TS changes, as provided in RG 1.177 and RG 1.200 (References 2 and 3 respectively), have been demonstrated by this analysis. Specifically:

1. The proposed change meets the current regulations.

   This LAR itself does not propose to deviate from existing regulatory requirements, and compliance with existing regulations is maintained by the proposed one time change to the plant's TS requirements.

2. The proposed change is consistent with the defense-in-depth philosophy.

   For the RHR train A subsystem outage during the extended CT, the redundant subsystems of RHR LPCI mode of operation and the ECCS diverse subsystems HPCS and LPCS as well as the redundant SPC and CSC subsystems will continue to be capable of performing the necessary assumed safety function consistent with Columbia FSAR Chapter 15 accident analysis assumption. Additionally, Columbia's normal administrative controls will be used during the extended completion time which includes protecting opposite train systems and support systems

3. The proposed change maintains sufficient safety margins.

   While in the proposed configuration, safety analysis acceptance criteria in the FSAR are met, assuming no additional failures.

4. The proposed change for the one-time CT extension quantitative results for ICCDP and ICLERP application are less than the guidance thresholds as shown in Table 2 and is consistent with the intent of the Commission's Safety Goal Policy Statement.

   A risk evaluation was performed that considers the impact of the proposed change with respect to the risks due to internal events, internal fires, seismic events and other external hazards. The evaluation of the quantitative impacts of internal event risks during the planned CT extension demonstrates that the impact on the likelihood of core damage and large early release is well below the risk acceptance guideline. The fire and seismic evaluation determined that the impact on the likelihood of fire and seismic related core damage is also below the risk acceptance guideline. In addition, recommended actions have been identified that further reduce the risk of the significant fire scenarios. The risk associated with other external hazards is negligible.

   None of the uncertainties identified in the evaluation were found to be key uncertainties that influence this RHR CT extension, based on the guidance provided

by EPRI 1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic risk Assessments," (Reference 4).

5. The impact of the proposed change will be monitored using Columbia's Maintenance Rule a(4) program.

   Columbia's configuration risk management program will effectively monitor the risk of emergent conditions during the period the proposed extended CT is in effect. This will ensure any additional risk increase due to emergent conditions is appropriately managed.

Risk management actions will be implemented to reduce risk as described in Section 3.6.

4.0 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS / CRITERIA

The following GDCs for the ECCS were evaluated to determine if these GDC continue to be met.

GDC 35 – Emergency Core Cooling (Criterion 35)

> *A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.*
>
> *Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure*

The ECCS are designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the reactor coolant pressure boundary (RCPB) including a complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel.

The ECCS consists of the HPCS subsystem, ADS subsystem, LPCS subsystem, and the RHR LPCI subsystem mode of operation.

The LPCI subsystem mode of RHR operation continues to provide RPV flooding with 2 of the 3 subsystems available during the proposed TS CT extension. The proposed change only affects the TS CT and does not involve a change to the design or method of operation of the ECCS. Conformance to the GDC criteria is not altered by the proposed change.

GDC 36—Inspection of emergency core cooling system (Criterion 36)

> *The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.*

The proposed CT extension does not alter any ECCS equipment. Therefore, conformance to the GDC criteria is not altered by the proposed change.

GDC 37—Testing of emergency core cooling system (Criterion 37)

> *The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.*

The proposed CT extension does not alter any ECCS equipment. Therefore, conformance to the GDC criteria is not altered by the proposed change.

4.2 PRECEDENT

A one-time 7 day extension to Columbia TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3 completion times (ML15031A002) was approved for use at Columbia under license amendment number 230 (ML15030A501). This precedent extended the completion time for 7 days to allow for restoration of the B train of RHR after completion of phase 2 fuel pool cooling assist modification. This precedent is similar in that it extended the same TS actions. The precedent used the same approach for assuring the proposed change is acceptable however, this precedent used the zero-maintenance PSA model. This license amendment request uses the average-maintenance PSA model. The precedent has similar compensatory measures. The precedent also affected a single train of RHR. The basis for license amendment 230 concluded that when the station is on-line, more ECCS systems and/or subsystems are available to mitigate a design basis accident such as HPCS, LPCS, LPCI mode of RHR and the ADS system. It also

concluded that a mode transition from operating to shutdown brings inherent risk and only one RHR train would be available for shutdown cooling. The risk exposure is reduced by staying online.

4.3 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed change will revise the completion time (CT) for Technical Specification (TS) condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A by adding a footnote for restoring residual heat removal (RHR) train A to each of the required actions to allow a one-time 7 day extension (14 day completion time). The footnote will state:

> (1) The Completion Time that one train of RHR (RHR-A) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-A following the modification activity governed by EC 14635. This footnote will expire at 23:59 PST February 28, 2019.

The proposed change will also delete the following footnote which is no longer applicable from TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A:

> *(1) The Completion Time that one train of RHR (RHR-B) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-B from the modification activity. Upon successful restoration of RHR-B, this footnote is no longer applicable and will expire at 05:00 PST on February 9, 2015.*

Energy Northwest has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of Amendment,” as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   The proposed amendment does not increase the probability of an accident because the residual heat removal (RHR) system cannot initiate an accident. The RHR system provides coolant injection to the reactor core, cooling of the suppression pool water inventory, and drywell sprays following a design basis accident.

   The proposed one time completion time (CT) change for RHR train A does not alter the conditions, operating configurations, or minimum amount of operating equipment assumed in the safety analysis for accident mitigation. No changes are proposed in the manner in which the emergency core cooling system (ECCS) provides plant protection or which create new modes of plant operation. In addition, a probabilistic safety assessment (PSA) evaluation concluded that the risk contribution of the

increased CT is a very small increase in risk. The proposed change in CT will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?**

   Response: No.

   The proposed amendment will not create the possibility of a new or different kind of accident because inoperability of one RHR subsystem is not an accident precursor. There are no hardware changes nor are there any changes in the method by which any plant system performs a safety function. This request does not affect the normal method of plant operation. The proposed amendment does not introduce new equipment, or new way of operation of the system which could create a new or different kind of accident. No new external threats, release pathways, or equipment failure modes are created. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this request.

   Therefore, the implementation of the proposed amendment will not create a possibility for an accident of a new or different type than those previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

   Response: No.

   Columbia's ECCS is designed with sufficient redundancy such that a low pressure ECCS subsystem may be removed from service for maintenance or testing. The remaining subsystems are capable of providing water and removing heat loads to satisfy the final safety analysis report (FSAR) requirements for accident mitigation or plant shutdown. A PSA evaluation concluded that the risk contribution of the CT extension is very small. There will be no change to the manner in which safety limits or limiting safety system settings are determined nor will there be any change to those plant systems necessary to assure the accomplishment of protection functions. There will be no change to post-LOCA peak clad temperatures.

   For these reasons, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, Energy Northwest concludes that the proposed amendment presents no significant hazards considerations under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.4 CONCLUSIONS

The proposed amendment will provide a one-time change to TS condition 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A completion times, to allow RHR train A to be inoperable for up to 14 days. Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be adverse to the common defense and security or the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

Title 10 of CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operations of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. Energy Northwest has evaluated the proposed change and determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessments needs to be prepared in connection with the issuance of the amendment. The basis for this determination, using the above criteria, follows:

Basis

As demonstrated in the “No Significant Hazards Consideration” evaluation, the proposed amendment does not involve a significant hazards consideration. There is no significant change in the operational transients. The proposed change does not involve any physical alteration of the plant. No new or different type of equipment will be installed. The proposed change does not involve any change in methods governing normal plant operation. There is no significant increase in individual or cumulative occupational radiation exposure.

6.0 REFERENCES

1. Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk- Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, dated May 2011

2. RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications," Revision 1, dated May 2011

3. RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 2, dated March 2009

4. EPRI 1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic risk Assessments,” dated December 2008

Technical Specification Markups

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

--NOTE--

LCO 3.0.4.b is not applicable to HPCS.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days[(1)]
B High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC System is required to be OPERABLE.	Immediately
	AND	
	B.2 Restore HPCS System to OPERABLE status.	14 days

[(1)] ~~The Completion Time that one train of RHR (RHR-B) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-B from the modification activity. Upon successful restoration of RHR-B, this footnote is no longer applicable and will expire at 05:00 PST on February 9, 2015.~~ The Completion Time that one train of RHR (RHR-A) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7

days to support restoration of RHR-A following pump and motor replacement. This footnote will expire at 23:59 PST February 28, 2019.

3.6 CONTAINMENT SYSTEMS

3.6.1.5 Residual Heat Removal (RHR) Drywell Spray

LCO 3.6.1.5 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1	Restore RHR drywell spray subsystem to OPERABLE status.	7 days[1]
B. Two RHR drywell spray subsystems inoperable.	B.1	Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	--------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------------- Be in MODE 3.	12 hours

[1] ~~The Completion Time that one train of RHR (RHR-B) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-B from the modification activity. Upon successful restoration of RHR-B, this footnote is no longer applicable and will expire at 05:00 PST on February 9, 2015.~~ The Completion Time that one train of RHR (RHR-A) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-A following pump and motor replacement. This footnote will expire at 23:59 PST February 28, 2019.

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days[(1)]
B. Required Action and associated Completion Time of Condition A not met.	B.1 ---------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------------- Be in MODE 3.	12 hours
C. Two RHR suppression pool cooling subsystems inoperable.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

[(1)] ~~The Completion Time that one train of RHR (RHR-B) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-B from the modification activity. Upon successful restoration of RHR-B, this footnote is no longer applicable and will expire at 05:00 PST on February 9, 2015.~~ The Completion Time that one train of RHR (RHR-A) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-A following pump and motor replacement. This footnote will expire at 23:59 PST February 28, 2019.

Clean Technical Specification Pages

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to HPCS.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days[1]
B High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC System is required to be OPERABLE.	Immediately
	<u>AND</u>	
	B.2 Restore HPCS System to OPERABLE status.	14 days

[1] The Completion Time that one train of RHR (RHR-A) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-A following pump and motor replacement. This footnote will expire at 23:59 PST February 28, 2019.

3.6 CONTAINMENT SYSTEMS

3.6.1.5 Residual Heat Removal (RHR) Drywell Spray

LCO 3.6.1.5 Two RHR drywell spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR drywell spray subsystem inoperable.	A.1	Restore RHR drywell spray subsystem to OPERABLE status.	7 days[1]
B. Two RHR drywell spray subsystems inoperable.	B.1	Restore one RHR drywell spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	--------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------------- Be in MODE 3.	 12 hours

(1) The Completion Time that one train of RHR (RHR-A) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-A following pump and motor replacement. This footnote will expire at 23:59 PST February 28, 2019.

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days[1]
B. Required Action and associated Completion Time of Condition A not met.	B.1 ---------------NOTE-------------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------------- Be in MODE 3.	12 hours
C. Two RHR suppression pool cooling subsystems inoperable.	C.1 Be in MODE 3.	12 hours
	AND	
	C.2 Be in MODE 4.	36 hours

[1] The Completion Time that one train of RHR (RHR-A) can be inoperable as specified by Required Action A.1 may be extended beyond the 7 day completion time up to 7 days to support restoration of RHR-A following pump and motor replacement. This footnote will expire at 23:59 PST February 28, 2019.

Summary of Regulatory Commitments

List of Regulatory Commitments

The following table identifies the regulatory commitments in this document. Any other statements in this submittal regarding intended or planned actions, are provided for information purposes, and are not considered to be regulatory commitments.

Commitment	Type Scheduled		Completion Date
	One-Time	Continuing Compliance	
Compensatory Measures outlined in section 3.6 of Attachment 1 of this letter will be implemented during the period of the proposed completion time	X		Upon implementation of the one-time completion time extension supporting RHR Train A pump and motor replacement.

Probabilistic Risk Assessment (PRA)

RHR Allowed Outage Time License Amendment Request Risk-Informed Evaluation

Preparer 10/25/16
Date

Reviewer 10/25/16
Date

Approver 10/25/16
Date

Table of Contents

1. Introduction

1.0 PURPOSE

This evaluation provides the calculations and assessment to support a technical specification (TS) completion time (CT) extension from seven days to 14 days for the following limiting conditions for operation (LCOs):

- LCO 3.5.1, Condition A for low pressure ECCS injection/spray subsystems
- LCO 3.6.1.5, Condition A for Residual Heat Removal (RHR) drywell spray subsystems
- LCO 3.6.2.3, Condition A for RHR suppression pool cooling subsystems.

1.1 BACKGROUND

1.1.1 <u>Technical Specification Changes</u>

Since the mid-1980s, the NRC has been reviewing and granting improvements to TS that are based, at least in part, on probabilistic risk assessment (PRA) insights. In its final policy statement on TS improvements of July 22, 1993, the NRC stated that it . . .

> *. . . expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs. . . Similarly, the NRC staff will also employ risk insights and PSAs in evaluating Technical Specifications related submittals. Further, as a part of the Commission's ongoing program of improving Technical Specifications, it will continue to consider methods to make better use of risk and reliability information for defining future generic Technical Specification requirements.*

The NRC reiterated this point when it issued the revision to 10 CFR 50.36, "Technical Specifications," in July 1995. In August 1995, the NRC adopted a final policy statement on the use of PRA methods in nuclear regulatory activities that encouraged greater use of PRA to improve safety decision-making and regulatory efficiency. The PRA policy statement included the following points:

1. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

2. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state of the art, to reduce unnecessary conservatism associated with current regulatory requirements.

3. PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

4. The Commission's safety goals and subsidiary numerical objectives are to be used with consideration of uncertainties in making regulatory judgments.

The movement of the NRC to more risk-informed regulation has led to the NRC identifying Regulatory Guides and associated processes by which licensees can submit changes to the plant design basis including Technical Specifications. Regulatory Guides 1.174 [Ref. 1] and 1.177 [Ref. 2] both provide processes to incorporate PRA input for decision makers regarding a Technical Specification modification.

1.2 REGULATORY GUIDES

Three Regulatory Guides (RG) provide primary guidance related to risk evaluations of a Technical Specification change. The relevance of these Regulatory Guides in relationship to this applicable are discussed in this section.

1.2.1 Regulatory Guide 1.200, Revision 2

Regulatory Guide 1.200, Revision 2 [Ref. 4] describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the PRA results, such that the PRA can be used in regulatory decision-making. This guidance is intended to be consistent with the NRC's PRA Policy Statement.

It is noted that RG 1.200, Revision 2 endorses Addendum A of the ASME/ANS PRA Standard [Ref. 5] as clarified in Appendix A of RG 1.200, Revision 2.

The Columbia Generating Station (CGS) full power internal events (FPIE) PRA model meets the requirements of RG 1.200 Revision 2. The fire and seismic PRA models do not meet all aspects of the RG 1.200 guidance, but the PRA models are judged to be of sufficient quality for this application and provide insights into dominant risk contributors for fire and seismic events. The technical adequacy of the CGS PRA models is further discussed in Section 5.0 of this assessment.

1.2.2 Regulatory Guide 1.174, Revision 2

Regulatory Guide 1.174 [Ref. 1] specifies an approach and acceptance guidelines for use of PRA in risk informed activities. RG 1.174 outlines PRA related acceptance guidelines for use of PRA metrics of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) for the evaluation of permanent TS changes. The guidelines given in RG 1.174 for determining what constitutes an acceptable permanent change specify that the delta (Δ) CDF and the ΔLERF associated with the change should be less than specified values, which are dependent on the baseline CDF and LERF, respectively.

RG 1.174 also specifies guidelines for consideration of external events. External events can be evaluated in either a qualitative or quantitative manner.

Since this LAR is for a one-time TS change, the ΔCDF and the ΔLERF of RG 1.1.74 do not specifically apply and are therefore, not evaluated in this assessment.

1.2.3 Regulatory Guide 1.177, Revision 1

Regulatory Guide 1.177 [Ref. 2] specifies an approach and acceptance guidelines for the evaluation of plant licensing basis changes. RG 1.177 identifies a three-tiered approach for the evaluation of the risk associated with a proposed TS change as identified below:

- Tier 1 is an evaluation of the plant-specific risk associated with the proposed TS change, as shown by the change in core damage frequency (CDF) and incremental conditional core damage probability (ICCDP). Where applicable, containment performance should be evaluated on the basis of an analysis of large early release frequency (LERF) and incremental conditional large early release probability (ICLERP). The acceptance guidelines given in RG 1.177 for determining an acceptable permanent TS change is that the ICCDP and the ICLERP associated with the change should be less than 1E-06 and 1E-07, respectively. RG 1.177 also addresses risk metric requirements for one-time TS changes, as outline in Section 1.2.4 of this risk assessment.

- Tier 2 identifies and evaluates, with respect to defense-in-depth, any potential risk-significant plant equipment outage configurations associated with the proposed change. The licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed TS change is out-of-service.

- Tier 3 provides for the establishment of an overall configuration risk management program (CRMP) and confirmation that its insights are incorporated into the decision-making process before taking equipment out-of-service prior to or during the completion time. Compared with Tier 2, Tier 3 provides additional risk management actions based on any additional risk significant configurations that may be encountered during maintenance scheduling over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance, testing, and corrective and preventive maintenance.

1.2.4 Acceptance Guidelines

Risk significance of a proposed permanent change to Technical Specifications is typically determined by the comparison of changes in CDF and LERF and values of ICCDP and ICLERP and compared to the acceptance guidelines given in RG 1.174 and RG 1.177. RG 1.174 specifies the acceptable changes in CDF and LERF for permanent changes. RG 1.177 specifies the acceptable ICCDP and ICLERP for permanent changes, usually associated with changing TS completion times.

RG 1.177 also directly addresses the risk metric requirements for one-time TS changes, which is most applicable to this risk application. The discussion of acceptance guidelines for one-time TS changes from RG 1.177 are provided below:

> *"For one-time only changes to TS CTs, the frequency of entry into the CT may be known, and the configuration of the plant SSCs may be established.*

Further, there is no permanent change to the plant CDF or LERF, and hence the risk guidelines of Regulatory Guide 1.174 cannot be applied directly. The following TS acceptance guidelines specific to one-time only CT changes are provided for evaluating the risk associated with the revised CT:

1. *The licensee has demonstrated that implementation of the one-time only TS CT change impact on plant risk is acceptable (Tier 1):*
   - *ICCDP of less than $1.0x10^{-6}$ and an ICLERP of less than $1.0x10^{-7}$, or*
   - *ICCDP of less than $1.0x10^{-5}$ and an ICLERP of less than $1.0x10^{-6}$ with effective compensatory measures implemented to reduce the sources of increased risk.*

2. *The licensee has demonstrated that there are appropriate restrictions on dominant risk-significant configurations associated with the change (Tier 2).*

3. *The licensee has implemented a risk-informed plant configuration control program. The licensee has implemented procedures to utilize, maintain, and control such a program (Tier 3).”*

The quantitative criteria shown in Table 1-1 are consistent with the acceptance guidelines in RG 1.177 and are judged to represent a reasonable set of acceptance guidelines for this application. These guidelines demonstrate that the risk impacts are acceptably low. These acceptance guidelines combined with effective compensatory measures to further reduce the risk will ensure that the TS temporary change meets the intent of small risk increases consistent with the Commission's Safety Goal Policy Statement.

Table 1-1
PROPOSED RISK ACCEPTANCE GUIDELINES

RISK ACCEPTANCE GUIDELINE	BASIS
ICCDP < 1E-6, or ICCDP < 1E-5 with effective compensatory measures implemented to reduce the sources of increased risk	ICCDP is an appropriate metric for assessing risk impacts of out of service equipment per RG 1.177. This guideline is specified in Section 2.4 of RG 1.177.
ICLERP < 1E-7, or ICLERP < 1E-6 with effective compensatory measures implemented to reduce the sources of increased risk	ICLERP is an appropriate metric for assessing risk impacts of out of service equipment per RG 1.177. This guideline is specified in Section 2.4 of RG 1.177.

1.3 SCOPE

This section addresses the requirements of RG 1.200, Revision 2, Section 3.1, which directs the licensee to define the treatment of the scope of risk contributors (i.e., internal initiating events, external initiating events, and modes of power operation at the time of the initiator). Discussion of these risk contributors is provided below:

- Full Power Internal Events (FPIE) – This calculation utilizes the CGS Revision 7.2.1 integrated probabilistic risk assessment (PRA) to calculate ICCDP and ICLERP for the internal events PRA. This CT extension applies only to power operation. It is reasonable to conclude that transition risk will be lower as a result of the extension, owing to the lower frequency that the plant will exceed the allowed outage time due to the greater ability to accommodate RHR pump maintenance within the extended completion time interval. This increase in safety has not been quantified for CGS.

- Low Power Operation - The FPIE model is judged to adequately capture risk contributors associated with low power plant operations. The FPIE analysis assumes that the plant is at full power at the time of any internal events transient, manual shutdown, or accident initiating event. This analytic approach results in conservative accident progression timings and systemic success criteria compared to what may otherwise be applicable to an initiator occurring at low power. As such, low power risk impacts are not discussed further in this risk assessment.

- Shutdown / Refueling – The intent is for the unit to remain at-power for the duration of the extended CT. Therefore, there is no increase in risk for shutdown condition. Additionally, it is reasonable to conclude that performing RHR overhauls on-line rather than during outages will increase RHR availability during outages. This should reduce shutdown risk by improving the availability of decay heat removal during shutdown. This increase in safety has not been quantified for CGS. Shutdown risk is not addressed further in this risk evaluation.

- Internal Fires – CGS has a fire PRA model, which was peer reviewed in 2004 and is currently undergoing a significant update. The current Fire PRA model [Ref. 13] is used to provide both quantitative and qualitative insights to the analysis of the RHR A CT extension.

- Seismic – CGS has a seismic PRA model, which provides quantitative results and is used in this risk assessment. Qualitative risk inputs are also provided for this analysis as well.

- Other External Events - Evaluation of high winds, external floods, volcanic eruption and other external events in the IPEEE per GL 88-20 were submitted to and reviewed by the NRC. The Staff Evaluation Report (Ref. 12) concluded that CGS IPEEE was capable of identifying the most likely severe accidents and severe accident vulnerabilities and the IPEEE met the intent of Supplement 4 to GL 88-20. The IPEEE determined that the recurrence frequency for the maximum tornado wind speed is approximately 1E-07/yr

and, as such, maximum wind speed was eliminated as a plant hazard per the Standard Review Plan. Other external events (e.g., external fire, external floods, high winds, volcanic eruption etc.) were considered to be insignificant contributors to severe accidents. The proposed changes are judged to have a negligible effect on the risk profiles from other external events.

2. Technical Approach

The integrated CGS PRA model of record, Revision 7.2.1, for internal events, fire and seismic was used to perform the calculations. Revision 7.2.1 was completed in 2016. The following subsections discuss the technical approach used to perform the calculations.

2.1 Residual Heat Removal (RHR) CT Extension Scope

To support the RHR CT extension, this evaluation calculates ICCDP and ICLERP for the internal events, seismic, and fire PRA for RHR A (the low pressure coolant injection [LPCI], suppression pool cooling and drywell spray modes of operation), for an AOT extension from 7 days to 14 days. These values of ICCDP and ICLERP are then compared to the acceptance criteria given in Table 1-1.

2.2 Modeling Notes and Assumptions

The following assumptions are made as part of this calculation:

1. This calculation evaluates the change to plant risk for the CT extension for the preventive maintenance case when an RHR A train is electively removed from service. As directed by PPM 1.3.83 Rev 21, “PROTECTED EQUIPMENT PROGRAM” [Ref. 9], when RHR train A is inoperable, RHR train B, diesel generator (DG) 2 and standby service water (SW) train B are protected. To ensure safety margins and defensive in depth for the RHR train A CT extension, the compensatory measures mentioned in the procedure above (protection of RHR train B, DG2 and SW train B) are assumed for this evaluation in the base case. A quantification is also performed without credit for the assumed protected equipment for comparison purposes.

2. This calculation uses equipment mean outage times (average maintenance unavailability). This calculation does not use the zero maintenance states.

3. By the use of house event flags, the RHR train A, which is out of service, is set to be unavailable (logical TRUE). Additionally, in order to fail the drywell spray mode of the RHR A train that is out of service, the PRA quantification sets to TRUE the pump failure-to-start basic event of the RHR train that is out of service (OOS).

4. The PRA model is quantified at a truncation limit of 2E-12. This is consistent with the base quantification of the PRA model and a truncation study performed and documented in the Quantification notebook [Ref.6]. Since the model is re-quantified as part of this evaluation (as opposed to using pre-solved cutsets) and the truncation study has shown the model converged at this truncation level, it is concluded that the truncation level is appropriate for this application.

2.3 Core Damage Frequency Calculations

2.3.1 Average Maintenance Model

The core damage frequency calculations employed the average maintenance model. The average maintenance is solved with all maintenance terms set to the base case average unavailability, except for the maintenance unavailability of protected trains, which are set to zero.

The maintenance unavailability for protected trains is set by flag files as follows:

- RHR A out of service: Flag file settings are:

  RHR----B----T3LL (RHR train B unavailable due to maintenance) EQU .F.
  EACEDG-2----T3D2 (DG-2 unavailable due to maintenance) EQU .F.
  SW-----B----T3LL (SW train B unavailable due to maintenance) EQU .F.

2.3.2 Treatment of Common Cause Failures.

The PRA evaluations examine preventive maintenance cases (i.e., train is electively taken out of service). The following subsections describe the methodology used to appropriately adjust common cause failure unavailability. The Tables A-1 through A-3 document the development in detail.

2.3.2.1 Preventive Maintenance Case

For preventive maintenance, RHR-A is electively removed from service. Common cause terms that involve the train in question are not applicable when the train is removed from service, and are set to zero, as the train removed from service is operable prior to removal from service [Ref. 10]. The common cause terms that involve the RHR pump trains that remain in service are recomputed assuming a common cause group of two, as shown in Tables A-1 and A-2.

The CCF adjustments are as follows:

- RHR A out of service: Flag file setting for the CCF values for the corrective maintenance case are:

  RHRP-MD-2ABCC3R2 EQU .F.
  RHRP-MD-2ABCC3S4 EQU .F.
  RHRP-MDABFTRC2LL EQU .F.
  RHRP-MDABFTSC2LL EQU .F.
  RHRP-MDBCFTRC2LL PROB 5.3E-6
  RHRP-MDBCFTSC2LL PROB 1.94E-5
  RHRP-MDACFTRC2LL EQU .F.
  RHRP-MDACFTSC2LL EQU .F.

2.4 LERF Calculations

The LERF and ICLERP were computed using the Rev. 7.2.1 PRA LERF modeling, in a manner similar to the quantifications described above for CDF.

2.5 Computation Approach

Results for each PRA quantification were placed in an Excel file for calculation of ICCDP ICLERP. The computation results are summarized in Table A-10 of Attachment A.

The following equations were used for the computations [Ref. 2]:

The ICCDP and ICLERP are computed using their definitions in RG 1.177. The formulas are as follows:

$$ICCDP(YOOS) = (CDF_{YOOS} - CDF_{BASELINE})*\Delta T$$

Where:

ICCDP(YOOS) is the ICCDP with train Y out of service,
CDF_{YOOS} is the CDF computed with train Y out of service,
$CDF_{BASELINE}$ is the baseline, average-maintenance case CDF, and
ΔT is the extension of the CT converted to units consistent with the CDF frequency units (14 days * 1 yr / 365 days = 3.84E-2 yr).

Similarly, ICLERP is computed as follows:

$$ICLERP(YOOS) = (LERF_{YOOS} - LERF_{BASELINE}) * 3.84E\text{-}2 \text{ yr}$$

Where:

ICLERP(YOOS) is the ICLERP with train Y out of service,
$LERF_{YOOS}$ is the LERF computed with train Y out of service, and
$LERF_{BASELINE}$ is the LERF baseline, average-maintenance case LERF.

3. Results

The ICCDP and ICLERP results are presented in Tables 3-1 and 3-2.

Table 3-1 represents the average maintenance results with RHR A out of service (OOS) without protected trains as required by PPM 1.3.83 (i.e., no compensatory measures).

Table 3-2, represents the base case average maintenance results with protected trains as directed by PPM 1.3.83.

The ICCDP and ICLERP were compared with guidance thresholds of less than or equal to an ICCDP of 1E-6 and an ICLERP of 1E-7 [Ref. 2]. All results for the RHR A CT extension application base case were less than the guidance thresholds. The results include significant margin to the maximum allowed changes (1E-5 ICCDP and 1E-6 ICLERP) while still having the total CDF < 1E-4/year and LERF < 1E-5/year. There is margin to accommodate additional risk due to non-RG 1.200 fire and seismic PRA models and still be within acceptable limits.

Table 3-1 14-DAY RHR CT EXTENSION RISK RESULTS Average Maintenance Model without protected Trains per PPM 1.3.83		
Risk Metric	**Acceptance Guideline**	**PRA Results**
RHR-A_OOS - ICCDP (Total)	< 1.0E-6	8.32E-7
RHR-A_OOS- ICCDP (Internal Events)		3.52E-7
RHR-A_OOS- ICCDP (Fire)		3.04E-7
RHR-A_OOS- ICCDP (Seismic)		1.75E-7
RHR-A_OOS - ICLERP (Total)	< 1.0E-7	1.23E-9
RHR-A_OOS - ICLERP (Internal Events)		1.15E-10
RHR-A_OOS - ICLERP (Fire)		1.11E-9
RHR-A_OOS - ICLERP (Seismic)		Negligible

Table 3-2 14-DAY EXTENSION PREVENTIVE MAINTENANCE Average Maintenance Model with protected Trains per PPM 1.3.83		
Risk Metric	**Acceptance Guideline**	**PRA Results**
RHR-A_OOS - ICCDP (Total)	< 1.0E-6	6.25E-7
RHR-A_OOS- ICCDP (Internal Events)		2.10E-7
RHR-A_OOS- ICCDP (Fire)		2.73E-7
RHR-A_OOS- ICCDP (Seismic)		1.41E-7
RHR-A_OOS - ICLERP (Total)	< 1.0E-7	6.90E-10
RHR-A_OOS - ICLERP (Internal Events)		7.67E-11
RHR-A_OOS - ICLERP (Fire)		6.14E-10
RHR-A_OOS - ICLERP (Seismic)		Negligible

3.1 Dominant Risk Contributors and Insights – RG 1.177 Tier 2

As part of addressing the RG 1.177 Tier 2 guidance, dominant sequences for the RHR A CT extension are calculated. Based on these results, dominant risk contributors and insights are discussed in the following subsections. These insights are found to be reasonable for the RHR A CT extension application, and support demonstrating the validity of the PRA for assessing the proposed TS changes. LERF was not addressed in this section because the ICLERP results are very small (< 1E-9).

3.1.1 <u>Insights Derived from Accident Sequences with Significant Change in the Birnbaum Importance Measure Relative to Base Case Results</u>

This discussion focuses on risk contributors with the most significant Birnbaum increases from the baseline model as a result of the assumed CT extension. These cutsets were identified by comparing the base model sequence results to the sequence results for the PRA quantification

assuming RHR A out of service. These sequences were then sorted by the largest change in the Birmbaum importance measure.

The cutsets below are the top cutsets for the corresponding sequences.

Internal Events Accident Sequences:

FPIE Cutset #1:

INIT-RY-TDC2 :REACTOR YEAR CONVRSN - LOSS OF E-DP-S1/2
EACENG-EDG3-S424 :EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN 24H
EACTRL-S----T3-- :TRANSFORMER TR-S OUT FOR MAINTENANCE (MRULE DATA)
EDCC1--2A2B-C3LL :COMMON CAUSE FAIL. OF BATTERY CHARGERS C1-2A AND C1-2B
-L1-SUCC-U2 :RCIC SUCCESS TERM
REC-L1-TDC2010 :Sequence Tag REC-L1-TDC2010

The above cutset belongs to sequence TDC2010**.** This sequence represents loss of DC Div-2 initiating event due to Common Cause failure of battery chargers C1-2A and C1-2B. Reactor Feed Water (RFW) failed due to failure of startup transformer (TR-S). High Pressure Core Spray (HPCS) failed due to random failure of Diesel Generator (DG)3. Reactor Core Isolation Cooling (RCIC) is available at the beginning of the sequence, but failed because suppression pool cooling is not available. Suppression Pool Cooling (SPC) is unavailable because RHR B failed due to failure of chargers to C1-2A and C1-2B and because RHR A is in maintenance. Sequence REC-L1-TDC2010's Birnbaum value has increased 820 times from its baseline value.

FPIE Cutset #2:

IE-FLD-TLO--RFWM :LARGE RFW LEAKS IN TURB BLDG - LOCA OUTSIDE CONTNMENT
EACENG-EDG3-S424:EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN 24 hrs
PRAAHUS--1B-S3LL :FAN PRA-FN-1B DOES NOT START ON DEMAND
RCI---------T3LL :RCIC UNAVAILABILITY DUE TO T & M (MRULE DATA)
REC-L1-FLMRF009 :Sequence Tag REC-L1-FLMRF009

The above cutset belongs to sequence FLMRF009. This sequence represents moderate breaks in RFW which results in the RFW initiating event. The break is assumed to damage all equipment in the turbine building. HPCS failed due to random failure of DG3. RCIC fails because it is out of service for maintenance. Depressurization and low pressure injection (Low Pressure Core Spray (LPCS) available) injection are successful. However, heat removal is not available. RHR B fails due to the random failure of SW-B room cooling and RHR A is OOS for maintenance. Containment vent fails because Control Air System (CAS) is unavailable due to RFW flood in the turbine building. Core damage occurs due to containment failure. Sequence REC-L1- FLMRF009's Birnbaum value has increased 313 times from its baseline value.

FPIE Cutset #3:

IE-FLD-RLO-RCICS :SMALL RCIC LEAK IN REACTOR BLDG - LOCA OUTSIDE containment
HPS---------T3LL : HPCS UNAVAILABILITY DUE TO T & M (MRULE DATA)
PRAAHUS--1B-S3LL: FAN PRA-FN-1B DOES NOT START ON DEMAND
REC-L1-FLSRC007 : Sequence Tag REC-L1-FLSRC007

The above cutset belongs to sequence FLSRC007. This sequence represents a small RCIC steam break loss of coolant accident (LOCA) outside containment initiating event. HPCS is unavailable because it is OOS for maintenance. RCIC is unavailable due to the break. Depressurization and low pressure injection are successful (LPCS available). However, heat removal is unavailable. RHR B fails due to a random failure of SW B room cooling and RHR A OOS for maintenance. It is conservatively assumed that containment vent failure occurs due to the flood caused by the RCIC line break. Core damage occurs due to containment failure. Sequence REC-L1- FLSRC007's Birnbaum value has increased 213 times from its baseline value.

Internal Fire Accident Sequences:

The following Fire event PRA contributors were found to be dominant sequences that have increased when RHR A is removed from service for preventive maintenance, relative to the base case results.

Fire Cutset #1:

LZT12 : TG-12 Full-PAU Burnup - Initiator
-L1-SUCC-U2 : RCIC SUCCESS TERM
REC-L1-T(E)N025 : Sequence Tag REC-L1-T(E)N025

The above cutset belongs to sequence T(E)N025. The above sequence represents a fire event in the turbine building corridor, TG-12. Fire damage produces a loss of offsite power, RHR B, RHR C and HPCS, SPC is unavailable (due to RHR A in maintenance). RCIC successfully provides injection, but long term operation of RCIC fails due to unavailability of SPC. Operators can depressurize and maintain level with LPCS, but eventually LPCS will fail because containment fails due to overpressurization. It is conservatively assumed that operators will not be able to vent containment during this sequence and operators cannot use firewater after containment failure. Sequence REC-L1-T(E)N025's Birnbaum value has increased 92 times from its baseline value.

Fire Cutset #2:

LZT12 : TG-12 Full-PAU Burnup - Initiator
PTM : FAILURE OF SRV'S RECLOSING FOR TM,TC & LOOP INITIATORS
SGTV-AO2A---W3LL : AIR OPERATED VALVE SGT-V-2A FAILS TO CLOSE
REC-L1-T(E)N081 : Sequence Tag REC-L1-T(E)N081

The above cutset belongs to sequence T(E)N081**.** The above sequence represents a fire event in the turbine building corridor, TG-12. Fire damage produces a loss of offsite power, RHR B, RHR C and HPCS. SRV is stuck open (PTM), but depressurization succeeds. SPC is unavailable (due to RHR A in maintenance). RCIC failed due to the stuck open relief valve. Containment vent fails due to the random failure of the Standby Gas Treatment (SGT) valve. Core damage occurs due to containment failure. Sequence REC-L1-T(E)N081's Birnbaum value has increased 23 times from its baseline value.

Fire Cutset #3:

T3W07 : RC-7 Detailed Scenario 3 - Initiator
EACENG-EDG3-S424 : EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN 24H
RCI---------T3LL : RCIC UNAVAILABILITY DUE TO T & M (MRULE DATA)
SGTV-AO2A---W3LL: AIR OPERATED VALVE SGT-V-2A FAILS TO CLOSE
REC-L1-TT024 : Sequence Tag REC-L1-TT024

The above cutset belongs to sequence (TT024). The above sequence represents a fire event in the division 2 electrical equipment room, RC07. Fire damage produces a turbine trip, and feedwater (FW), power conversion system (PCS), RHR B and C and control rod drive (CRD) are unavailable due to fire damage. RCIC is out of service for maintenance. HPCS is unavailable due to DG3 random failure. Depressurization and low pressure injection are successful, but SPC is unavailable due to RHR A in maintenance and containment vent fails due to random failure of SGT. Core damage occurs due to containment failure and failure of fire water and SW cross tie due to fire. Sequence REC-L1-TT024's Birnbaum value has increased 11 times from its baseline value.

Seismic Accident Sequences:

The following seismic events PRA contributors were found to be the dominant sequences that have increased when train RHR A is removed from service for preventive maintenance relative to the base case results.

Seismic Cutset #1:

SDS14 : SDS 26 BOP CST LOOP N2 Tank SLOCA - Initiator
EACENG-EDG2SS4D2 : EMERGENCY DG-2 DOES NOT CONTINUE TO RUN FOR 6 HRS
NRAC24H-SEIS: No Recovery of Offsite AC w/in 24 hrs. (Seismic)
REC-L1-SDS14-22: Sequence Tag REC-L1-SDS14-22

The above cutset belongs to sequence SDS14-22 and represents a seismic event that results in seismic damage state 14, which is a loss of power conversion system (PCS), the condensate storage tank, offsite power and a small LOCA. RHR B is unavailable due to the unavailability of the division 2 DG (EACENG-EDG2SS4D2). For this particular sequence, high pressure injection fails, but low pressure injection succeeds. Core damage occurs as a result of the unavailability of heat removal systems (i.e., SPC and containment vent). Sequence REC-L1-SDS14-22's Birnbaum value has increased 157 times from its baseline value.

Seismic Cutset #2:

SDS26 : SDS 26 BOP CST LOOP N2 Tank SLOCA - Initiator
EACENG-EDG2SS4D2 : EMERGENCY DG-2 DOES NOT CONTINUE TO RUN FOR 6 HRS
NRAC24H-SEIS: No Recovery of Offsite AC w/in 24 hrs. (Seismic)
REC-L1-SDS26-49 : Sequence Tag REC-L1-SDS26-49

The above cutset belongs to sequence SDS26-49 and represents a seismic event that results in seismic damage state 26, which is a loss of PCS, the condensate storage tank, offsite power, and a small LOCA. RHR B is unavailable due to the unavailability of the division 2 DG (EACENG-EDG2SS4D2). The division 1 and 3 DGs fail to initially start, due to release of nitrogen in their vicinity, but are recovered. For this particular sequence, high pressure injection

fails, but low pressure injection succeeds. Core damage occurs as a result of the unavailability of heat removal systems. Sequence REC-L1-SDS26-49's Birnbaum value has increased 157 times from its baseline value.

Seismic Cutset #3:

SDS8 : SDS 8 BOP CST LOOP SLOCA - Initiator
EACENG-EDG2SS4D2 : EMERGENCY DG-2 DOES NOT CONTINUE TO RUN FOR 6 HRS
NRAC24H-SEIS: No Recovery of Offsite AC w/in 24 hrs. (Seismic)
REC-L1-SDS8-49: Sequence Tag REC-L1-SDS8-49

The above cutset belongs to sequence REC-L1-SDS8-49 and represents a seismic event that results in seismic damage state 8, which is a loss of PCS, the condensate storage tank, offsite power and a small LOCA. RHR B is unavailable due to the unavailability of the division 2 DG (EACENG-EDG2SS4D2). For this particular sequence, high pressure injection fails, but low pressure injection succeeds. Core damage occurs as a result of the unavailability of heat removal systems. Sequence REC-L1-SDS8-49's Birnbaum value has increased 142 times from its baseline value.
Based on the comparison of the above dominant sequences for Internal, Fire and Seismic events relative to base case results, the following insights were derived:

- The Birnbaum value relative to their baseline contribution has increased because RHR-A is removed from service for preventive maintenance.
- This insight is found to be reasonable for the RHR-A CT extension application, and supports the validity of the PRA for assessing the proposed TS changes, especially for fire and seismic.
- Based on the results of the first FPIE cutset, the DC Div 2 battery charger room will be protected.

3.1.2 Insights derived from the most Significant Accident Sequence Contributions.

This discussion focuses on the risk contributors that are most significant when a train of RHR-A is removed from service. The sequences with the top five Birnbaum measures are discussed below.

Internal Events

The following internal events sequences were found to be the dominant contributors when a train of RHR-A is removed from service for preventive maintenance.

FPIE Cutset #1:

IE-FLD-T106CONDS : SMALL COND BREAK IN T106
HPS---------T3LL: HPCS UNAVAILABILITY DUE TO T & M (MRULE DATA)
-L1-SUCC-U2 : RCIC SUCCESS TERM
PRAAHUS--1B-S3LL: FAN PRA-FN-1B DOES NOT START ON DEMAND
REC-L1-FLTSW006 : Sequence Tag REC-L1-FLTSW006

The above cutset belongs to sequence FLTSW006, and involves an internal flood that causes loss of plant service water. HPCS is unavailable due to maintenance, but RCIC is available for

injection. Core damage occurs due to the unavailability of heat removal systems (i.e., SPC and containment venting). RHR B fails due to the unavailability of SW B room cooling and RHR A is out of service. Containment venting fails due to flood damage.

FPIE Cutset #2:

IE-FLD-TLO--MS-S : SMALL MS LEAKS IN TURB BLDG - LOCA OUTSIDE CONTNMENT
EACENG-EDG3-S424: EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN 24H
PRAAHUS--1B-S3LL : FAN PRA-FN-1B DOES NOT START ON DEMAND
REC-L1-FLSMS007: Sequence Tag REC-L1-FLSMS007

The above cutset belongs to sequence FLMS007, and involves a small main steam leak outside containment. High pressure injection is unavailable due to DG3 random failures and RCIC is assumed unavailable due to the steam leak. Reactor Pressure Vessel (RPV) depressurization and low pressure injection succeed. Core damage occurs due to the unavailability of containment heat removal. RHR B failed due to the random failure of SW B room cooling and RHR-A is out for maintenance. Containment venting failed due to the flood.

FPIE Cutset #3:

IE-FLD-TLO--MS-U: MOD MS LEAKS IN TURB BLDG - LOCA OUTSIDE CONTNMENT
EACENG-EDG3-S424: EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN 24H
PRAAHUS--1B-S3LL: FAN PRA-FN-1B DOES NOT START ON DEMAND
REC-L1-FLUMS007: Sequence Tag REC-L1-FLUMS007

The above cutset belongs to sequence FLUMS007, and involves a moderate main steam leak outside containment that causes an initiating event. High pressure injection is unavailable due to DG3 random failures and RCIC is assumed unavailable due to the steam leak. RPV depressurization and low pressure injection succeed. Core damage occurs due to the unavailability of containment heat removal. RHR B failed due to the random failure of SW B room cooling and RHR-A is out for maintenance. Containment venting failed due to the flood. This sequence is similar to the above sequence.

FPIE Cutset #4:

IE-TF : LOSS OF FEEDWATER FREQUENCY IN EVENTS PER REACTOR YEAR
CF-FAILS-HPCS-INJ : HPCS INJECTION FAILS DUE TO CONTAINMENT FAILURE
RHRHUMNSP-COOLLL: FAILURE TO ALIGN RHR TO SUPPRESSION POOL COOLING
RHRHUMNSWCRTIEXX: OPERATOR FAILS TO CONNECT SW TO RHR-B POST CONTAINMENT FAILURE
FP-HUMNSYS62H3LL: OPERATOR FAILS TO CONNECT FIREWATER TO CONDENSATE POST CONTAINMENT FAILURE
VENTFAIL: OPERATOR FAILS TO VENT CONTAINMENT
REC-L1-TF006: Sequence Tag REC-L1-TF006

The above cutset belongs to sequence TF006, and involves a loss of feedwater initiating event, with success of HPCS, but failure of containment heat removal due to human action (RHRHUMNSP-COOLLL). Containment venting is failed due to human action (VENTFAIL). Eventually containment fails, which fails HPCS (CF-FAILS-HPCS-INJ). Depressurization succeeds but alternate injection via fire water and the SW cross-tie are not available due to human actions (FP-HUMNSYS62H3LL, RHRHUMNSWCRTIEXX).

FPIE Cutset #5:

IE-FLD-TLO—RFWS: SMALL RFW LEAKS IN TURB BLDG - LOCA OUTSIDE CONTNMENT
EACENG-EDG3-S424: EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN 24H
-L1-SUCC-U2: RCIC SUCCESS TERM
PRAAHUS--1B-S3LL ; FAN PRA-FN-1B DOES NOT START ON DEMAND
REC-L1-FLSRF006: Sequence Tag REC-L1-FLSRF006

The above cutset belongs to sequence FLSRF006, and involves a small reactor feedwater leak outside containment that causes an initiating event. HPCS is unavailable due to random failure of DG3, but RCIC is available for injection. Core damage occurs due to the unavailability of containment heat removal.

For additional information on accident sequence contribution to CDF for the base case where RHR A is OOS, see Table A-11 for significant internal event accident sequences percentage contribution to CDF.

Based on the review of the above dominant sequences, the following insight are identified.

1. RCIC, HPCS, DG3 and SW-B are very important. To avoid risk-significant plant configurations, RCIC, HPCS, and DG3 will be protected. SW B will be protected per PPM 1.3.83. TR-S and HPCS-SW will be protected to support HPCS. RHR C will be protected to provide additional defense in depth for low pressure injection when RHR A is OOS for maintenance.

2. The above cutsets include initiating events from flood in the turbine building. For this reason, the plant will initiate a flood watch tour in the turbine building when RHR A is taken out of service. This measure is not credited in the PRA.

3. Operator actions to cross-tie SW with RHR and to inject with Fire water after containment failure are important. Human actions to align SPC and to vent containment are important. Successful depressurization is also important. Operator awareness briefings will be provided to increase awareness for these actions. In addition, in order to perform the containment venting, the following support systems will be protected: Standby Gas Treatment (SGT) and Control and Service Air (CAS). These measures are not credited in the PRA.

Internal Fire

Fire Cutset #1:

LZT12: TG-12 Full-PAU Burnup - Initiator
-L1-SUCC-U2: RCIC SUCCESS TERM
REC-L1-T(E)N025: Sequence Tag REC-L1-T(E)N025

The above sequence (T(E)N025**)** represents a fire event in the turbine building corridor, TG-12. Fire damage produces a loss of offsite power, and unavailability of RHR B, RHR C and HPCS. Depressurization succeeds. SPC is unavailable (due to RHR A in maintenance), RCIC successfully provides injection, but long term operation of RCIC fails due to unavailability of SPC.

Fire Cutset #2:

F4R1D : R-1D Detailed Scenario 4 - Initiator
PRAAHUS--1B-S3LL: FAN PRA-FN-1B DOES NOT START ON DEMAND
REC-L1-TF027: Sequence Tag REC-L1-TF027

The above sequence (TF027**)** represents a fire event in the NW quadrant of the reactor building 471' elevation. Fire damage produces a loss of feedwater / condensate, HPCS, RCIC, and LPCS. High pressure injection is unavailable (due to fire), but depressurization succeeds. LPCI B and C and the service water crosstie are unavailable due to unavailability of service water pump B HVAC (PRAAHUS--1B-S3LL). LPCI A is unavailable due to maintenance. Core damage occurs due to a loss of core cooling.

Fire Cutset #3:

F4R1J: R-1J Detailed Scenario 4 - Initiator
HPS---------T3LL: HPCS UNAVAILABILITY DUE TO T & M (MRULE DATA)
HS-RHRV-MO-23: RHR-V-23 FAILURE CAUSED BY HOT SHORT
REC-L1-TF023: Sequence Tag REC-L1-TF023

The above sequence (TF023**)** represents a fire event in the reactor building 522' elevation. Fire damage produces a loss of feedwater / condensate, RCIC, RHR A, LPCI C and LPCS. HPCS is unavailable due to maintenance (HPS---------T3LL), but depressurization and LPCI B succeed. SPC from Loop B is unavailable due to hot short of RHR-V-23 (HS-RHRV-MO-23). LPCI / RHR A are unavailable due to maintenance. Core damage occurs due to loss of decay heat removal.

Fire Cutset #4:

T3W07 : RC-7 Detailed Scenario 3 - Initiator
EACENG-EDG3-S424: EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN 24H
-L1-SUCC-U2 : RCIC SUCCESS TERM
LPS---------T3LL: LPCS UNAVAILABILITY DUE TO MAINTENANCE (MRULE DATA)
REC-L1-TT018: Sequence Tag REC-L1-TT018

The above sequence (TT018) represents a fire event in the division 2 electrical equipment room, RC-07. Fire damage produces a turbine trip, and FW / PCS, RHR B and C and CRD are unavailable due to fire damage. RCIC is successful initially, but SPC is unavailable for continued operation for RCIC operation (RHR B is unavailable due to the fire and RHR A is out of service). CRD is unavailable once RCIC reaches operational limits. Late depressurization succeeds, but alternate injection fails due to RHR B and C unavailable due to fire, LPCS is unavailable for maintenance, condensate and SW cross-tie unavailability due to fire.

Fire Cutset #5:

LZW03: RC-3 Full-PAU Burnup - Initiator
REC-L1-SBO-R044: Sequence Tag REC-L1-SBO-R044

The above sequence (SBO-R044**)** represents a full Physical Analysis Unit (PAU) area burn-up fire event in the cable chase, RC-03. Fire suppression fails, and this scenario leads directly to core damage due to fire impacts.

For additional results, see Table A-12 for Significant Fire Accident Sequences percentage contribution to CDF.

Based on the review of the above dominant fire sequences, the following insights can be identified.

1. RCIC success, HPCS, DG3, SW-B, and LPCS are very important. To avoid high risk configurations that could exist from equipment taken out of service, the same equipment protected for the internal event will be protected with the addition of LPCS (insight derived from the fire PRA model).

2. The above cutsets include initiating events from fires in the turbine building corridor, TG12, the reactor building 471' and 522', cable chase, and the division 2 electrical and switchgear rooms. This underscores the benefit of establishing a fire watch tour in these areas when RHR-A is taken out of service. Heightened awareness in the form of shift briefs or pre-job walkdown to reduce and manage transient combustibles prior to entrance into the extended completion time will be used to alert the staff about the increased sensitivity to fire. These compensatory measures are not credited in the PRA, but will further reduce the fire risk due to this maintenance configuration.

Seismic

Seismic Cutset #1:

SDS2 : SDS 2 BOP CST LOOP SS LOCA - Initiator
EACENG-EDG2SS4D2: EMERGENCY DG-2 DOES NOT CONTINUE TO RUN FOR 6 HOURS
NRAC24H-SEIS: No Recovery of Offsite AC w/in 24 hrs. (Seismic)
REC-L1-SDS2-50: Sequence Tag REC-L1-SDS2-50

The above sequence (SDS2-50**)** represents a seismic event that results in seismic damage state 2, which is a loss of PCS, the condensate storage tank, offsite power and a small-small LOCA. RHR B is unavailable due to the unavailability of the division 2 DG (EACENG-EDG2SS4D2). For this particular sequence, high pressure injection fails due to CST unavailable and operator failed to transfer HPCS to the suppression pool, but depressurization and low pressure injection (LPCS) are successful. Recovery of offsite power failed (NRAC24H-SEIS**).** Core damage occurs as a result of the unavailability of suppression pool cooling and containment venting.

Seismic Cutset #2:

SDS42: SDS 42 Failure of RPV and/or Category I Bldgs. - Initiator
SEIS-MITGTN-FAIL: NO ADDITIONAL MITIGATION POSSIBLE (SEISMIC SCENARIO)
REC-L1-SDS42-02: Sequence Tag REC-L1-SDS42-02

The above sequence (SDS42-02) represents a seismic event that results in seismic damage state 42, which is a failure of the reactor pressure vessel and proceeds directly to core damage.

Seismic Cutset #3:

SDS41: SDS 41 Wide-spread failure of SSEL equipment - Initiator
SEIS-LONGTERM : PROB OF LONG TERM CORE DAMAGE SEQ FOR SDS41
REC-L1-SDS41-01: Sequence Tag REC-L1-SDS41-01

The above sequence (SDS41-01) represents a seismic event that results in widespread failure of ECCS and proceeds directly to core damage.

Seismic Cutset #4:

SDS41: SDS 41 Wide-spread failure of SSEL equipment - Initiator
SEIS-SHRTTERM: PROB OF SHORT TERM CORE DAMAGE SEQ FOR SDS41
REC-L1-SDS41-02: Sequence Tag REC-L1-SDS41-02

The above sequence (SDS41-02) represents a seismic event that results in widespread failure of ECCS and proceeds directly to core damage.

Seismic Cutset #5:

SDS2: SDS 2 BOP CST LOOP SS LOCA - Initiator
PRAAHUS--1B-S3LL: FAN PRA-FN-1B DOES NOT START ON DEMAND
REC-L1-SDS2-47: Sequence Tag REC-L1-SDS2-47

The above sequence (SDS2-47) represents a seismic event that results in seismic damage state 2, which is a loss of PCS, the condensate storage tank, offsite power and a small-small LOCA. RHR B is unavailable due to the unavailability of SW B HVAC (PRAAHUS--1B-S3LL). For this particular sequence, high pressure injection fails because CST fail and operator fails to transfer HPCS to the suppression pool, but depressurization and low pressure injection (LPCS) succeed. Offsite power is recovered within 24 hours, but core damage occurs as a result of the unavailability of SPC and containment venting.

For additional results, see Table A-13 for Significant Seismic Event Accident Sequences percentage contribution to CDF.

No additional insights were identified based on the review of the above dominant seismic sequences.

3.2 Avoidance of Risk-Significant Plant Configurations (RG 1.177 Tier 2)

The following insight was derived from risk significant contributors based on basic events importance measures (Fussell-Vesely and risk achievement worth (RAW)) listed in Table A-4 through A-9.

Maintenance unavailability basic events for HPCS, TRS, DG3, SW-HPCS, RCIC, and LPCS are risk-significant plant contributors. The insights gained based on the review of importance measures are consistent with the results of the review of dominant sequences.

Examination of the risk reductions for avoiding these plant configurations are examined included in Section 6.

Table 3-3
Risk-Significant Plant Configurations based on Fussell-Vesely Importance

Train Unavailable due to Maintenance	RHR A Out of Service		
	Internal Events	Fire	Seismic
HPCS (HPS---------T3LL)	x	x	
TRS (EACTRL-S----T3--)	x	x	
DG3 (EACEDG-3----T3D3)	x	x	
SW-HPCS (SW-----HPCS-T3LL)	x	x	
LPCS (LPS---------T3LL)		x	
RHR C (RHR----C----T3LL)			
RCIC (RCI---------T3LL)	x	x	

Division 2 DC battery chargers basic event (EDCC1--2A2B-C3LL) is a risk-significant plant contributor. The insights gained based on the review of RAW importance measures are consistent with the results of the review of dominant sequences.

3.3 Configuration Risk Management Program (RG 1.177 Tier 3)

Tier 3 is not discussed in this report. The main body of the LAR submittal will address the Tier 3 element.

3.4 Qualitative Evaluation of External Events and Weather-related Risk

Evaluation of high winds, external floods, volcanic eruption and other external events in the IPEEE per GL 88-20 were submitted to and reviewed by the NRC. The Staff Evaluation Report concluded that CGS IPEEE was capable of identifying the most likely severe accidents and severe accident vulnerabilities and the IPEEE met the intent of Supplement 4 to GL 88-20. The IPEEE determined that the recurrence frequency for the maximum tornado wind speed is approximately 1E-07/yr and, as such, maximum wind speed was eliminated as a plant hazard per the Standard Review Plan. Other external events (e.g., external fire, external floods, high winds, volcanic eruption etc.) were considered to be insignificant contributors to severe accidents. It is judged that the proposed changes should have a negligible effect on the risk profiles from other external events.

External events risk was examined explicitly through the quantification of the internal fire and seismic evaluations. In addition, fire risk management actions are taken per PPM 1.3.85 as part of the Maintenance Rule fire 10 CFR 50.65(a)4 configuration risk management program. Plant vulnerabilities related to weather conditions are explicitly treated on an average basis through modeling of weather-related impacts to the frequency of offsite power loss. Although no credit is assumed for avoiding weather related impacts for planned preventative maintenance conditions, scheduling of planned maintenance of risk significant equipment and systems during forecasted hazardous weather conditions are avoided through the plant risk management procedures and process.

4. PRA Assumptions and Uncertainties

The assessment of uncertainty establishes the level of confidence that can be placed in a decision or conclusion based on a quantitative assessment of risk, with the aim of providing reasonable assurance that the risk-informed decision made based on RG 1.177 guidelines is robust [Ref. 2].

The CGS fire and seismic PRAs are quantified for the RHR CT LAR. The PRAs are based on the Rev. 7.2.1 internal events model. Although the PRAs do not meet all aspects of RG 1.200, the insights derived from them for the RHR CT LAR are judged to be reasonable and applicable based upon the IPEEE development and subsequent updates of the PRA models. However, no formal assessment of modeling uncertainties has been performed for the fire and seismic PRAs.

4.1 Types of Uncertainty

There are three general types of probabilistic risk assessment uncertainty, and each is addressed by this evaluation for the FPIE PRA model [Ref. 6]:

1. Parameter uncertainty – The parametric uncertainty evaluation compares the PRA quantification point estimate results to the parametric mean estimates to ensure adequate agreement.

2. Model uncertainty – Model uncertainty is typically handled by making assumptions. Dispositions for assumptions are treated by assuming alternate assumptions and performing sensitivity analyses as described in Section 4.3.

3. Completeness Uncertainty – The ASME PRA Standard and RG 1.200 provide confidence in the completeness of the treatment of the things we know (that is, scope and level of detail). Safety margins provide protection and margin against the things we do not know. Therefore, completeness uncertainty focuses on evaluation of peer review findings that might require changes to the PRA model. Completeness uncertainty is addressed in Section 4.4, and in the review of PRA quality (Section 5).

4.2 Parametric Uncertainty

The parametric evaluation performed for the base internal events model demonstrated that the point estimate mean and parametric mean are judged to be acceptable [Ref. 6].

A mathematical assessment of uncertainty was performed. This evaluation performed a Monte Carlo calculation for 15000 samples, using the sampling characteristics of the distributions in the type code file.

The uncertainty results indicate that the mean CDF, with uncertainty included, is about 6.45E-06 per year, which is about 4.3E-07 per year higher than the plant CDF equation best estimate of 6.02E-06 per year, or about 7% higher. This increase is typical for an uncertainty analysis where most variables are lognormally distributed.

The close agreement between the mean and the point estimate values show that the PRA model produces a reasonable representation of plant risk considering the uncertainty associated with the individual basic event values. Therefore, it is concluded that the parametric uncertainty is adequately addressed.

4.3 Modeling Uncertainties

To evaluate the influence of modeling assumptions and uncertainties on the RHR completion time extension, the CGS potential key sources of modeling uncertainty for internal events [Ref. 6] are reviewed in Table 4-1. The review identified no modeling uncertainties that challenge the RHR CT extension.

Table 4-1 Review of Generic Sources of Model Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Initiating Event Analysis (IE)						
1. Grid stability	The LOOP frequency is a function of several factors including switchyard design, the number and independence of offsite power feeds, the local power production and consumption environment and the degree of plant control of the local grid and grid maintenance. Two different aspects relate to this issue: 1a. LOOP initiating event frequency values and recovery probabilities	1a) LOOP / SBO sequences	The LOOP frequency for the CGS PRA is the plant-specific frequency developed in NUREG/CR-6890. The LOOP events in NUREG/CR-6890 were analyzed for CGS to derive a curve for probability of non-recovery that was applicable to CGS. This calculation used the same methods, assumptions and principles as NUREG/CR-6890, except that three weather-related events and seven grid-related events were removed. Two of the weather-related events were not used for the CGS calculation because they are characteristic of Atlantic coast storms and are not applicable to the CGS site. A tornado-related event was excluded on the basis that the two power supplies are very unlikely to be disabled by such an event. The BPA power grid has a large proportion of hydro-power and is not vulnerable to the grid disturbances that led to the August 2003 grid blackout. The events from the 2003 grid blackout have been excluded from the calculation of offsite power non-recovery probabilities for Columbia.	Based on the stability of the BPA grid and regional location, offsite power recovery probabilities assume that the 2003 blackout and specific weather-related LOOPs events are judged to not be applicable to the estimation of offsite power non-recovery probabilities.	The offsite power non-recovery probabilities that influence the CGS risk profile have the following importance measurements. Only NR30M meets the risk significance threshold: NR19M - RAW = 1, FV = 0.00015 NR24-AVE - RAW = 1.057, FV = 0.00196 NR30M - RAW = 1.011, FV = 0.01189 NR5-AVE - RAW = 1, FV = 0.00001 NRAC10-24AVE - RAW = 1, FV = 0 NRAC24 - RAW = 1.062, FV = 0.0002 NRAC4 - RAW = 1.009, FV = 0.00052 NRAC5 - RAW = 1.001, FV = 0.00004 NRAC5-24AVE - RAW = 1.024, FV = 0.00025 NRAC6 - RAW = 1, FV = 0 NRAC7 - RAW = 1.005, FV = 0.00013 NRAC9 - RAW = 1.011, FV = 0.00019	Based on the plant-specific approach and impact to the CGS risk profile, offsite power recovery probabilities are identified as a source of model uncertainty. Applications of the PRA should take into consideration the assumptions and approach for estimation of LOOP non-recovery probabilities, and the possible need to perform sensitivity studies. The source of uncertainty related to offsite power non-recovery probabilities does not significantly impact the RHR CT LAR, as the importance of offsite power loss does not change significantly for the CT extension cases from the base PRA quantification results.

Table 4-1 Review of Generic Sources of Model Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Grid stability (continued)	1b) Conditional LOOP probability	1b) All	The industry consequential LOOP probability is 5.3E-3 for the 1997-2004 period. This probability is judged to not be representative of CGS. Based on the strong stability of the BPA grid due to its large hydro-electric base within the Federal Columbia River Transmission System (FCRTS) network (the grid supply of offsite power to CGS), the assumptions associated with grid stability are well supported. CGS plant scrams from power have not resulted in a subsequent loss of an offsite power source due to the BPA grid becoming unstable over the history of operation.	Based on the strong stability and in-depth analysis by BPA along with agreements to periodically re-evaluate the stability of the network, the consequential LOOP modeled in the CGS PRA is assigned to be 1E-3.	The risk-importance of the consequential LOOP probability is: FV = 2.4E-3 and RAW = 3.4, which is risk significant.	The consequential LOOP probability is identified to be a source of model uncertainty, as a plant-specific approach has been applied. Applications of the PRA should take into consideration the assumptions and approach for estimation of the consequential LOOP probability, and the possible need to perform sensitivity studies. The source of uncertainty related to offsite power conditional LOOP probability does not significantly impact the RHR CT LAR, as the importance of offsite power loss does not change significantly for the CT extension cases from the base PRA quantification results. Notwithstanding the insignificant influence of offsite power loss on the RHR CT extension, a sensitivity is performed in which the offsite power loss frequency and conditional LOOP probability are increased.
Success Criteria (SC)						

Table 4-1 Review of Generic Sources of Model Uncertainty						
Topic	**Discussion of Issue**	**Part of Model Affected**	**Plant-Specific Approach Taken**	**Assumptions Made**	**Impact on Model**	**Characterization Assessment**
8. Core cooling success following containment failure or venting through non hard pipe vent paths	Loss of containment heat removal leading to long-term containment over-pressurization and failure can be a significant contributor in some PRAs. Consideration of the containment failure mode might result in additional mechanical failures of credited systems. Containment venting through "soft" ducts or containment failure can result in loss of core cooling due to environmental impacts on equipment in the reactor/auxiliary building, loss of NPSH on ECCS pumps, steam binding of ECCS pumps, or damage to injection piping or valves. There is no definitive reference on the proper treatment of these issues.	Core cooling success following containment failure or venting through non hard pipe vent paths	With the exception of HPCS, containment failure causes loss of all other ECCS pumps due to environmental impacts. HPCS is environmentally qualified for LOCA conditions. A containment failure releasing steam into the HPCS pump room would be assumed to fail HPCS. However, if containment failure occurs in the lower third of the structure, HPCS is conservatively modeled to fail. If failure is in the upper two-thirds of the structure than HPCS is modeled to not be impacted by environmental impacts. HPCS pump NPSH and suction temperature requirements were reviewed, and were determined to be adequate after the containment failure and SP depressurization (Ref. 6).	Conservatively, the CGS PRA assumes that if the containment is failed in the lower one-third, the environmental impacts would fail HPCS.	Likelihood for HPCS failure following containment failure, basic event CF-FAILS-HPCS-INJ, has a significant impact on the CGS risk profile – FV = 2.68E-1, RRW = 1.36, RAW = 3.82	The unavailability modeled for HPCS following containment failure is identified as a source of modeling uncertainty, based on its significant impact on the risk profile and potential conservatism. Applications of the PRA should include consideration of this source of uncertainty. The source of uncertainty related to unavailability of HPCS following containment failure does not significantly impact the RHR CT LAR, as the importance of the applicable basic event does not change significantly for the CT extension cases from the base PRA quantification results.
Systems Analysis (SY)						

Table 4-1 Review of Generic Sources of Model Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Supplementary Topic 12 - Digital instrumentation and control	Some plants have incorporated digital systems into their designs or to replace existing analog systems. There are model uncertainties associated with modeling digital systems, such as those related to determining the failure modes of these systems and components.	1) Loss of reactor feedwater sequences 2) Transient sequences with RFW available	1) Failure of RFW digital control for IEs reflected in Bayesian update of loss of RFW IE 2) Failure of digital control post-trip modeled explicitly	1) No applicable assumptions 2) Likelihood of digital failure control is estimated to be RFW-DIGITALFW = 2E-3	1) Digital control failure reflected in historical plant data 2) Risk significant: RRW = 1.005, RAW = 3.67, FV = 5.3E-3	1) No applicable uncertainties 2) Source of uncertainty for the base model based on risk significance. Applications that encompass RFW system availability may need to examine more closely the reliability of the digital feedwater control system. The source of uncertainty related to reliability of the digital feedwater control system does not significantly impact the RHR CT LAR.

4.4 Completeness Uncertainty

Completeness uncertainty as it relates to the RHR CT LAR is documented in Table 4-2.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty						
Topic	**Discussion of Issue**	**Part of Model Affected**	**Plant-Specific Approach Taken**	**Assumptions Made**	**Impact on Model**	**Characterization Assessment**
Accident Sequence Analysis (AS)						
For the LOOP event tree, the development of the failure branch for P (SORV) excludes the possibility to utilize RCIC for high pressure injection.	Inclusion of RCIC operation for this branch would add considerable complexity to these relatively low-contributing accident sequences.	LOOP accident sequences T(E)N062, T(E)N065, T(E)N069, T(E)N072, T(E)N074, T(E)N078, T(E)N081, T(E)N082, and T(E)N083.	Modeling of RCIC was excluded from development of the failure branch for P (SORV) in the LOOP event tree.	Inclusion of RCIC operation was conservatively excluded.	Based on examination of the LOOP event tree [Ref. 6], the LOOP CDF may decrease by about 3.6E-10/rx-year if RCIC operation is taken into account, which is equivalent to a Fussell Vesely of 6E-5.	Impacts LOOP CDF by small impact to overall risk profile. This is retained as an uncertainty area to consider for PRA applications. The source of uncertainty related to RCIC modeling for LOOP sequences does not significantly impact the RHR CT LAR, as the importance of offsite power loss does not change significantly for the CT extension cases from the base PRA quantification results.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Reactor recirculation pump seal leakage rate for LOOP and SBO modeling	The assumed reactor recirculation pump seal leakage rate is an area of uncertainty for plant PRAs.	A 12.5 gpm reactor recirculation pump seal leakage rate results in RCIC unavailability due to backpressure trip at about 10 hours [MAAP calculation CGS080102, Refs. 22 and 21]. If a 36 gpm rate is assumed, backpressure trip occurs at about 6.5 hours [MAAP calculation CGS080104a, Ref. 6]. Assuming the maximum expected rate of 36 gpm would reduce the time available to recover offsite power from LOOP sequences S27 and S29 and offsite power and diesel recoveries for SBO-I sequences S60 and S61 to about 6.5 hours.	A seal leakage rate of 12.5 gpm is judged to be appropriate estimate for most CGS accident sequences. For scenarios involving extended operation of RCIC using the EDG 4 crosstie, a maximum postulated seal leakage rate of 36 gpm is used to examine timing. The issue of seal leakage rate is evaluated in the treatment of uncertainties and assumptions in the quantification notebook [Ref. 6].	See plant-specific approach taken.	A sensitivity was performed to examine a 6-hour time to recover offsite power for LOOP sequences T(E)027 and T(E)029, and offsite power and diesel recoveries for SBO-I sequences S60 and S61, instead of a 9-hour time. There was no impact to CDF [Ref. 6].	No impact to CDF for the base PRA. This area of uncertainty is retained for consideration for PRA applications. The source of uncertainty related to recirc seal leakage during a loss o of offsite power conditional LOOP probability does not significantly impact the RHR CT LAR, as the importance of offsite power loss does not change significantly for the CT extension cases from the base PRA quantification results.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty						
Topic	**Discussion of Issue**	**Part of Model Affected**	**Plant-Specific Approach Taken**	**Assumptions Made**	**Impact on Model**	**Characterization Assessment**
SW cross-tie to the RHR system and Fire Protection water modeling for loss of offsite power sequences	Due to potential complications during LOOP conditions, credit for alternate injection from SW or FP was not credited. EDG 2 is required for the SW cross-tie, and FP operation relies on the diesel-driven fire pumps.	LOOP sequences T(E)N018, T(E)N027, T(E)N047, T(E)N057, T(E)N074, T(E)N082, T(E)N099, T(E)N0107.	Modeling assumption based on judgment.	SW cross-tie to the RHR system and Fire Protection water to the condensate system are assumed to not be available for loss of offsite power sequences.	If alternate injection was available, LOOP sequences T(E)N027, T(E)N047, T(E)N057, T(E)N074, T(E)N082 may reduce significantly. The total CDF for these sequences is about 1.5E-8/rx-year.	Although there is no significant impact on the overall plant risk profile, consideration of this potential area of uncertainty may be needed for PRA applications. The source of uncertainty related to the SW cross-tie to the RHR system and FP water modeling for loss of offsite power sequences does not significantly impact the RHR CT LAR, as the importance of offsite power loss does not change significantly for the CT extension cases from the base PRA quantification results.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty						
Topic	**Discussion of Issue**	**Part of Model Affected**	**Plant-Specific Approach Taken**	**Assumptions Made**	**Impact on Model**	**Characterization Assessment**
Control Room Habitability	Control room heat-up and habitability require more analysis.	Loss of control room HVAC modeling	The control room is modeled to be habitable given a loss of control room HVAC	The operators are assumed to open doors / install fans given a loss of HVAC, however, ABN-HVAC does not direct these actions. Also, the control room is expected to not temperatures that challenge habitability, however, room heat-up calculations for non-SBO for periods of 24 hours and greater have not been modeled. Room heat-up was found to be relatively fast based on analyses for other plants [Ref. 6].	A conservative sensitivity analysis was performed. The impact to CDF is on the order of 1E-5/rx-year. For the sensitivity, the following conservative assumptions were made: - A simplified fault tree that accounts for a loss of normal HVAC and a loss of emergency HVAC due to a loss of SW A, B or loss of SW A, B and C was prepared. - On a loss of the control room HVAC, operators will not open doors to control room temperature (ABN-HVAC does not direct opening doors) - Given a loss of control room HVAC, the control room will become uninhabitable (no room heat-up calculations have been performed for non-sbo conditions) - The likelihood for operators to evacuate the control room and shut down the plant using remote shutdown is 1E-3. - It is assumed that operators have RCIC as well as manual containment vent, for shutting down the plant. Operators can utilize RCIC only for core cooling; low pressure injection is unavailable due to the loss of service water.	Based on the risk significance of this completeness uncertainty, the assumption that operators will open main control room doors, given a loss of HVAC is a key assumption. All PRA applications should consider the potential impact that this assumption has on risk-informed decisions. In the event of a loss of control room habitability, alternate shutdown credits RCIC, not LPCI, and manual containment venting, not suppression pool cooling. This source of completeness uncertainty therefore does not impact the RHR CT extension application.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
ISLOCA modeling – Refine the modeling phenomenological impacts	Examine whether the ISLOCA event tree modeling can be refined, now that credit is taken for the reactor building floor drain isolation valves (the valves are now periodically tested, and credit for this was taken in Rev. 7.2 for the internal flooding analysis).	ISLOCA	The ISLOCA accident sequence modeling was developed with the assumption that the floor drain isolation valves could not be credited. This modeling could be refined as applicable.	See plant-specific approach.	The potential CDF reduction is 1%.	Potential risk-significant impact on the overall plant risk profile; consideration of this potential area of uncertainty may be needed for PRA applications. The ISLOCA modeling is conservative, from the perspective that it doesn't credit isolation of floor drains in the reactor building basement. ISLOCA is not a significant contributor related to the RHR CT extension quantification results.
Recovery of EDG 1 or EDG 2 given successful crosstie of EDG 3	In the SBO event trees, given a failure to recover offsite power, and given that the EDG 3 crosstie is successful, EDG power is assumed available to either Div 1 or Div. 2 and the EAC recovery node is not questioned (i.e., compare sequences SBO-I008 and SBO-I078). The EAC node could have been considered on Sequence SBO-I008 and SBO-I078 (SBO-R008 and SBO-R052) such that success of the EAC node would mean that two EDGs are available (i.e., EDG power to both Division 1 and 2). However, this level of detail was not included in the model.	SBO sequences SBO-I008, SBO-I078, SBO-R008, and SBO-R052	Assumption made to reduce model complexity. Adding an EAC node also may have limited value, as dependencies may exist between the EDG 3 cross tie and EDG 1 and EDG 2 generator repair.	See discussion of issue.	SBO sequences SBO-I008, SBO-I078, SBO-R008, and SBO-R052 make no contribution to CDF.	There is no significant impact on the overall plant risk profile, but consideration may be needed for PRA applications. The source of uncertainty related to recovery of the EDGs does not significantly impact the RHR CT LAR, as the importance of SBO sequences does not change significantly for the CT extension cases from the base PRA quantification results.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
DC power load shedding for SBO sequences that involve a SORV	DC power load shedding was not credited for the SORV sequences in the SBO-I event tree to reduce complexity. The SBO modeling is split among two event trees and currently involve considerable complexity.	SBO-I sequence SBO-I104	Modeling assumption to reduce model complexity.	DC power load shedding is not credited for the SORV sequences in the SBO-I event tree.	The CDF for SBO-I104 is below the 2E-12/yr truncation.	There is no significant impact on the overall plant risk profile, but consideration may be needed for PRA applications. The source of uncertainty related DC load shedding does not significantly impact the RHR CT LAR, as the importance of SBO sequences do not change significantly for the CT extension cases from the base PRA quantification results.
Phenomenological / environmental impacts to ECCS given an ISLOCA leak.	ISLOCA leak (i.e., leak rather than rupture) is postulated to impact the train that experienced the failed pressure boundary and one other train due to environmental impacts.	ISLOCA sequences IS005, IS011	Modeling assumption for analytical convenience.	Given an ISLOCA leak, phenomenological / environmental impacts to ECCS were assumed to disable LPCS and LPCI-A train	The modeling approach introduces some degree of modeling asymmetry. ISLOCA does not contribute significantly to the overall risk profile.	There is no significant impact on the overall plant risk profile, but consideration may be needed for PRA applications. ISLOCA is not a significant contributor related to the RHR CT extension quantification results.
SRV success criteria for turbine trip ATWS from 25% power	For turbine trip ATWS from 25% power, the turbine bypass system can handle all power generated from the reactor core, and the SRVs would not be required.	TTC2 sequences TTC2026 and TTC2050.	Modeling assumption based on judgment.	The SRV success criteria for turbine trip ATWS from 25% power are assumed to be the same as for full power Turbine Trip ATWS.	These TTC2 sequences contribute 3.6E-12/rx-year to the overall risk profile.	There is no significant impact on the overall plant risk profile, but consideration may be needed for PRA applications. ATWS is not a significant contributor related to the RHR CT extension quantification results.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Calculation of Representative Mission Time for Diesel Generators	This calculation was performed using Rev. 5.0 of the CGS PRA.	LOOP and SBO event tree sequences	An update of the calculation of representative mission time for diesel generators has not yet been performed. It will be considered in a future revision. The mission time used for the diesel generators is chosen to prevent masking of significant core damage sequences by conservative estimates for the failure of AC power events due to conservatism in the diesel run failure rate. The selection of a six-hour mission time in combination with the lumped recovery method leads to acceptable results	The mission time for the diesel generators is assigned to be 6 hours.	A longer mission time would impact the SBO / LOOP contribution to the risk profile.	Selection of a 6-hour mission time is judged to be a consensus approach. Consideration of the potential uncertainty should be made, however, for PRA applications. The source of uncertainty related to the DG mission times does not significantly impact the RHR CT LAR, as the importance of offsite power loss does not change significantly for the CT extension cases from the base PRA quantification results.
Credit for Manual Containment Vent	The FLDAC event tree (as well as DAC, and DDC event trees) do not model the W2 function and therefore do not include modeling for manual containment vent (ABN-CONT-VENT). Now that manual containment venting is modeled, the W2 function can be added to these event trees.	FLDAC, DAC, DDC	The W2 function was not previously creditable. W2 can now be added to the FLDAC, DAC, DDC event trees.	None	FLDAC sequence 8 could reduce as a result of this observation. FLDAC sequence 8 contributes 1.6E-7/yr, to the total core damage frequency and therefore a potential 3% total CDF reduction is possible.	There is a risk-significant impact to the risk profile, therefore this is a key area of uncertainty. The modeling is conservative, however, and doesn't contribute significantly to the RHR CT extension.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Non-recovery Probabilities for Losses of Divisional AC or DC Power due to Common Cause Failure	For nuclear power plant safety-related systems, non-recovery probabilities at two hours generally are about 0.2, based on a study performed for EPRI [Ref. 6]. However, the evaluation of non-recovery for electrical systems in the study focused on diesel generators. Since no specific information was found regarding the likelihood for recovery divisional dc or ac power buses, engineering judgment, including considerations of time and manpower available, was used to assess the following recovery failure probabilities.	Event tree sequences for dual loss of 125 VDC Division 1 and 2 power and dual loss of 4160 VAC Division 1 and 2 power (DDC and DAC)	As no industry non-recovery values were identified, non-recovery / non-repair probabilities were assumed. The failure probabilities are judged to be appropriately conservative given the lack of guidance / data / research. There is no significant impact on the overall plant risk profile, as discussed in the quantification notebook [Ref. 6] but consideration may be needed for PRA applications.	Recovery / repair would rely heavily on skill-of-craft operator actions. The following non-recovery / non-repair probabilities were assigned based on engineering judgment: Non-recovery in: 5 hours – 0.5 6 hours – 0.5 7 hours – 0.5 10 hours – 0.3 24 hours – 0.1	The RAW and RRW values for the non-recovery events are 1.0, indicating a negligible impact on the CGS risk profile.	No alternate approaches currently available. Contribution to risk profile for these initiators is small and judged to be reasonable. PRA applications that are influenced by the availabilities of electrical equipment should take into consideration that the non-recovery values are based on engineering judgment. The modeling of dual losses of ac power and dc power don't contribute significantly to the RHR CT extension results.
Success Criteria (SC)						

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
The potential need to turn off low pressure ECCS pumps that are running on minimum flow within 30 minutes to avoid pump damage has been cited by CGS Operations.	Some in the industry have expressed a potential issue for ECCS pumps operating on minimum flow for extended periods, potentially leading to pump damage.	Sequences involving low pressure injection	The PRA model does not model a potential impact on ECCS pumps if they run extended periods on minimum flow.	The ECCS pumps are assumed to be capable of running an extended period on minimum flow without damage (hours).	The PRA modeling approach conforms to other plant PRAs. Potential non-conservatism in the PRA results. There could be a need to model an operator action to manually turn off the pumps and turn them back on as needed.	This is retained as a candidate modeling uncertainty. PRA applications should consider in more detail the realism of this PRA modeling, and whether additional modeling needs to be considered (e.g., model a human failure event for failure to secure ECCS pumps not in use). The PRA modeling approach conforms to other plant PRAs and represents the industry-consensus approach. This is judged to not be a source of modeling uncertainty for the RHR CT extension.
Systems Analysis (SY)						

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Low Pressure Injection and Suppression Pool Cooling System Dependencies	Examine the need for these dependencies for LPCI and suppression pool cooling: 1) Do the RHR pump seals require seal cooling, as modeled by the PRA? 2) Do the RHR pumps require room cooling as modeled by the PRA?	Event Tree Function V1	RHR pump seal cooling is modeled as not required for LPCI pump C, per a CVI calculation [Ref. 6]. It is likely that RHR pumps A and B also should not include the dependency on seal cooling. RHR pump room cooling calculations were performed for the FSAR assuming suppression pool cooling was available. The CDF core damage sequences modeled by the PRA involve failure of containment heat removal, and therefore the FSAR room heatup calculations are not applicable for the PRA. According to the post fire safe shutdown analysis, NE-02-85-19, the LPCI C room heats up to 155 degrees F within one hour which is very close to the pump motor design limit of 166 degrees F, (ME-02-98-18), so certainly more research / analysis is needed.	Room cooling is assumed to be required for the RHR pumps A, B, and C.	If the seal bearing cooling dependency is removed from RHR pumps 1A and 1B, to be consistent the LPCI pump C, there is no significant change in CDF (as the SW dependency for the RHR pumps is retained due to the dependency on room cooling). If the room cooling dependency were to be removed from the RHR pumps, the total CDF reduces by 3.3E-7/yr, or 5.4%.	The impact on the model is risk-significant. Therefore this is a key source of modeling uncertainty. The PRA modeling that models seal bearing cooling as required is conservative. Model refinement to remove the conservatism is judged to not change the conclusions of this analysis (for example, the conclusion that the RHR SW crosstie must be protected for the corrective maintenance case is judged to not change as a result of addressing this completeness issue).

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Suction Transfer to the Suppression Pool	Suction transfer to the suppression pool is modeled conservatively.	All	The current plant EOPs credit RCIC suction from the suppression pool up to 240°F. Rev. 7.2 of the PRA conservatively did not take this credit, and this is an area of completeness uncertainty. Additionally, it was discovered later in the development of Rev.7.2 that the HPCS suction fault tree logic is also conservative. The signal for automatic transfer is not credited (the valve transfers are modeled, but the fault tree logic requires operators to perform the suction transfer). This is an area of completeness uncertainty.	No assumption made. This is an area of completeness uncertainty.	If RCIC suction from the suppression pool is credited, CDF reduces by 1.98E-8/yr (FV 0.0033). If HPCS suction credits the automatic signal for swapping to the suppression pool, CDF reduces by 2.86E-8/yr (FV 0.0047). The combined CDF reduction is 4.8E-8/yr (FV 0.008).	PRA applications should consider whether any impacts arise from this modeling incompleteness (producing slightly conservative results). The PRA models RCIC / HPCS suction conservatively. Model refinement to remove the conservatism is judged to not change the conclusions of this analysis (for example, the conclusion that the RHR SW crosstie must be protected for the corrective maintenance case is judged to not change as a result of addressing this completeness issue).

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
System Walkdowns	System walkdowns have not been performed recently to confirm that the systems analysis correctly reflects the as-built, as-operated plant. The ASME/ANS Standard requires system walkdowns, and therefore maintaining the PRA to represent the as-built as-operated plant ought to include system walkdowns.	All	System walkdowns have not been performed recently to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	None	Potentially risk-significant impacts.	Development of PRA applications should include consideration for performing system walkdowns. It is judged that system walkdowns would not significantly alter the conclusions of the analysis. Interviews with system engineers were performed recently in 2014, which identified only minor potential enhancements for system fault tree modeling, none of which are related to RHR.
Human Reliability Analysis (HR)						
Use of a combination of multiple methods for treatment of cognitive errors for selected HEPs	Given that the Cause Based Decision Tree Method (CBDTM) does not explore the full impact of timing on the induced stress for cognitive decision making, the Rev. 7.1 HRA method combined the cognitive HEPs generated from CBDTM and the Time Reliability Correlation as allowed by the EPRI HRA Calculator®.	The cognitive HEPs calculated for post initiation events are affected.	For Rev. 7.2 most HEPs were revised to use either the CBDTM or HRC/ORE methods. This approach follows the industry consensus and best practice.	Industry consensus approach employed. Some HFEs still employ the combination sum technique and are candidates for refinement in a future update.	There are three risk-significant HFEs that still employ the CBDTM+ASEP combination sum technique. Conservatism arising from this technique may affect the plant risk profile: MS-HUMNLOUMSH3LL OPERATOR FAILS TO ISOLATE MOD MS LOCA OUTSIDE CONT FV = 0.0057 REAHUMNHVMCCH3LL FAILURE TO PROVIDE ALT HVAC FOR REACTOR BLDG MCC ROOMS FV = 0.01137	PRA applications should consider the potential conservatism introduced by the HEP development for MS-HUMNLOUMSH3LL and REAHUMNHVMCCH3LL. These HFEs do not significantly impact the RHR CT extension quantification results.
Internal Flooding (IF)						

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty						
Topic	**Discussion of Issue**	**Part of Model Affected**	**Plant-Specific Approach Taken**	**Assumptions Made**	**Impact on Model**	**Characterization Assessment**
Barrier Integrity	Inter-area barriers pass no fluid until their design capacity is reached. Barriers fail when the design capacity is reached.	Timing of propagation	Assume complete failure when flood levels reach the design capacity of barriers.	Assume complete failure when flood levels reach the design capacity of barriers.	This assumption can result in minor changes in timing of propagation or equipment failure.	For lower elevations in the reactor building, barriers are designed to withstand 44 feet of water. A better estimate of barrier strength would result in longer times to propagate. However, the volume of water to reach 44-feet is so large that these changes would show only minor impact to the overall results. For higher reactor building elevations, accumulation above 6 inches is not possible so little change would be expected. Other buildings are provided with large openings that prevent accumulation and, therefore, challenging barriers. The effect of barrier integrity could impact specific applications. Water floods in the reactor building do not significantly impact the RHR CT extension quantification results.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Reactor Building Sump Isolation Valves	Sump drain isolation valves in the reactor building are modeled in Rev. 7.2, as the valves are now routinely tested. However, the model does not yet include common cause failures for these valves.	Flood propagation and timing	Issue of completeness – common cause failures not yet modeled for the drain isolation valves.	Completeness issue	There is no impact to the CDF results if CCF is modeled for the sump isolation valves based on a sensitivity evaluation. Common cause failure is below the quantification truncation, indicating that there are no cutsets above the truncation that contain more than one sump isolation valve failure basic event (no combinations of two or three valves, FDR-V-607, FDR-V-608, FDR-V0609.	Reactor building floods are not a significant contribution to overall risk. Scenarios that involve drain valve failures are a small subset of the reactor building flood risk. Although not significant to the overall plant risk profile, the lack of common cause modeling for the drain isolation valves may warrant discussion /consideration as part of PRA applications. Water floods in the reactor building do not significantly impact the RHR CT extension quantification results.
Flood Damage Heights for Division 1 and Division 2 Electrical Equipment in Vital Island	At the time of the flooding walkdown, flood damage heights for the 4KV buses SM-7 and SM-8 could not be confirmed.	Internal flooding scenario results for the radwaste and control building.	A flood damage height of 0 inches was assumed.	See plant-specific approach.	A higher flood damage height (for example six inches) is judged to not significantly impact the current results.	Refinements to radwaste and control building flood scenarios should be considered for a future PRA update, which will examine potential plant modifications and will take into account refinements in flood damage heights of equipment in the vital island. For PRA applications, examine this area of model completeness for potential impacts to risk-informed decisions. Control building floods do not significantly impact the RHR CT extension quantification results.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Flood damage to the Division 1 and 2 battery chargers	For floods grouped with the FLDAC initiator (flood-induced loss of divisional ac power, division 1 and division 2), investigate whether the division 1 and division 2 battery chargers survive these floods, based on flood damage height.	Internal flooding event tree FLDAC	Flooding in the switchgear rooms is postulated to damage the battery chargers, in addition to the switchgear, once the flood damage height is reach.	See plant-specific approach taken	This is judged to likely be a documentation issue. No significant impact to the risk profile is expected.	For PRA applications for which the FLDAC CDF contribution could influence the risk-informed decision, evaluate the potential that the battery chargers are not impacted for switchgear room floods. FLDAC does not significantly impact the RHR CT extension quantification results.
Human-Induced Floods	PSA-2-FL-0003 Section 2.1, Methodology, includes a discussion on "Maintenance-Related Flooding Frequencies", which concluded that that maintenance-induced flooding events are negligible contributors to overall initiating event frequency based on a calculated frequency of 2.70E-5/yr. This conclusion was not reconciled in the documentation with dominant flooding initiators IE-FLD-C502TSW-U, IE-FLD-C507TSW-M, and IE-FLD-C508TSW-M, which are all around 1E-6/yr.	Internal Flooding	Maintenance-induced flooding events are not modeled by the internal flooding PRA.	Maintenance-induced flooding events are negligible contributors to internal flooding risk	Not likely to be risk-significant, as maintenance-induced floods are expected to be quickly detected and corrected by plant operators prior to damage of plant equipment due to flooding impacts.	This should not be a source of model uncertainty in most applications. For applications in which internal flooding is particularly applicable, consideration should be given to the impacts from human-induced floods. This source of modeling uncertainty is judged to not significantly impact the quantification results for the RHR CT extension.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Postulated piping break in the flooding scenario fails the affected system / train with the break	System operation potentially can be recovered if the portion of the system leaking is isolated and the effects of the flooding does not impact system availability.	Internal flooding scenarios involving TSW, SW, and CST piping.	The postulated piping break in the flooding scenario fails the affected system / train with the break.	The postulated piping break in the flooding scenario is assumed to fail the affected system / train with the break.	The risk-significant internal events floods with FV > or equal to 0.005 were evaluated and none would produce a different result if the assumption for postulated piping break in the flooding scenario fails the affected system / train with the break were altered to give some credit for system available of the affected system.	No significant impact to CDF for the base PRA [Ref. 6]. This area of uncertainty is retained for consideration for PRA applications. The PRA modeling assumes that a leaking plant system is unavailable for plant shutdown. This conforms to the industry consensus approach. Any relaxation of this assumption is judged to not significantly impact the RHR CT extension quantitative results.
Data Analysis (DA)						
Estimating Plant-specific Demands	The number of plant-specific demands on standby components was mainly documented for the maintenance rule and MSPI components. Tier 3 PSA documents for PSA-2-DA-0002 show the details. Estimates based on the surveillance tests and maintenance acts as described in this SR should be performed even though the major components have been included in the MSPI data. Estimates based on the surveillance tests and maintenance acts as described in DA-C6 and DA-C7 should be performed for significant components whose data are not tracked in the MSPI data.	Unavailability data for significant components not tracked in the MSPI data.	See discussion of issue.	None.	Potential impact to system unavailabilities	Evaluate this source of modeling uncertainty for PRA applications. This modeling uncertainty impacts risk-insignificant components and this uncertainty is judged to not impact the RHR CT extension quantification results.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Component Failure Mode	There is unmeasured uncertainty in the assignment of a NUREG-6928 failure mode to the failure mode of the CGS PRA. The classification of events into failure modes by NUREG/CR-6928 is not specified. It is not possible to match it with the CGS definition of the same event.	All	Generic data has been utilized appropriately.	None.	n/a	No alternative approaches exist at this time. This is not a candidate for modeling uncertainty.
Quantification (QU)						
Use of the rare event approximation for PRA solutions	The CGS PRA is solved by CAFTA, which uses the minimum cutset upper bound solution. The minimum cutset upper bound solution can be seen when the cutset file (*.cut) is opened. However, the results reported in PRAQuant use the rare event approximation.	All portions of the Level 1 and Level 2 PRAs	The CGS PRA is solved using CAFTA, which uses the minimum cutset upper bound. The results reported in PRAQuant use the rare event approximation. All accident sequence and LERF cutsets have frequencies less than 1E-6/yr. The total CDF is less than 5E-5/yr. The rare event approximation is therefore valid.	None required. Industry-accepted software program used, and the requirements for the rare event approximation are met.	n/a	The rare event approximation is appropriate for the PRA quantification. For any PRA application that utilizes a result for which basic event unavailabilities or frequencies are greater 0.1 (for example, the unavailability of HPCS for some specific application), the validity of the rare event approximation will need to be considered when utilizing the PRAQuant results. That is, the PRA practitioner will need to compare the PRAQuant results to those in the cutset (*.cut) file (typically very small differences). The CT extension calculation are based on the minimum cutset upper bound, and therefore this source of uncertainty is not applicable to this application.

Table 4-2 Review of Sources of Model Uncertainty and Completeness Uncertainty

Topic	Discussion of Issue	Part of Model Affected	Plant-Specific Approach Taken	Assumptions Made	Impact on Model	Characterization Assessment
Success Branch Modeling	For PRA applications, event tree branch probabilities may increase (if an electrical bus is out of service for example). This becomes an issue when the down-branch probability becomes greater than 0.1, as the rare event approximation is challenged. (Note that if the down branch is TRUE, CAFTA produces the correct solution, since the PRA quantifies the success branch as FALSE in these instances).	All	PRA Rev. 7.2 employs static success branch modeling for risk-significant success branch probabilities. The PRA does not employ automated success branch modeling, and therefore the PRA may produce conservative results.	None	Depending on the application, core damage frequencies and particularly LERFs may be conservative. Typically this conservatism is expected to be no more than 20% for PRA applications (result could be up to 20% less if automated success branch were employed).	Any PRA application should take into consideration the potential that CDF / LERF is overestimated due to success branch modeling. Care should be taken if the margin for the risk-informed decision is small, to consider whether success branch modeling has an impact. A future PRA version can consider automating the success branch modeling, using the XNEG feature of FTREX. Employing XNEG would require joint HFE dependent events to be modeled explicitly in the top logic fault tree, rather than being applied in cutset post-processing, which has the disadvantage that it cannot be as easily updated as the cutset post-processing approach to applied dependent HFEs. The current PRA modeling is conservative, and judged to not significantly impact the RHR CT extension application.

5. PRA Quality

5.1 FPIE Model Technical Adequacy

The CGS PRA model of record is Revision 7.2.1. The Columbia PRA meets Capability Category II of Addendum A of the ASME ANS Standard, ASME/ANS RA-Sa-2009 [Ref.5], as clarified by RG 1.200, Rev. 2 [Ref. 4], based on the following considerations:

- In 2004 the CGS Revision 5.0 internal events PRA received a full scope peer review against the Capability Category II (CC-II) requirements of the ASME/ANS PRA Standard, ASME RA-Sa-2003, as clarified by Regulatory Guide (RG) 1.200 (DRAFT), using the industry peer review process guidelines described in Nuclear Energy Institute (NEI) NEI-00-02, Revision A-3, "Probabilistic Risk Assessment Peer Review Process Guidance."

- In 2009, the Columbia PRA Rev 7.0 received a full scope peer review from the Boiling Water Reactor Owners' Group (BWROG) against the ASME/ANS PRA Standard RA-Sa-2009 [Ref. 5], as clarified by Regulatory Guide 1.200, Rev. 2 [Ref.4], using the industry peer review process guidelines in NEI-05-04, Revision 2, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard."

- The Columbia PRA Rev. 7.1 resolved all F&O *Findings* from the 2004 and 2009 Peer Review, with the exception of Finding 2-2 for supporting requirement (SR) DA-C6 (SR DA-C6 meets Capability Category II, as assessed by the 2009 peer review) and 2-14 for supporting requirement SY-A4 [Ref. 8].

- Since the development of Rev. 7.1 in 2010, the CGS plant design has been reviewed for permanent plant changes, such as design or operational practices. For the Rev. 7.2 PRA update, those changes that were found to have an impact on the PRA model have been incorporated into the baseline PRA model.

- As part of the risk-informed technical specification initiative 5b license amendment request, the open findings from the 2009 peer review, 2-2 and 2-14, were resolved. However, finding 2-2 has not been incorporated into the model. As noted in Section 6.3, a sensitivity case was performed for this issue and it was concluded that this finding is judged to have no impact on the RHR A CT extension.

- In 2016, PRA Rev. 7.2.1 was issued as part of the risk-informed Inservice Inspection. Selected dependent HEP values were modified for use in the non-LERF Level 2 release term quantification and separate recovery files were constituted to implement these recoveries. There was no effect on the CDF and LERF quantifications.

Columbia maintains a PRA Configuration Control Program. The Configuration Control Program is governed by SYS-4-34. SYS-4-34 contains the following key elements:

- A process for monitoring PSA inputs and collecting new information,

- A process that maintains and updates the PSA to be consistent with the as-built, as-operated plant,

- A process that ensures that the cumulative impact of pending changes is considered when applying the PSA,
- A process that maintains configuration control of computer codes used to support PSA quantification, and
- Documentation of the program.

To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plant the above activities are routinely performed. In accordance with this guidance, a review of PRA inputs and new information nominally occurs on a four year cycle; shorter intervals may be required if plant changes, procedure enhancements, or model changes result in significant risk changes.

CGS RG 1.200 compliance database is Columbia's PRA model update tracking system. All self assessment (SA) and Plant Design Changes (PDC) that impact the PRA model are input into the database. The database is reviewed during each PRA update and all items that significantly impact the PRA are resolved and incorporated into the model to keep it consistent with the as-built, as-operated plant. Table 5-1 shows the model revision and major changes. The table shows that Columbia's PRA model reflects the as-built as-operated plant since the Rev 7.0 model update.

Table 5-1
PRA Model Revision Summary

Rev #	***Issue Date***	***Revisions Highlights and Documentation***	***Results***
7.0	7/2009	• PSA upgrade in which the remaining 2004 peer review F&Os were resolved, as well as self-assessment F&Os, prior to the 2009 peer review. Updated Level 2 PRA.	
7.1	6/2010	• Resolved most 2009 peer review F&O findings and some suggestions.	CDF = 7.6E-6/yr, LERF = 3.6E-7/yr
7.2	6/2014	• Resolved self-assessment comments collected since 2010 as well as more 2009 peer review F&O findings and suggestions. Converted the PSA from WinNUPRA to CAFTA.	CDF = 6.0E-6/yr LERF = 3.3E-7/yr
7.2.1	6/2016	• Selected dependent HEP values were modified for use in the non-LERF Level 2 release term quantification and separate recovery files were constituted to implement these recoveries. There was no effect on the CDF and LERF quantifications.	CDF = 6.0E-6/yr LERF = 3.3E-7/yr

A review of the database was performed to determine the impact of any open PDCs, since the last PRA update in 2014 that could potentially impact this risk assessment. The open items are summarized in Table 5-2.

Table 5-2
Open PDCs that Potentially Impact RHR A CT Extension

PDC Number	Description	Disposition	Impact on RHR A CT Extension
EC 12954	In this modification, Electrical Panel E-PP-7AA (located in the control room) is being split into two panels. The right panel will become a safety related panel (E-PP-7AA) and the left panel will become the non-safety related panel (E-PP-US/D). Non-safety related loads presently connected to E-PP-7AA will be moved to EPP-US/D. Some safety related loads will be rearranged on E-PP-7AA. Panel E-PP-US/D will be re-fed from the spare fuse disconnect from panel E-PP_US.	Revise FT modeling as applicable	No impact
EC 14838	Feedwater oil pressure logic change.	Update the change to the RFW logic for control oil pressure. The elimination of an SPV should reduce the CDF.	No impact. Feedwater is not significant to this risk assessment.
EC 14786	Changing power source to RRA-FN-21, and RRA-FN-9 so that the steam tunnel cooling fans will remain powered under certain load shed conditions The load centers that the fans are fed from now, E-MC-7C and EMC-8C are load shed given.	Revise the power supplies for the steam tunnel fans per EC 14786. This modification will mean that MSIVs can remain open, if the isolation signal is only due to a loss of steam tunnel cooling.	This plant design change further reduces the risk-significance of suppression pool cooling, and thus, reduces the risk of RHR-A CT. It is scheduled for implementation in 2017. This analysis conservatively doesn't credit this design change.

Table 5-2
Open PDCs that Potentially Impact RHR A CT Extension

PDC Number	Description	Disposition	Impact on RHR A CT Extension
EC 13094	Hardened Containment Vent	Update the model to incorporate the hardened and vent into the model	This plant design change further reduces the risk-significance of RHR-A CT. It is scheduled for implementation in 2017. This analysis conservatively doesn't credit this design change.

Conclusion

Based on these considerations, the Columbia PRA is therefore technically adequate to address PRA applications that require Capability Category II. Additionally, the review of modeling assumptions, uncertainties, and PRA quality, identified no model limitations that would significantly impact the one-time CT extension application.

5.2 Fire Model Technical Adequacy

The fire PRA is based upon the IPEEE development and received a subsequent update in 2002, 2004 and 2006. The fire PRA has also been updated to utilize the current internal events model of record. The Rev. 7.2.1 internal events PRA model of record serves as the basis for the current Fire PRA model.

The Fire PRA received a Peer Review in January 2004. The peer review team used the Revision A-3 NEI draft "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance", NEI 00-02, dated June 2, 2000 as the basis for the process to conduct the review. However, the criteria for the review were derived from the then current applicable sources.

The specific technical items and criteria for assignment of Capability Categories for the 2004 Peer review were based on the following:

- Fire Events: Checklists are developed from available fire PRA best practices, EPRI Guidance documents, NUREGs, and NRC Guidance.

While the peer review process and PRA standards have evolved since 2004, this peer review provided a valid, critical detailed review of the Fire PRA model.

Subsequently, the 2006 Fire PRA model was developed following the general methods discussed in the EPRI Fire PRA Implementation Guide and some of the methods of NUREG/CR-6850. This Fire PRA update addressed all of the findings from the January 2004 Fire Peer Review.

A current effort is underway to upgrade the existing Fire PRA to meet all aspects of the ASME/ANS PRA standard at a capability category (CC) II level. While the current Fire PRA

does not meet all of the PRA standard supporting requirements at a CC II level, the following reasons are provided as to why the Fire PRA is adequate for this application.

- The mapping of fire zones to risk significant equipment and cables is adequate to ensure that the impacts from fires are identified. In other words, the Fire PRA appropriately captures the spatial relationship of risk significant equipment to fire areas in the plant. The Fire PRA model results, as discussed in Section 3.1.1, are reasonable and provide assurance that the Fire PRA produces useful results for identifying appropriate risk management actions to minimize risk.

- The Fire PRA has been used to obtain risk insights for past risk-informed licensing requests including a permanent DG allowed outage time TS change request and a LAR submittal to implement TS initiative 5b (approval pending).This risk-informed application is similar in that it requires the same level of technical quality of the fire PRA.

Based on the reasons provided above, CGS concludes that the Fire PRA is technically adequate for use in this application.

5.3 Seismic Model Technical Adequacy

The seismic PRA is based upon the IPEEE development and it has been updated in 2004 and 2007. The seismic PRA has also been updated to utilize the current internal events model of record. The Rev. 7.2.1 internal events PRA model of record serves as the basis for the current seismic PRA model.

The quantification results for the seismic PRA are provided in Section 3. A reasonableness check of the results was performed and is discussed in Section 3.1.1.

The seismic PRA has not been peer reviewed, but is used in this application to obtain insights, where applicable. The quantitative results are well within the acceptance criteria for ICCDP and ICLERP. The quantitative results are supplemented by qualitative insights as discussed below.

A fundamental concept of the seismic modeling is that similar components at the same location of a building will experience similar seismic forces. All three RHR pumps are located on the basement elevation of the Reactor Building and would be expected to experience the same seismic acceleration from a seismic event.

In addition, with each pump being the same style, make, and model, the pumps would be expected to have the same fragility. Motor driven pumps generally have high seismic capacity and a relatively low probability of failure compared to other components during a seismic event. For the purposes of a seismic risk evaluation, the RHR pumps can be treated as completely seismically correlated (i.e., a seismic event that can cause failure of one RHR pump is likely to cause failure of all RHR pumps). Therefore, unavailability of the RHR pump would not have a significant impact on overall seismic risk.

Based on the low contribution of seismic risk to the risk spectrum, it is concluded that the seismic PRA model combined with qualitative risk insights is technically adequate to support this risk application.

6. Sensitivity and Uncertainty Analyses

The following sensitivity analyses were performed as part of this evaluation (see Attachment B):

6.1 Sensitivity Case 1: Assumed Compensatory Measures

Compensatory measures are assumed to be in place to avoid the risk-significant plant configurations and to reduce the risk increase which results from the proposed RHR A OOS for preventive maintenance. The results are shown in Table B-1. The following compensatory measures are assumed for case 1.

HPCS, TR-S, DG3, LPCS, RCIC, RHR-C and HPCS-SW are protected. The maintenance unavailability for the protected trains is set to zero. The maintenance unavailability for the protected trains is set by flags files as follows:

```
HPS---------T3LL      EQU   .F.
SW-----HPCS-T3LL      EQU   .F.
EACTRL-S----T3--      EQU   .F.
EACEDG-3----T3D3      EQU   .F.
RCI---------T3LL      EQU   .F.
RHR----C----T3LL      EQU   .F.
LPS---------T3LL      EQU   .F.
```

Based on this sensitivity case with the additional compensatory measures in place, the ICCDP was reduced by almost half for both the FPIE and Fire PRA results.

6.2 Sensitivity Case 2: LOOP and Conditional LOSP Probability

To examine the sensitivity of offsite power loss frequency and consequential loss of offsite power (LOSP) probability, the frequency and probability were doubled for RHR A OOS base case. All quantitative results for ICCDP and ICLERP for this sensitivity were less than the guidance thresholds, with adequate safety margin. See Table B-2. The frequency and the probability were doubled by flag file as follow:

```
EAC-CONSEQ-LOOP       PROB 2E-3
TE                    PROB 6E-2
```

The results show that the base case is not sensitive to LOSP because LOSP is not a significant contributor to Columbia’s risk profile.

6.3 Sensitivity Case 3: Completion Uncertainty related to Failure Estimates

To address completeness uncertainty, a sensitivity was performed for the RHR A OOS base case to examine sensitivity to the resolution for F&O 2-2. For the resolution of F&O 2-2, estimates for significant events not tracked in the MSPI data are now based on surveillance test and maintenance records.

This F&O resolution impacted the following component failure modes:

C---W2 (compressor fails to start),
C---W4 (compressor fails to run),
FN--R3 (fan fails to start),
FR--W4 (fan fails to run),
AHUS---S3 (air handling unit fails to start),
AHUR---W4 (air handling unit fails to run).

The revised failure data will be utilized in the next PRA update. As the revised failure data is not part of the base model, a sensitivity was performed to examine the sensitivity of the RHR CT LAR to this resolution. See Table B-3.

The results show that the base case is not sensitive to the F&O 2-2 resolution not yet incorporated into the PRA model. The resolution of this F&O has no impact on the application for the RHR-A extension.

7. Compensatory Measures and Conclusions

Based on the risk-significant contributors to ICCDP the following compensatory measures will be taken during the 14 days CT for RHR A OOS to address the risk-significant PRA scenarios:

- Besides the protected equipment required by PPM 1.3.83 (RHR-B, DG2 and SW-B), the following equipment will be protected HPCS, HPCS-SW, TRS, DG3, RCIC, LPCS and RHR-C.

- Also, DC Div 2 battery chargers, CAS and SGT will be protected, but no credit was taken in the PRA evaluation for this protected equipment. Therefore, the impact of these actions has not been quantified.

The following additional compensatory measures will be implemented to address the risk-significant PRA scenarios identified during the review of cutsets and importance measures. No credit was taken in the PRA and therefore the impact of these actions has not been quantified.

- Establish flood watch tour of the Turbine Building to provide potential early detection of internal floods.

- Establish fire watch tour for the turbine building corridor (TG-12), reactor building 471' and 522', cable chase, and the division 2 electrical and switchgear rooms when RHR A is taken out of service.

- Increase awareness of fire risk by performing shift briefs or pre-job walk downs to reduce and managed transient combustibles.

- Establish fire (a)(4) risk management actions required per PPM 1.3.85 [Ref.13] which are shown in Table 7-1.

Table 7-1
Maintenance Rule fire (a)(4) Risk Management Actions based on the Fire Zones

Fire Risk	Zone Description	Risk Management Actions
RW 437 N	Waste Tank Floor Area C106; RW 437 Behind Shield Wall; Offgas Dryer Rooms C130 and C131	Initiate Fire Prevention Evaluation Permit per PPM 1.3.10 to: - implement hourly fire tours for Waste Tank Floor Area C106 per FPP-1.7 - ensure no unnecessary fire hazards and no obstructed fire equipment in Waste Tank Floor Area C106 - confirm area detection, suppression and fire doors available for Waste Tank Floor Area C106, RW 437 Behind Shield wall and Offgas Dryer rooms C130 and C131 - prohibit hot work in Waste Tank Floor Area C106, RW 437 Behind shield wall and Offgas Dryer rooms C130 and C131 - update plant status page with plant fire risk status and affected fire area for station awareness
RW 467 RPS 2 CHG 2	Division 2 RPS Room C213; Division 2 Battery Charger Room C224	Initiate Fire Prevention Evaluation Permit per PPM 1.3.10 to: - implement hourly fire tour for Division 2 RPS Room C213 and Division 2 Battery Charger Room C224 per FPP-1.7 - ensure no unnecessary fire hazards and no obstructed fire equipment in Division 2 RPS Room C213 and Division 2 Battery Charger Room C224 - confirm area detection, suppression and fire doors available for Division 2 RPS Room C213 and Division 2 Battery Charger Room C224 - prohibit hot work in Division 2 RPS Room C213 and Division 2 Battery Charger Room C224 - update plant status page with plant fire risk status and affected fire area for station awareness
RW 525 HVAC 1	Division 1 HVAC Equipment Room C507	Initiate Fire Prevention Evaluation Permit per PPM 1.3.10 to: - implement hourly fire tour for Division 1 HVAC Equipment Room C507 per FPP-1.7 - ensure no unnecessary fire hazards and no obstructed fire equipment in Division 1 HVAC Equipment Room C507 - confirm area detection, suppression and fire doors available for Division 1 HVAC Equipment Room C507 - prohibit hot work in Division 1 HVAC Equipment Room C507 - update plant status page with plant fire risk status and affected fire area for station awareness

Table 7-1
Maintenance Rule fire (a)(4) Risk Management Actions based on the Fire Zones

Fire Risk	Zone Description	Risk Management Actions
RW 525 CHLR	Emergency Chiller Room C502; Communications Room C503; Instrument Shop Room C510	Initiate Fire Prevention Evaluation Permit per PPM 1.3.10 to: - implement hourly fire tour for Emergency Chiller Room C502; Communications Room C503 and Instrument Shop Room C510 per FPP-1.7 - ensure no unnecessary fire hazards and no obstructed fire equipment in Emergency Chiller Room C502; Communications Room C503 and Instrument Shop Room C510 - confirm area detection, suppression and fire doors available for Emergency Chiller Room C502; Communications Room C503 and Instrument Shop Room C510 - prohibit hot work in Emergency Chiller Room C502; Communications Room C503 and Instrument Shop Room C510 - update plant status page with plant fire risk status and affected fire area for station awareness
TG 441 W	Condenser Bay West End T105; H2 Seal Oil Room T117; TG 441 West General Area T106	Initiate Fire Protection Evaluation Permit per PPM 1.3.10 to: - implement hourly fire tour for H2 Seal Oil Room T117 and TG 441 west general area T106 per FPP-1.7 - ensure no unnecessary fire hazards and no obstructed fire equipment in H2 Seal Oil Room T117 and TG 441 west general area T106 - confirm area detection, suppression and fire doors available for Condenser bay west end T105; H2 seal oil room T117; and TG 441 west general area T106 - prohibit hot work in Condenser bay west end T105; H2 seal oil room T117; and TG 441 west general area T106 - update plant status page with plant fire risk status and affected fire area for station awareness
TG 471 Center	TG 471 Heater Bay	Initiate Fire Prevention Evaluation Permit per PPM 1.3.10 to: - confirm Operations shiftly camera tour of the TG 471 Heater Bay - ensure no unnecessary fire hazards and no obstructed fire equipment in TG 471 Heater Bay (via camera tour) - confirm area detection, suppression and fire doors available for TG 471 Heater Bay - prohibit hot work in TG 471 Heater Bay - update plant status page with plant fire risk status and affected fire area for station awareness

- Increase operator awareness of the increased significance of these actions by performing an operator briefing prior to taking RHR A out of service for the extended completion time:

    - OPERATOR FAILS TO VENT CONTAINMENT

    - OPERATOR FAILS TO DEPRESSURIZE THE RPV

    - FAILURE TO ALIGN RHR TO SUPPRESSION POOL COOLING

    - OPERATOR FAILS TO CONNECT FIREWATER to CONDENSATE POSTCONTAINMENT FAILURE

    - OPERATOR FAILS TO CONNECT SW TO RHR-B POST CONTAINMENT FAILURE

- Verify hazardous weather conditions are not forecasted.

The following conclusions were reached as a result of this analysis:

- All quantitative results for ICCDP and ICLERP for the CT extension application are less than the guidance thresholds for:

    1. RHR A case OOS (see result in Table 3-1)
    2. RHR A OOS with only protected trains required per PPM 1.3.83. (See result in Table 3-2)
    3. RHR A OOS with additional compensatory measures in place (See results in Table B-1).

- Additional compensatory measures are proposed and have not been directly quantified in the PRA model, but are judged to further reduce the risk for this plant configuration.

- None of the PRA uncertainties identified for this application were found to be key uncertainties that influence this RHR CT extension, based on the guidance provided by EPRI [Ref. 7].

- The PRA models are adequate to support this risk assessment and the resulting risk is acceptable and consistent with the NRC safety goals.

8. References

1. Regulatory Guide 1.174, Rev. 2, 2011.
2. Regulatory Guide 1.177, Rev. 1, 2011.
3. *Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty.* EPRI, Palo Alto, CA: 2012. 1026511.
4. Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic risk Assessment Results for Risk-informed Activities”, Office of Nuclear Regulatory Research, Rev. 2, March 2009.

5. ASME/ANS RA-Sa–2009, Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, 2009.
6. Quantification, PSA-2-QU-0001, Rev. 6, 2016.
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8. CGS RG 1.200 compliance database
9. PPM 1.3.83, Protected Equipment Program, Rev. 21, 7/18/16.
10. Updated CCF Data for CGS, PSA-2-DA-0004, Rev. 5, 2016.
11. PSA Internal Event Summary, PSA-1-SM-0001, Rev. 7, 2016.
12. Staff Evaluation Report of Individual Plant Examination of External Events (IPEEE) Submittal on Columbia Generating Station, Energy Northwest, Docket No. 50-397, February 26, 2001.
13. PPM 1.3.85, "ON-LINE FIRE RISK MANAGEMENT" Rev. 5

Attachment A

Data Inputs and Analyses

Table A-1 **PRA Rev. 7.2.1 Data for RHR Pump CCF Failure to Start**				
MDP FTS Clean Systems	Alpha Factor	Adjustment per CCF methodology	Final	CCF
2/2	2.19E-02	1	2.19E-02	1.94E-05

Calculated with CCFWin 4.0.2007.5 [Ref. 10]

Table A-2 **PRA Rev. 7.2.1 Data for CCF Failure to Run**				
MDP FTS Clean Systems	Alpha Factor	Adjustment per CCF methodology	Final	CCF
2/2	1.65E-02	1	1.65E-02	5.30E-06

Calculated with CCFWin 4.0.2007.5 [Ref. 10]

Table A-3 **RHR A CCF Adjustments**				
Event ID	Description	Point Estimate for Base Case	Value Used	Notes
RHRP-MDBCFTRC2LL	CCF FAIL TO RUN RHR P-2B AND RHR P-2C	1.35E-06	5.30E-06	Since RHR A OOS, compute CCF as failure 2 of 2 to run: 5.3E-6 [per Table A-2 in Attachment A].
RHRP-MDBCFTSC2LL	CCF OF PUMPS 2B AND 2C FAILTO START	4.04E-06	1.94E-05	Since RHR A OOS, compute CCF as failure 2 of 2 to start: 1.94E-5 [per Table A-1 in Attachment A].

Table A-4
Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case – Internal Events

Event	Probability	Fus Ves	BirnBm	Description
CF-FAILS-HPCS-INJ	8.30E-02	2.18E-01	2.93E-05	HPCS INJECTION FAILS DUE TO CONTAINMENT FAILURE
ADSHUMNSTARTH3LT	3.30E-04	1.77E-01	6.15E-03	OPERATOR FAILS TO DEPRESSURIZE THE RPV (Transient)
VENTFAIL	7.30E-05	1.52E-01	2.37E-02	OPERATOR FAILS TO VENT CONTAINMENT
HPS---------T3LL	8.76E-03	1.48E-01	1.93E-04	HPCS UNAVAILABILITY DUE TO T & M (MRULE DATA)
RHRHUMNSP-COOLLL	1.00E-05	1.07E-01	1.23E-01	FAILURE TO ALIGN RHR TO SUPPRESSION POOL COOLING
EACENG-EDG3-S424	3.95E-02	1.02E-01	2.97E-05	EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN FOR 24H
IE-FLD-TLO--MS-S	8.58E-04	9.24E-02	1.24E-03	SMALL MS LEAKS IN TURB BLDG - LOCA OUTSIDE CONTNMENT
IE-TF	1.60E-01	8.27E-02	5.94E-06	LOSS OF FEEDWATER FREQUENCY IN EVENTS PER REACTOR YEAR
RHRHUMNSYS62H3LL	1.60E-04	8.20E-02	5.87E-03	FAILURE TO ALIGN RHR TO SUPPRESSION POOL COOLING
IE-TC	1.50E-01	7.74E-02	5.93E-06	LOSS OF CONDENSER FREQUENCY IN EVENTS PER REACTOR YEAR
IE-FLD-TLO--MS-U	5.61E-04	6.96E-02	1.43E-03	MOD MS LEAKS IN TURB BLDG - LOCA OUTSIDE CONTNMENT
CM	2.15E-06	6.51E-02	3.33E-01	MECHANICAL FAILURE OF SCRAM SYSTEM NUREG/CR-5500
FP-HUMNSYS62H3LL	6.50E-03	6.49E-02	1.15E-04	OPERATOR FAILS TO CONNECT FIREWATER TO CONDENSATE POST CONTAINMENT FAILURE
RHRHUMNSWCRTIEXX	6.80E-04	6.38E-02	1.08E-03	OPERATOR FAILS TO CONNECT SW TO RHR-B POST CONTAINMENT FAILURE
PRAAHUS--1B-S3LL	9.01E-04	6.17E-02	7.54E-04	FAN PRA-FN-1B DOES NOT START ON DEMAND
SW-P-MDSWP1BS3LB	8.84E-04	6.05E-02	7.54E-04	FAILURE OF SSW PUMP MOTOR TO START ON DEMAND, MECHANICAL
HPSHUMN-CTL-HNAT	4.20E-03	6.02E-02	1.65E-04	FAILURE TO CONTROL HPCS INJECTION (NON-ATWS)
HPS-CTL-COND----	6.82E-03	6.02E-02	1.01E-04	HPCS CONTROL REQUIR ED
IE-FLD-TLO--RFWS	4.77E-04	4.86E-02	1.17E-03	SMALL RFW LEAKS IN TURB BLDG - LOCA OUTSIDE CONTNMENT
ATWH-HPLPRSTH3XX	1.00E+00	4.68E-02	5.38E-07	OP. CNTRLS RPV LVL ADQUTLY (NOT TOO LOW) -- ATWS W/HPCS AVAL
INIT-RY-TF	3.11E+02	4.45E-02	1.64E-09	REACTOR YEAR CONVRSN - LOSS OF RFW IE GROUP
IE-FLD-T106CONDS	2.71E-03	4.40E-02	1.86E-04	SMALL COND BREAK IN T106
RHRH-ATWSDC-H3XX	4.50E-01	4.36E-02	1.11E-06	OPERATOR FAILS TO BYPASS RHR-SDC INTERLOCKS (ATWS)
CRDHUMNALTCLH3LL	2.00E-02	4.35E-02	2.50E-05	OP FAILS TO ALIGN ALTERNATE COOLING TO CRD

Table A-4
Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case – Internal Events

Event	Probability	Fus Ves	BirnBm	Description
HPSV-MO----4P2LL	2.43E-03	4.25E-02	2.00E-04	HPCS-V-4 MO GATE VALVE NC-FTO
HPSV-MO---12P2LL	2.43E-03	4.25E-02	2.00E-04	HPCS-V-12, MIN FLOW PROTECTION VALVE NC-FTO ON DEMAND
IE-FLD-T106CONDU	2.52E-03	4.09E-02	1.86E-04	MODERATE OR MAJ COND BREAK IN T106/107/108/109
RRAAHUSRFC03S3D2	9.01E-04	3.98E-02	4.95E-04	AHU RRA-FN-03 DOES NOT START ON DEMAND
RHRP-MD---2BS3LL	8.84E-04	3.90E-02	4.95E-04	RHR-P-2B MOTOR DRIVEN PUMP FAILS TO START
IE-FLD-TLO--MS-M	3.16E-04	3.74E-02	1.36E-03	LARGE MS LEAKS IN TURB BLDG - LOCA OUTSIDE CONTNMENT
EACEDG-3----T3D3	1.27E-02	3.65E-02	3.29E-05	DG-3 OUT FOR MAINTENANCE (MRULE DATA)
TE	2.98E-02	3.60E-02	1.39E-05	LOSS OF OFF-SITE POWER FREQUENCY IN EVENTS PER YEAR
INIT-RY-LOSP	8.50E-01	3.60E-02	4.87E-07	CONVERTS CRITICAL YEARS TO RX YEARS INIT EVENTS
INIT-RY-MS	3.11E+02	3.50E-02	1.29E-09	REACTOR YEAR CONVERSION - MAN SHUTDOWN IE GROUP
MOC-RHRB-F	4.00E-04	3.50E-02	9.54E-04	MOC ASS'Y FAILURE FOR RHR B FAILURE
IE-TIA	3.60E-02	3.31E-02	1.06E-05	LOSS OF NON-SAFETY CIA INIT EVENT PER REACTOR YEAR
DMATE-----32W2LL	1.85E-03	3.22E-02	1.99E-04	TEMPERATURE SENSOR FOR DAMPER 32 LOSS OF FUNCTION
IE-FLD-TLO--RFWU	3.12E-04	3.17E-02	1.17E-03	MOD RFW LEAKS IN TURB BLDG - LOCA OUTSIDE CONTNMENT
CRDHUMNLTINJH3XX	5.40E-05	2.91E-02	6.18E-03	OP FAILS TO ALIGN STANDBY CRD TRAIN FOR LONG TERM INJ
IE-FLD-T106TSW-S	1.72E-03	2.79E-02	1.86E-04	SMALL TSW BREAK IN T106
RCI---------T3LL	2.41E-02	2.53E-02	1.21E-05	RCIC UNAVAILABILITY DUE TO T & M (MRULE DATA)
TSWHUMNIC525H3LL	1.00E+00	2.51E-02	2.89E-07	OP FAILS TO ISOL MAJOR TSW LEAK ON RAD/CNTRL BLDG ELEV 525
INIT-RY-TM	3.11E+02	2.26E-02	8.33E-10	REACTOR YEAR CONVRSN - MSIV CLOSURE IE GROUP
SW-OPER	1.12E-01	2.24E-02	2.26E-06	STANDBY SERVICE WATER OPERATING
SW-P-MDSWP1BS4LB	3.21E-04	2.17E-02	7.43E-04	FAILURE OF SSW PUMP MOTOR TO KEEP RUN- NING FOR 24 HR
OP-RHR-MNFL	1.00E+00	2.01E-02	2.31E-07	FAILURE TO DIAGNOSE ECCS MIN FLOW VALVES FAILED CLOSED
EDCC1--EC1-7W4D1	1.24E-04	1.97E-02	1.83E-03	125V DC BATTERY CHARGER C1-7 FAILS
CF-FAIL-CRDH	6.60E-01	1.96E-02	3.41E-07	CRDH INJECTION FAILS DUE TO CONTAINMENT FAILURE
IE-TT	7.00E-01	1.94E-02	3.19E-07	CGS TURBINE TRIP FREQUENCY IN EVENTS PER YEAR
F-RHR-MOCSTRT	1.00E+00	1.91E-02	2.19E-07	CONDITIONAL PROBABILITY OF SW START
OP-ECCS-SW	1.00E+00	1.91E-02	2.19E-07	FAILURE OF OP TO DIAGNOSE LACK OF SW FOR ECCS PUMP
EACENG-EDG3-R3D3	6.36E-03	1.91E-02	3.44E-05	EMERGENCY DG-3 DOES NOT START
IE-FLD-RLO-RWCUU	4.65E-04	1.89E-02	4.67E-04	MOD RWCU LK REACTOR BLDG - LOCA OUTSIDE CONTNMENT
HPSRMS----S2W2LL	1.08E-03	1.83E-02	1.95E-04	HPCS-RMS-P/1 (E22B-S2) SWITCH FAILURE

Table A-4
Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case – Internal Events

Event	Probability	Fus Ves	BirnBm	Description
SWB-XHE-RE-RHRSW	2.70E-04	1.82E-02	7.40E-04	HUMAN ERROR TO RESTORE SW B TRAIN FROM TEST AND MAINT
IE-FLD-TLO--RFWM	1.76E-04	1.78E-02	1.17E-03	LARGE RFW LEAKS IN TURB BLDG - LOCA OUTSIDE CONTNMENT
EACTRL-S----T3--	1.46E-03	1.76E-02	1.38E-04	TRANSFORMER TR-S OUT FOR MAINTENANCE (MRULE DATA)
RRARMS-RFC03W2D2	3.59E-04	1.58E-02	4.93E-04	FAILURE OF CONTACT 1-2 ON MANUAL SWICH RRA-RMS-FN/03
RRAAHUSRFC04S3D3	9.01E-04	1.52E-02	1.94E-04	AHU RRA-FN-04 DOES NOT START ON DEMAND
HPSP-MD----1S3LL	8.84E-04	1.50E-02	1.94E-04	HPCS PUMP FAILS TO START
SW-P-MDHPSP2S3LC	8.84E-04	1.50E-02	1.94E-04	HPCS-P-2 SSW PUMP MOTOR FAILS TO START ON DEMAND
ADS-XHE-FO-S2W	5.50E-04	1.49E-02	2.95E-04	OPERATOR FAILS TO DEPRESSURIZE THE RPV (Small Water LOCA)
RHRP-MD---2BS4LL	3.21E-04	1.41E-02	4.92E-04	RHR-P-2B MOTOR DRIVEN PUMP FAILS TO RUN 24 HRS
SW-V-MODV-29P2LC	8.11E-04	1.37E-02	1.93E-04	MECHANICAL FAILURE OF MOTOR ACTUATED VALVE SW-V-29
PTT	2.20E-02	1.35E-02	7.04E-06	FAILURE OF SRV'S RECLOSING FOR TT AND TF INITIATORS
IE-MS	1.10E+00	1.35E-02	1.39E-07	MANUAL SHUTDOWN FREQUENCY IN EVENTS PER YEAR
IE-TM	2.90E-02	1.32E-02	5.24E-06	MSIV CLOSURE FREQUENCY IN EVENTS PER REACTOR YEAR
RCITDP-----1R3LL	9.54E-03	1.31E-02	1.58E-05	RCIC TDP PUMP FAILS TO START
IE-SR	1.00E-02	1.30E-02	1.49E-05	CGS RX LEVEL INSTRUMENT LINE BRK FREQUENCY,EVENTS/YR
OP-SW-PMP	3.70E-01	1.29E-02	4.01E-07	FAILURE TO DIAGNOSE DEAD HEAD OPS BEFORE PUMP FAILURE
MS-HUMNP413-C3LL	9.50E-04	1.28E-02	1.54E-04	COMMON CAUSE HUMAN ERROR OF MISCAL & MISALIGN MS-PS-413ABCD
SW-V-MODV-2BP2LB	1.89E-04	1.26E-02	7.35E-04	MOV SW-V-2B FAILS TO OPEN ON DEMAND
SW-V-MOV-12BP2LB	1.89E-04	1.26E-02	7.35E-04	MOV SW-V-12B FAILS TO OPEN ON DEMAND
IE-FLD-C507TSW-M	1.39E-06	1.26E-02	1.04E-01	MAJOR TSW BREAK IN C507
IE-FLD-C508TSW-M	1.39E-06	1.26E-02	1.04E-01	MAJOR TSW BREAK IN C508
IE-FLD-RLO-RWCUS	4.55E-04	1.25E-02	3.15E-04	SMALL RWCU LEAK IN REACTOR BLDG - LOCA OUTSIDE CONTNMENT
RHRHUMN-SDC-H3XX	1.10E-01	1.20E-02	1.25E-06	OPERATOR FAILS TO BYPASS RHR-SDC INTERLOCKS (NON-ATWS)
SW-STR-SST3BE4LB	1.77E-04	1.18E-02	7.34E-04	SW PUMP 1B MOTOR BEARING STRAINER SW-ST-3B PLUGGED
NRAC24	3.23E-03	1.09E-02	3.89E-05	NO RECOVERY OF OFFSITE POWER WITHIN 24 HOURS
EACTR--7-73-W4D1	2.17E-05	1.07E-02	5.66E-03	TRANSFORMER TR-7-73 LOSS OF FUNCTION

Table A-4 Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case – Internal Events				
Event	Probability	Fus Ves	BirnBm	Description
ADS-XHE-FO-SORV	3.40E-04	1.03E-02	3.48E-04	OPERATOR FAILS TO DEPRESSURIZE THE RPV (SORV)
SW-HUMNIC525H3LL	1.00E+00	1.03E-02	1.18E-07	OP FAILS TO ISOL MAJOR SW LEAK ON RAD/CNTRL BLDG 525 ELEV

Table A-5 **Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case - Fire**				
Event	Probability	FusVes	BirnBm	Description
LZT12	3.92E-06	2.20E-01	9.25E-01	TG-12 Full-PAU Burnup - Initiator
HS-RHRV-MO-23	3.00E-01	1.24E-01	6.74E-06	RHR-V-23 FAILURE CAUSED BY HOT SHORT
HPS---------T3LL	8.76E-03	1.12E-01	2.08E-04	HPCS UNAVAILABILITY DUE TO T & M (MRULE DATA)
EACENG-EDG3-S424	3.95E-02	9.07E-02	3.76E-05	EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN FOR 24H
T3W07	4.74E-04	7.40E-02	2.58E-03	RC-7 Detailed Scenario 3 - Initiator
F4R1J	1.32E-04	6.35E-02	7.93E-03	R-1J Detailed Scenario 4 - Initiator
LZW03	1.02E-06	6.19E-02	1.00E+00	RC-3 Full-PAU Burnup - Initiator
ADSHUMNSTARTH3LT-F	6.60E-04	5.54E-02	1.35E-03	FIRE - OPERATOR FAILS TO DEPRESSURIZE THE RPV (Transient)
HS-EAC-TRS	3.00E-01	5.12E-02	2.78E-06	HOT SHORT DISABLES TR-S
TZT1C	2.49E-03	4.89E-02	3.24E-04	TG-1C Full-PAU Burnup - Initiator
EAC-CONSEQ-LOOP	1.00E-03	4.40E-02	7.00E-04	CONSEQUENTIAL LOOP FOLLOWING PLANT TRIP
F3W13	5.20E-05	3.64E-02	1.15E-02	RC-13 Detailed Scenario 3 - Initiator
LAW14	1.07E-04	3.54E-02	5.46E-03	RC-14 Detailed Scenario A - Initiator
LPS---------T3LL	5.12E-03	3.51E-02	1.08E-04	LPCS UNAVAILABILITY DUE TO MAINTENANCE (MRULE DATA)
HPSV-MO----4P2LL	2.43E-03	3.38E-02	2.25E-04	HPCS-V-4 MO GATE VALVE NC-FTO
HPSV-MO---12P2LL	2.43E-03	3.38E-02	2.25E-04	HPCS-V-12, MIN FLOW PROTECTION VALVE NC-FTO ON DEMAND
HS-ADS-OPEN	3.00E-01	3.19E-02	1.74E-06	ADS VALVE(S) STUCK OPEN DUE TO HOT SHORT
CF-FAILS-HPCS-INJ	8.30E-02	3.08E-02	6.05E-06	HPCS INJECTION FAILS DUE TO CONTAINMENT FAILURE
MS-HUMNP413-C3LL	9.50E-04	3.03E-02	5.04E-04	COMMON CAUSE HUMAN ERROR OF MISCAL & MISALIGN MS-PS-413ABCD
EACEDG-3----T3D3	1.27E-02	2.96E-02	3.82E-05	DG-3 OUT FOR MAINTENANCE (MRULE DATA)
T5W07	7.37E-04	2.73E-02	6.11E-04	RC-7 Detailed Scenario 5 - Initiator
F4W08	1.32E-03	2.64E-02	3.30E-04	RC-8 Detailed Scenario 4 - Initiator
DMATE-----32W2LL	1.85E-03	2.56E-02	2.24E-04	TEMPERATURE SENSOR FOR DAMPER 32 LOSS OF FUNCTION
F6R1J	5.11E-05	2.43E-02	7.83E-03	R-1J Detailed Scenario 6 - Initiator

Table A-5
Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case - Fire

Event	Probability	FusVes	BirnBm	Description
EACENG-EDG2-S4D2	1.00E-02	2.34E-02	3.72E-05	EMERGENCY DG-2 DOES NOT CONTINUE TO RUN FOR 6 HOURS
LPSV-MO----5P5LL	2.62E-03	2.33E-02	1.41E-04	LPCS-V-5, MOV, FAILS TO OPEN (ANNUALLY TESTED)
F4R1D	6.37E-05	2.24E-02	5.79E-03	R-1D Detailed Scenario 4 - Initiator
LPSV-MO---11P2LL	2.43E-03	2.15E-02	1.40E-04	LPCS-FCV-11, MIN. FLOW PROTECTION VLV ,NC-FTO ON DEMAND
OP-SW-PMP-F	1.00E+00	2.14E-02	3.52E-07	FIRE - FAILURE TO DIAGNOSE DEAD HEAD OPS BEFORE PUMP FAILURE
F6W04	6.36E-04	2.11E-02	5.48E-04	RC-4 Detailed Scenario 6 - Initiator
FZR1B	3.76E-05	2.08E-02	9.11E-03	R-1B Full-PAU Burnup - Initiator
PRAAHUS--1B-S3LL	9.01E-04	2.06E-02	3.74E-04	FAN PRA-FN-1B DOES NOT START ON DEMAND
SW-P-MDSWP1BS3LB	8.84E-04	2.03E-02	3.74E-04	FAILURE OF SSW PUMP MOTOR TO START ON DEMAND, MECHANICAL
T4W07	5.16E-04	1.90E-02	6.08E-04	RC-7 Detailed Scenario 4 - Initiator
F3R1D	4.83E-05	1.90E-02	6.48E-03	R-1D Detailed Scenario 3 - Initiator
RHRHUMN-SDC-H3XX-F	4.40E-01	1.82E-02	6.79E-07	FIRE - OPERATOR FAILS TO BYPASS RHR-SDC INTERLOCKS (NON-ATWS)
EACENG-EDG3-R3D3	6.36E-03	1.69E-02	4.33E-05	EMERGENCY DG-3 DOES NOT START
EACTRL-S----T3--	1.46E-03	1.61E-02	1.77E-04	TRANSFORMER TR-S OUT FOR MAINTENANCE (MRULE DATA)
HPS-CTL-COND----	6.82E-03	1.60E-02	3.86E-05	HPCS CONTROL REQUIR ED
HPSHUMN-CTL-HNAT-F	8.40E-03	1.57E-02	3.07E-05	FIRE - FAILURE TO CONTROL HPCS INJECTION (NON-ATWS)
MS-HUMNP413CX3LL	1.70E-03	1.49E-02	1.39E-04	MS-PS-413C MISCALIBRATION
EACENG-EDG2-R3D2	6.36E-03	1.48E-02	3.70E-05	EMERGENCY DG-2 DOES NOT START
HPSRMS----S2W2LL	1.08E-03	1.47E-02	2.21E-04	HPCS-RMS-P/1 (E22B-S2) SWITCH FAILURE
ADSHUMNSTARTH3LT	3.30E-04	1.38E-02	6.79E-04	OPERATOR FAILS TO DEPRESSURIZE THE RPV (Transient)
F3W04	4.19E-04	1.37E-02	5.37E-04	RC-4 Detailed Scenario 3 - Initiator
MOC-SWB-F	4.00E-04	1.30E-02	4.60E-04	MOC ASSY FAILS FOR SWB
F7W04	3.93E-04	1.28E-02	5.37E-04	RC-4 Detailed Scenario 7 - Initiator
RRAAHUSRFC04S3D3	9.01E-04	1.22E-02	2.20E-04	AHU RRA-FN-04 DOES NOT START ON DEMAND
HPSP-MD----1S3LL	8.84E-04	1.20E-02	2.20E-04	HPCS PUMP FAILS TO START
SW-P-MDHPSP2S3LC	8.84E-04	1.20E-02	2.20E-04	HPCS-P-2 SSW PUMP MOTOR FAILS TO START ON DEMAND

Table A-5 **Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case - Fire**				
Event	Probability	FusVes	BirnBm	Description
F5R1J	2.33E-05	1.10E-02	7.81E-03	R-1J Detailed Scenario 5 - Initiator
SW-V-MODV-29P2LC	8.11E-04	1.10E-02	2.19E-04	TG-12 Full-PAU Burnup - Initiator
RHRHUMNSYS62H3LL	1.60E-04	1.02E-02	1.04E-03	RHR-V-23 FAILURE CAUSED BY HOT SHORT
OP-RHR-ALGN	1.00E+00	1.01E-02	1.66E-07	HPCS UNAVAILABILITY DUE TO T & M (MRULE DATA)
OP-RHR-MNFL	1.00E+00	1.01E-02	1.66E-07	EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN FOR 24H
FZT1D	3.49E-03	1.00E-02	4.73E-05	RC-7 Detailed Scenario 3 - Initiator

Table A-6
Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case - Seismic

Event	Probability	FusVes	BirnBm	Description
SDS2	6.66E-05	4.36E-01	5.33E-02	SDS 2 BOP CST LOOP SS LOCA - Initiator
NRAC24H-SEIS	7.30E-01	4.19E-01	4.61E-06	No Recovery of Offsite AC w/in 24 hrs. (Seismic)
SEIS-MITGTN-FAIL	1.00E+00	2.93E-01	2.39E-06	NO ADDITIONAL MITIGATION POSSIBLE (SEISMIC SCENARIO)
SDS42	2.38E-06	2.92E-01	1.00E+00	SDS 42 Failure of RPV and/or Category I Bldgs. - Initiator
EACENG-EDG2SS4D2	4.00E-02	2.58E-01	5.01E-05	EMERGENCY DG-2 DOES NOT CONTINUE TO RUN FOR 6 HOURS 24 HRS - SEISMIC
SDS41	1.64E-06	1.51E-01	7.50E-01	SDS 41 Wide-spread failure of SSEL equipment - Initiator
SEIS-LONGTERM	5.00E-01	1.01E-01	8.20E-07	PROB OF LONG TERM CORE DAMAGE SEQ FOR SDS41
SEIS-SHRTTERM	5.00E-01	1.01E-01	8.20E-07	PROB OF SHORT TERM CORE DAMAGE SEQ FOR SDS41
EACENG-EDG2-S4D2	1.00E-02	6.46E-02	4.90E-05	EMERGENCY DG-2 DOES NOT CONTINUE TO RUN FOR 6 HOURS
EACENG-EDG2-R3D2	6.36E-03	4.10E-02	4.89E-05	EMERGENCY DG-2 DOES NOT START
CF-FAILS-HPCS-INJ	8.30E-02	2.06E-02	1.50E-06	HPCS INJECTION FAILS DUE TO CONTAINMENT FAILURE
S523	1.40E-07	1.72E-02	1.00E+00	SDS 5, SDS 23 BOP CST LOOP SS LOCA EDG 1 and 2 Div. III - Initiator
SLAC	1.08E-07	1.33E-02	1.00E+00	SDS17, SDS19, SDS35 BOP CST LOOP MLOCA EDG 1 and 2 Div. III OSP Not Recov - Init
S725	1.02E-07	1.25E-02	1.00E+00	SDS 7, SDS 25 BOP CST LOOP SS LOCA Div. I and II Div. III OSP Not Recov - Initia
DMATE-----21W2LL	1.85E-03	1.20E-02	4.87E-05	TEMPERATURE SENSOR FOR DAMPER 21 LOSS OF FUNCTION
DMATE-----22W2LL	1.85E-03	1.20E-02	4.87E-05	TEMPERATURE SENSOR FOR DAMPER 22 LOSS OF FUNCTION
SDS38	9.45E-08	1.16E-02	1.00E+00	SDS 38 BOP CST LOOP N2 Tank EDGs stalled and not re-started - Initiator
SDS20	1.77E-06	1.15E-02	5.31E-02	SDS 20 BOP CST LOOP N2 Tank SS LOCA - Initiator
SDS4	5.43E-07	1.11E-02	1.67E-01	SDS 4 BOP CST LOOP SS LOCA EDG 1 and 2 - Initiator
EACENG-EDG3-S424	3.95E-02	1.05E-02	1.78E-06	EMERGENCY DG SYSTEM DOES NOT CONTINUE TO RUN FOR 24H
S624	5.08E-07	1.03E-02	1.64E-01	SDS 6, SDS 24 BOP CST LOOP SS LOCA Div. I and II OSP Not Recov - Initiator

Table A-6
Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case - Seismic

Event	Probability	FusVes	BirnBm	Description
S2P3	4.57E-07	1.02E-02	1.81E-01	SDS 2 SDS 2 BOP CST LOOP SS LOCA SBO WITHOUT RCIC - Initiator
PRAAHUS--1B-S3LL	9.01E-04	7.73E-03	6.62E-05	FAN PRA-FN-1B DOES NOT START ON DEMAND
SW-P-MDSWP1BS3LB	8.84E-04	7.59E-03	6.62E-05	FAILURE OF SSW PUMP MOTOR TO START ON DEMAND, MECHANICAL
N24-AVE-SEIS	8.54E-01	6.11E-03	5.04E-08	Average NRAC for First 24 Hrs. (Seismic)
DMAAHUS--21-S3LL	9.01E-04	5.81E-03	4.86E-05	FAN DMA-FN-21 DOES NOT START ON DEMAND (EMERGENCY)
EACP-MD-10B-S3LL	8.84E-04	5.70E-03	4.86E-05	OIL PUMP DO-P-1B FAILS TO START

Table A-7
Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case – Internal Events

Event	Probability	RAW	Description
IE-VESSEL-RUPTUR	1.00E-08	74807.474	REACTOR VESSEL RUPTURE
EACCB-7383SOC2LL	1.16E-07	67151.065	CIRCUIT BREAKER 7-73 AND 8-83 FTRC DUE TO CCF
CM	2.15E-06	28984.033	MECHANICAL FAILURE OF SCRAM SYSTEM NUREG/CR-5500
EDCC1--2A2B-C3LL	1.94E-07	19841.012	COMMON CAUSE FAIL. OF BATTERY CHARGERS C1-2A AND C1-2B
RHRHUMNSP-COOLLL	1.00E-05	10738.121	FAILURE TO ALIGN RHR TO SUPPRESSION POOL COOLING
IE-FLD-C507TSW-M	1.39E-06	9064.8	MAJOR TSW BREAK IN C507
IE-FLD-C508TSW-M	1.39E-06	9064.8	MAJOR TSW BREAK IN C508
IE-FLD-C205-FP-U	4.22E-07	9060.377	MODERATE FPW BREAK IN C205 C212 OR C214
IE-FLD-C304-FP-U	4.52E-08	9036.58	MOD OR MAJOR FPW BREAK IN C304
SW-V-MO2AB29C3LL	1.33E-06	7035.774	FAILURE OF DISCHARG MOV'S SW-2A, SW-2B AND SW-29(ATC 4/18)
SW-SCR-SCRN-C3LL	7.29E-07	6948.485	CCF BLOCKAGE OF ALL SW INTAKE SCREENS
RHRV-MO--8-9C3RR	7.88E-08	5540.107	COMMON CAUSE RUPTURE FAILURE OF RHR-V-8 AND -9
IE-DDC	1.80E-07	3390.263	LOSS OF DIV 1 AND DIV 2 DC POWER INITIATING EVENT
MTM	1.60E-08	2067.035	SUCCESS CRITERIA 11 SRV'S OPEN OUT OF 18 NOT MET
VENTFAIL	7.30E-05	2060.633	OPERATOR FAILS TO VENT CONTAINMENT
IE-FLD-C502TSW-U	2.52E-06	2030.965	MOD OR MAJOR TSW BREAK IN C502
MTT	1.60E-08	1530.639	SUCCESS CRITERIA 7 SRV'S OPEN OUT OF 18 NOT MET
IE-DAC	1.00E-07	1481.924	LOSS OF SM-7 AND SM-8 DUE TO CCF INITIATING EVENT
EACSM--8----W4D1	1.67E-06	1339.705	4160 VOLT BUS SM-8 LOSS OF FUNCTION
SW-SCR-S1BH2C2LL	1.16E-06	1155.104	CCF BLOCKAGE OF SW-P1B AND HPCS-P2 SCREENS
IE-FLD-C507SSWAM	4.89E-06	1050.471	MAJOR SW A BREAK IN C507
IE-FLD-C508SSWBM	4.89E-06	1050.471	MAJOR SW B BREAK IN C508

Table A-7
Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case – Internal Events

Event	Probability	RAW	Description
IE-FLD-C212SSWAU	2.05E-07	1044.998	MOD OR MAJ SW A BREAK IN C212 OR C509
IE-FLD-C212SSWBU	2.05E-07	1044.998	MOD OR MAJ SW B LEAK IN C212 OR C509
EDCC1--127--C3LL	3.78E-08	873.557	CCF BATTERY CHARGERS 1A, 1B 2A, 2B, AND 7
EDCC1--12---C3LL	3.50E-08	819.901	COMMON CAUSE FAIL. OF BATTERY CHARGERS C1-1A, 1B, 2A, 2B
EACSM--7----W4D1	1.67E-06	645.185	4160 VOLT BUS SM-7 LOSS OF FUNCTION
EDCPP--S17--W4LL	1.68E-06	634.214	FAILURE OF BUS E-DP-S1/7
CRDHUMNLTINJH3XX	5.40E-05	538.71	OP FAILS TO ALIGN STANDBY CRD TRAIN FOR LONG TERM INJ
EACCB--7-73-G4D1	4.00E-06	536.617	CIRCUIT BREAKER 7-73 FTRC
ADSHUMNSTARTH3LT	3.30E-04	535.694	OPERATOR FAILS TO DEPRESSURIZE THE RPV (Transient)
EACMC--7A---W4D1	1.67E-06	519.731	MOTOR CONTROL CENTER MC-7A LOSS OF FUNCTION
EACSL--73---W4D1	1.67E-06	519.731	480 VOLT AC BUS SL-73 LOSS OF FUNCTION
RHRHUMNSYS62H3LL	1.60E-04	511.635	FAILURE TO ALIGN RHR TO SUPPRESSION POOL COOLING
EACTR--7-73-W4D1	2.17E-05	493.655	TRANSFORMER TR-7-73 LOSS OF FUNCTION
EACCB--73-7AG4D1	4.00E-06	470.684	CIRCUIT BREAKER FROM SL-73 TO MC-7A FTRC
EDCB1--127--C3LL	2.27E-07	447.32	COMMON CAUSE FAIL OF E-B1-1, B1-2 AND B1-7
EDCB1--12721C3LL	2.12E-07	444.181	C C FAILURE OF ALL EXIDE MDL 2GN BATTS B1-1,-2,-7& B2-1
SP-FL-----BSI-LL	9.99E-06	306.931	CCF SUPP POOL STRAINERS FOR BOC IS - SM LOCAS
EACSM--2----W4D3	1.67E-06	265.595	4160 VOLT SM-2 LOSS OF FUNCTION
EACTR--663--W4--	2.17E-05	252.429	TRANSFORMER JUN-63 LOSS OF FUNCTION
EACTR--7-71-W4D1	2.17E-05	240.642	TRANSFORMER TR-7-71 LOSS OF FUNCTION
IE-FLD-W53ASSWAU	3.06E-06	229.732	MOD OR MAJOR SW A BREAK INSIDE WMA-AH-53A
IE-FLD-W52ASSWAU	3.69E-06	229.61	MOD OR MAJOR SW A BREAK INSIDE WMA-AH-52A
IE-FLD-W52BSSWBU	3.69E-06	229.61	MOD OR MAJOR SW B BREAK INSIDE WMA-AH-52B

Table A-7
Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case – Internal Events

Event	Probability	RAW	Description
IE-FLD-W53BSSWBU	3.06E-06	229.314	MOD OR MAJOR SW B BREAK INSIDE WMA-AH-53B
EACCB--636B-G4--	4.00E-06	224.904	CIRCUIT BREAKER FRM SL-63 TO MC-6B SPURIOUS TRIP
EACCB--663--G4--	4.00E-06	224.904	CIRCUIT BREAKER JUN-63 SPURIOUS TRIP
EACCB--7-71-G4D1	4.00E-06	223.189	CIRCUIT BREAKER 7-71 FTRC
SW-P-MD1A-1BC2	3.16E-05	218.061	COMON CAUSE FAILURE FOR SW-P-1A,B FAIL TO START & RUN

Table A-8 Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case - Fire			
Event	Probability	RAW	Description
L8W02	4.87E-08	60646.898	RC-2 Detailed Scenario 8 - Initiator
LZW03	1.02E-06	60646.84	RC-3 Full-PAU Burnup - Initiator
LZT12	3.92E-06	56109.622	TG-12 Full-PAU Burnup - Initiator
L5W02	3.60E-07	19735.674	RC-2 Detailed Scenario 5 - Initiator
L6W02	3.60E-07	8320.781	RC-2 Detailed Scenario 6 - Initiator
L2W1A	3.81E-07	2764.382	RC-1A Detailed Scenario 2 - Initiator
L3W1A	1.98E-07	2746.856	RC-1A Detailed Scenario 3 - Initiator
CM	2.15E-06	2347.621	MECHANICAL FAILURE OF SCRAM SYSTEM NUREG/CR-5500
SW-V-MO2AB29C3LL	1.33E-06	1997.988	FAILURE OF DISCHARG MOV'S SW-2A, SW-2B AND SW-29(ATC 4/18)
SW-SCR-SCRN-C3LL	7.29E-07	1993.822	CCF BLOCKAGE OF ALL SW INTAKE SCREENS
L4W10	3.56E-07	1919.841	RC-10 Detailed Scenario 4 - Initiator
F3W13	5.20E-05	700.647	RC-13 Detailed Scenario 3 - Initiator
F5W13	1.32E-05	693.995	RC-13 Detailed Scenario 5 - Initiator
F4W13	1.14E-05	692.962	RC-13 Detailed Scenario 4 - Initiator
F2W13	2.43E-06	674.983	RC-13 Detailed Scenario 2 - Initiator
F1W13	1.37E-06	664.867	RC-13 Detailed Scenario 1 - Initiator
FZR1B	3.76E-05	553.297	R-1B Full-PAU Burnup - Initiator
F4R1J	1.32E-04	481.59	R-1J Detailed Scenario 4 - Initiator
F1R1J	1.70E-05	478.925	R-1J Detailed Scenario 1 - Initiator
F6R1J	5.11E-05	475.601	R-1J Detailed Scenario 6 - Initiator
F5R1J	2.33E-05	474.367	R-1J Detailed Scenario 5 - Initiator
F2R1J	1.45E-05	473.455	R-1J Detailed Scenario 2 - Initiator
FZR1K	2.19E-05	453.614	R-1K Full-PAU Burnup - Initiator
F1R1D	1.81E-05	438.435	R-1D Detailed Scenario 1 - Initiator
F3R1D	4.83E-05	393.718	R-1D Detailed Scenario 3 - Initiator
F1W10	4.78E-06	377.874	RC-10 Detailed Scenario 1 - Initiator
F4R1D	6.37E-05	352.324	R-1D Detailed Scenario 4 - Initiator
F2R1D	1.00E-05	351.517	R-1D Detailed Scenario 2 - Initiator
FZM09	7.32E-06	351.255	M-9 Full-PAU Burnup - Initiator
SW-SCR-S1BH2C2LL	1.16E-06	341.157	CCF BLOCKAGE OF SW-P1B AND HPCS-P2 SCREENS
L7W02	3.57E-07	332.718	RC-2 Detailed Scenario 7 - Initiator

Table A-8
Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case - Fire

Event	Probability	RAW	Description
LAW14	1.07E-04	331.973	RC-14 Detailed Scenario A - Initiator
F1W1A	1.17E-05	304.062	RC-1A Detailed Scenario 1 - Initiator
F6W1A	2.55E-06	293.836	RC-1A Detailed Scenario 6 - Initiator
F4W1A	3.96E-07	269.623	RC-1A Detailed Scenario 4 - Initiator
L2W10	9.59E-07	249.618	RC-10 Detailed Scenario 2 - Initiator
EAC-SWA-SWB-CCF	4.00E-05	202.572	CCF OF SWA/SWB
SW-SCR-S1AH2C2LL	1.16E-06	176.281	CCF BLOCKAGE OF SW-P1A AND HPCS-P2 SCREENS
EACEDG-123FRC3LL	5.43E-05	169.639	CCF OF ALL 3 DG FAIL TO RUN
EACEDG-123FSC3LL	2.32E-05	169.272	CCF OF ALL 3 DG FAIL TO START
EACM-S-10AB2C3LL	1.40E-05	169.081	COMMON CAUSE FAIL- URE OF OIL PUMPS 10A, 10B, AND 2 TO START
EACM-R-10AB2C3LL	8.48E-07	166.346	COMMON CAUSE FAIL- URE OF OIL PUMPS 10A, 10B, AND 2 TO RUN
EACCB--DG123C3LL	4.41E-07	165.931	CCF OF OUTPUT BRKRS ON ALL THREE DG'S DG1,DG2,DG3
T3W07	4.74E-04	157.125	RC-7 Detailed Scenario 3 - Initiator
SW-P-MD1A-1BC2	3.16E-05	152.123	COMON CAUSE FAILURE FOR SW-P-1A,B FAIL TO START & RUN
SW-STR-ST3ABC2E4	1.19E-05	148.241	CCF OF SW-P-1A & 1B MOTOR BEARING STRAINERS
T2W07	1.37E-05	146.925	RC-7 Detailed Scenario 2 - Initiator
SW-SCR-S1A1BC2LL	1.16E-06	131.379	CCF BLOCKAGE OF SW-P-1A AND -1B SCREENS
SW-V-MO12A-BC3	1.15E-06	131.267	COMON CAUSE FAILURE FOR POND RETRN VLVE SW-12 TO OPEN
Event	Probability	Ach W	Description
L8W02	4.87E-08	60646.898	RC-2 Detailed Scenario 8 - Initiator

Table A-9 Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case - Seismic			
Event	Probability	RAW	Description
S1129	1.76E-08	122820.635	SDS 11, SDS 29 BOP CST LOOP SLOCA EDG 1 and 2 Div. III - Initiator
S1331	1.63E-08	122820.635	SDS 13, SDS 31 BOP CST LOOP SLOCA Div. I and II Div. III OSP Not Recov - Initiat
S21P2	1.25E-09	122820.635	SDS 21 BOP CST LOOP N2 Tank SS LOCA Div. III SBO - Initiator
S3P2	4.04E-09	122820.635	SDS 3 BOP CST LOOP SS LOCA Div. III SBO - Initiator
SDS38	9.45E-08	122820.635	SDS 38 BOP CST LOOP N2 Tank EDGs stalled and not re-started - Initiator
SDS40	7.93E-09	122820.635	SDS 40 Seismic Failure to Scram and Failure to Mitigate - Initiator
S523	1.40E-07	122820.621	SDS 5, SDS 23 BOP CST LOOP SS LOCA EDG 1 and 2 Div. III - Initiator
S725	1.02E-07	122820.621	SDS 7, SDS 25 BOP CST LOOP SS LOCA Div. I and II Div. III OSP Not Recov - Initia
SLAC	1.08E-07	122820.621	SDS17, SDS19, SDS35 BOP CST LOOP MLOCA EDG 1 and 2 Div. III OSP Not Recov - Init
SDS42	2.38E-06	122820.342	SDS 42 Failure of RPV and/or Category I Bldgs. - Initiator
SDS41	1.64E-06	92115.579	SDS 41 Wide-spread failure of SSEL equipment - Initiator
S2P3	4.57E-07	22231.672	SDS 2 SDS 2 BOP CST LOOP SS LOCA SBO WITHOUT RCIC - Initiator
S20P3	1.15E-08	21592.012	SDS 20 BOP CST LOOP N2 Tank SS LOCA SBO WITHOUT RCIC - Initiator
SDS4	5.43E-07	20484.567	SDS 4 BOP CST LOOP SS LOCA EDG 1 and 2 - Initiator
SDS22	1.83E-07	20468.614	SDS 22 BOP CST LOOP N2 Tank SS LOCA EDG 1 and 2 - Initiator
S624	5.08E-07	20203.36	SDS 6, SDS 24 BOP CST LOOP SS LOCA Div. I and II OSP Not Recov - Initiator
S1230	4.28E-08	20142.901	SDS 12, SDS 30 BOP CST LOOP SLOCA Div. I and II OSP Not Recov - Initiator
S1836	4.71E-08	20142.901	SDS 18, SDS 36 BOP CST LOOP MLOCA Div. I and II OSP Not Recov - Initiator
S8P2	1.91E-09	16943.753	SDS 8 BOP CST LOOP SLOCA SBO - Initiator
SDS16	2.68E-08	16935.252	SDS 16 BOP CST LOOP MLOCA EDG 1 and 2 - Initiator
SDS10	2.52E-08	16927.335	SDS 10 BOP CST LOOP SLOCA EDG 1 and 2 - Initiator
SDS28	1.87E-08	16887.24	SDS 28 BOP CST LOOP N2 Tank SLOCA EDG 1 and 2 - Initiator
SDS34	1.92E-08	16887.24	SDS 34 BOP CST LOOP N2 Tank MLOCA EDG 1 and 2 - Initiator
S26P2	4.78E-10	15656.341	SDS 26 BOP CST LOOP N2 Tank SLOCA SBO - Initiator
SDS9	2.73E-08	7318.496	SDS 9 BOP CST LOOP SLOCA Div. III - Initiator
SDS15	2.69E-08	7275.748	SDS 15 BOP CST LOOP MLOCA Div. III – Initiator

Table A-9 Importance Measures Evaluation for RHR A Unavailable with Protected Trains Case - Seismic			
Event	Probability	RAW	Description
SDS27	1.99E-08	7181.453	SDS 27 BOP CST LOOP N2 Tank SLOCA Div. III – Initiator
SDS33	1.82E-08	7157.026	SDS 33 BOP CST LOOP N2 Tank MLOCA Div. III - Initiator
SDS3	6.12E-07	7010.337	SDS 3 BOP CST LOOP SS LOCA Div. III - Initiator
SDS21	1.98E-07	6983.314	SDS 21 BOP CST LOOP N2 Tank SS LOCA Div. III - Initiator
SDS2	6.66E-05	6551.553	SDS 2 BOP CST LOOP SS LOCA - Initiator
SDS20	1.77E-06	6520.228	SDS 20 BOP CST LOOP N2 Tank SS LOCA - Initiator
SDS8	2.97E-07	6464.259	SDS 8 BOP CST LOOP SLOCA - Initiator
SDS14	8.28E-08	6406.118	SDS 14 BOP CST LOOP MLOCA - Initiator
SDS26	8.01E-08	6365.761	SDS 26 BOP CST LOOP N2 Tank SLOCA - Initiator
SDS32	2.79E-08	6257.519	SDS 32 BOP CST LOOP N2 Tank MLOCA - Initiator
RHRHUMNSP-LOCALL	1.00E-05	116.643	FAILURE TO ALIGN RHR TO SUPPRESSION POOL COOLING
PRAAHUS--1B-S3LL	9.01E-04	9.125	FAN PRA-FN-1B DOES NOT START ON DEMAND
SW-P-MDSWP1BS3LB	8.84E-04	9.125	FAILURE OF SSW PUMP MOTOR TO START ON DEMAND, MECHANICAL
SW-P-MDSWP1BS4LB	3.21E-04	9.125	FAILURE OF SSW PUMP MOTOR TO KEEP RUN- NING FOR 24 HR
SW-STR-SST3BE4LB	1.77E-04	9.124	SW PUMP 1B MOTOR BEARING STRAINER SW-ST-3B PLUGGED
SW-V-MODV-2BP2LB	1.89E-04	9.124	MOV SW-V-2B FAILS TO OPEN ON DEMAND
SW-V-MOV-12BP2LB	1.89E-04	9.124	MOV SW-V-12B FAILS TO OPEN ON DEMAND
SWB-XHE-RE-RHRSW	2.70E-04	9.124	HUMAN ERROR TO RESTORE SW B TRAIN FROM TEST AND MAINT
SW-SCR-SCR1BE4LB	1.12E-04	9.123	BLOCKAGE OF INTAKE SCREEN FOR SW-P-1B
PRAAHUS--1B-W4LL	9.13E-05	9.115	PRA-FC-1B CHILLER (COOLING COIL+FAN) FAILURE TO RUN
SW-PT--PS-1BW2LB	6.89E-05	9.105	FAILURE OF PRESSURE SENSOR SSW-PS-1B SIGNAL
EDCDIS1-C1-2W4LL	2.39E-05	9.078	BATTERY CHARGER C1-2 TO S1-2 DISCON NECT SWITCH FAILS
EDCDISCS1-2AW4LL	2.39E-05	9.078	FAILURE OF 200 AMP. FUSED DISCONNECT TO E-DP-S1/2A
Event	Probability	Ach W	Description
S1129	1.76E-08	122820.635	SDS 11, SDS 29 BOP CST LOOP SLOCA EDG 1 and 2 Div. III - Initiator

Table A-10
PRA Computation Results Summary

CASE	**RISK METRIC**		
Baseline	CDF/LERF		
L1_INT	6.02E-06		
L1_FIRE	9.37E-06		
L1_SEIS	4.46E-06		
L2_INT	3.27E-07		
L2_FIRE	2.69E-07		
L2_SEIS	1.85E-06		
RHR-A OOS	CDF/LERF	ΔCDF/ΔLERF	ICCDP/ICLERP
Total ICCDP			8.32E-07
L1_INT_RHRA	1.52E-05	9.18E-06	3.52E-07
L1_FIRE_RHRA	1.73E-05	7.93E-06	3.04E-07
L1_SEIS_RHRA	9.03E-06	4.57E-06	1.75E-07
Total ICLERP			1.23E-09
L2_INT_RHRA	3.30E-07	3.00E-09	1.15E-10
L2_FIRE_RHRA	2.98E-07	2.90E-08	1.11E-09
L2_SEIS_RHRA	1.85E-06	0.00E+00	Negligible
Protected Trains	CDF/LERF	ΔCDF/ΔLERF	ICCDP/ICLERP
Total ICCDP			6.25E-07
L1_INT_PROT	1.15E-05	5.48E-06	2.10E-07
L1_FIRE_PROT	1.65E-05	7.13E-06	2.73E-07
L1_SEIS_PROT	8.14E-06	3.68E-06	1.41E-07
Total ICLERP			6.90E-10
L2_INT_PROT	3.29E-07	2.00E-09	7.67E-11
L2_FIRE_PROT	2.85E-07	1.60E-08	6.14E-10
L2_SEIS_PROT	1.85E-06	Negligible	Negligible
Compensatory Measures	CDF/LERF	ΔCDF/ΔLERF	ICCDP/ICLERP
Total ICCDP			3.95E-07
L1_INT_COMP	8.88E-06	2.86E-06	1.10E-07
L1_FIRE_COMP	1.32E-05	3.83E-06	1.47E-07
L1_SEIS_COMP	8.08E-06	3.62E-06	1.39E-07
Total ICLERP			-1.30E-09[1]
L2_INT_COMP	3.16E-07	-1.10E-08	-4.22E-10[1]
L2_FIRE_COMP	2.46E-07	-2.30E-08	-8.82E-10[1]
L2_SEIS_COMP	1.85E-06	Negligible	Negligible

Table A-10 PRA
Computation Results Summary

CASE	CDF/LERF	ΔCDF/ΔLERF	ICCDP/ICLERP
LOSP Sensitivity			
Total ICCDP			6.71E-7
L1_INT_SENS	1.20E-05	5.98E-06	2.29E-7
L1_FIRE_SENS	1.72E-05	7.83E-06	3.00E-7
L1_SEIS_SENS	8.14E-06	3.68E-06	1.41E-7
Total ICLERP			1.04E-9
L2_INT_SENS	3.33E-07	6.00E-09	2.30E-10
L2_FIRE_SENS	2.90E-07	2.10E-08	8.05E-10
L2_SEIS_SENS	1.85E-06	0.00E+0	0.00E+0
F&O 2-2 Baseline	CDF/LERF	ΔCDF/ΔLERF	ICCDP/ICLERP
L1_INT_RAI_BASE	6.79E-06		
L1_FIRE_RAI_BASE	1.12E-05		
L1_SEIS_RAI_BASE	4.48E-06		
L2_INT_RAI_BASE	3.33E-07		
L2_FIRE_RAI_BASE	2.92E-07		
L2_SEIS_RAI_BASE	1.85E-06		
F&O 2-2 Sensitivity	CDF/LERF	ΔCDF/ΔLERF	ICCDP/ICLERP
Total ICCDP			7.96E-07
L1_INT_RAI	1.55E-05	8.71E-06	3.34E-07
L1_FIRE_RAI	1.92E-05	8.00E-06	3.07E-07
L1_SEIS_RAI	8.51E-06	4.03E-06	1.55E-07
Total ICLERP			9.21E-10
L2_INT_RAI	3.35E-07	2.00E-09	7.67E-11
L2_FIRE_RAI	3.14E-07	2.20E-08	8.44E-10
L2_SEIS_RAI	1.85E-06	Negligible	Negligible

[1] When crediting the additional comp measures for HPCS, LPCS and RHR-C, the LERF value result is less than the baseline result.

Table A-11
Significant Internal Events Accident Sequences: RHR A OOs with Protected system as directed by PPM 1.3.83

Sequence ID	Sequence Description	% Contribution to CDF
REC-L1-FLTSW006	Internal event flood that causes loss of plant service water. HPCS unavailable. RCIC success but fail eventually due to back pressure. Decay heat removal unavailable. Containment venting fail due to flood	13.4%
REC-L1-FLSMS007	Small main steam leak outside containment. High Pressure injection unavailable. Depressurization and low pressure succeeded. Decay heat removal unavailable. Containment venting fail due to flood.	6.1%
REC-L1-FLUMS007	Moderate main steam leak outside containment. High Pressure injection unavailable. Depressurization and low pressure succeeded. Decay heat removal unavailable. Containment venting fail due to flood.	4.0%
REC-L1-TF006	Loss of Feedwater initiating, HPCS success, failure of heat removal due to operator action fail. Containment vent fail due to operator action. Eventually containment fail which will fail HPCS. Operator success to depressurize but low pressure injection fail due to operator action	3.5%
REC-L1-FLSRF006	Involves a small reactor feedwater leak outside containment that causes an initiating event. HPCS is unavailable due to random causes of DG3, but RCIC is available for injection. Core damage occurs due to the unavailability of containment heat removal.	3.3%

Table A-12
Significant Fire Accident Sequences: RHR A OOs with Protected system as directed by PPM 1.3.83

Sequence ID	Sequence Description	% Contribution to CDF
REC-L1-T(E)N025	(T(E)N025**)** represents a fire event in the turbine building corridor, TG-12. Fire damage produces a loss of offsite power, and unavailability of RHR B, RHR C and HPCS. Depressurization succeeds. SPC is unavailable (due to RHR A in maintenance), RCIC successfully provides injection, but long term operation of RCIC fails due to unavailability of SPC.	24.8%
REC-L1-TF027	(TF027**)** represents a fire event in the NW quadrant of the reactor building 471' elevation. Fire damage produces a loss of feedwater / condensate, HPCS, RCIC, and LPCS. High pressure injection is unavailable (due to fire), but depressurization succeeds. LPCI B and C and the service water crosstie are unavailable due to unavailability of service water pump B HVAC (PRAAHUS--1B-S3LL). LPCI A is unavailable due to maintenance. Core damage occurs due to a loss of core cooling.	14.1%
REC-L1-TF023	(TF023**)** represents a fire event in the reactor building 522' elevation. Fire damage produces a loss of feedwater / condensate, RCIC, RHR A, LPCI C and LPCS. HPCS is unavailable due to maintenance (HPS---------T3LL), but depressurization and LPCI B succeed. SPC from Loop B is unavailable due to hot short of RHR-V-23 (HS-RHRV-MO-23). LPCI / RHR A are unavailable due to maintenance. Core damage occurs due to loss of decay heat removal.	12.1%
REC-L1-TT018	TT018) represents a fire event in the division 2 electrical equipment room, RC-07. Fire damage produces a turbine trip, and FW / PCS, RHR B and C and CRD are unavailable due to fire damage. RCIC is successful initially, but SPC is unavailable for continued operation for RCIC operation (RHR B is unavailable due to the fire and RHR A is out of service). CRD is unavailable once RCIC reaches operational limits. Late depressurization succeeds, but alternate injection fails due to RHR B and C unavailable due to fire, LPCS is unavailable for maintenance, condensate and SW cross-tie unavailability due to fire.	9.3%
REC-L1-SBO-R044	The above sequence (SBO-R044**)** represents a full PAU burn-up fire event in the cable chase, RC-03. Fire suppression fails, and this scenario leads directly to core damage due to fire impacts.	7.8%

Table A-13
Significant Seismic Accident Sequences: RHR A OOs with Protected system as directed by PPM 1.3.83

Sequence ID	Sequence Description	% Contribution to CDF
REC-L1-SDS2-50	SDS2-50**)** represents a seismic event that results in seismic damage state 2, which is a loss of PCS, the condensate storage tank, offsite power and a small-small LOCA. RHR B is unavailable due to the unavailability of the division 2 DG (EACENG-EDG2SS4D2). For this particular sequence, high pressure injection fails due to CST unavailable and operator failed to transfer HPCS to the suppression pool, but depressurization and low pressure injection (LPCS) are successful. Recovery of offsite power failed (NRAC24H-SEIS**).** Core damage occurs as a result of the unavailability of suppression pool cooling and containment venting.	39.1%
REC-L1-SDS42-02	(SDS42-02**)** represents a seismic event that results in seismic damage state 42, which is a failure of the reactor pressure vessel and proceeds directly to core damage.	29.2%
REC-L1-SDS41-01	(SDS41-01**)** represents a seismic event that results in widespread failure of ECCS and proceeds directly to core damage	10.1%
REC-L1-SDS41-02	(SDS41-02**)** represents a seismic event that results in widespread failure of ECCS and proceeds directly to core damage.	10.1%
REC-L1-SDS2-47	(SDS2-47**)** represents a seismic event that results in seismic damage state 2, which is a loss of Power Conversion System (PCS), the condensate storage tank, offsite power and a small-small LOCA. RHR B is unavailable due to the unavailability of SW B HVAC (PRAAHUS--1B-S3LL). For this particular sequence, high pressure injection fails because CST fail and operator fails to transfer HPCS to the suppression pool, but depressurization and low pressure injection (LPCS) succeed. Offsite power is recovered within 24 hours, but core damage occurs as a result of the unavailability of SPC and containment venting.	3.5%

Attachment B

This attachment provides the results for the sensitivity studies documented in Section 6.

Table B-1 14-DAY EXTENSION RHR PREVENTIVE MAINTENANCE Average Maintenance Model - With Protected Trains per PPM 1.3.83 and Compensatory Measures: HPCS (and support systems), RCIC, LPCS, TRS, and RHR-C		
Risk Metric	**Acceptance Guideline**	**PRA Results**
RHR A - ICCDP (Total)	< 1.0E-6	3.95E-7
RHR A - ICCDP (Internal Events)		1.10E-7
RHR A - ICCDP (Fire)		1.47E-7
RHR A - ICCDP (Seismic)		1.39E-7
RHR A - ICLERP (Total)	< 1.0E-7	-1.30E-9[1]
RHR A - ICLERP (Internal Events)		-4.22E-10[1]
RHR A - ICLERP (Fire)		-8.82E-10[1]
RHR A - ICLERP (Seismic)		Negligible

Note to Table B-1:

[1] Negative ICLERF results indicate that the LERF results are not sensitive to RHR A being unavailable and the compensatory measures have a greater influence on risk that the unavailability of RHR A.

Table B-2 14-DAY EXTENSION RHR PREVENTIVE MAINTENANCE Average Maintenance Model - With Protected Trains per PPM 1.3.83 and Loss Offsite Power Frequency Doubled		
Risk Metric	**Acceptance Guideline**	**PRA Results**
RHR A - ICCDP (Total)	< 1.0E-6	6.71E-7
RHR A - ICCDP (Internal Events)		2.29E-7
RHR A - ICCDP (Fire)		3.00E-7
RHR A - ICCDP (Seismic)		1.41E-7
RHR A - ICLERP (Total)	< 1.0E-7	1.04E-9
RHR A - ICLERP (Internal Events)		2.30E-10
RHR A - ICLERP (Fire)		8.05E-10
RHR A - ICLERP (Seismic)		Negligible

Table B-3 14-DAY EXTENSION RHR PREVENTIVE MAINTENANCE Average Maintenance Model – With Protected Trains per PPM 1.3.83 Sensitivity for Resolution of F&O 2-2		
Risk Metric	**Acceptance Guideline**	**PRA Results**
RHR A - ICCDP (Total)	< 1.0E-6	7.96E-7
RHR A - ICCDP (Internal Events)		3.34E-7
RHR A - ICCDP (Fire)		3.07E-7
RHR A - ICCDP (Seismic)		1.55E-7
RHR A - ICLERP (Total)	< 1.0E-7	9.21E-10
RHR A - ICLERP (Internal Events)		7.67E-11
RHR A - ICLERP (Fire)		8.44E-10
RHR A - ICLERP (Seismic)		Negligible

Technical Adequacy of Columbia PRA for TSTF-425

1.0 OVERVIEW

The following is the discussion of the Columbia Probabilistic Risk Assessment (PRA) - commonly identified as Probabilistic Safety Assessment (PSA) technical adequacy as provided in the Energy Northwest License Amendment Request (LAR) for Adoption of Technical Specification Task Force Traveler (TSTF)–425, REVISION 3 and as supplemented by responses to requests for additional information for said LAR (ADAMS Accession Nos. ML15093A178, ML15260A570, ML15302A492, ML160984A387, and ML16174A432). The PRA technical adequacy also meets the requirements for the proposed one-time change in the Completion Time (CT) for Technical Specification (TS) conditions 3.5.1.A, 3.6.1.5.A, and 3.6.2.3.A. There are no open F&Os and the PRA model satisfies capability category II of the internal events standard, which is adequate to support a CT change evaluation of TS. Since this information is applicable to the current submittal, it is therefore provided below.

2.0 DOCUMENTATION OF PROBABILISTIC RISK ASSESSMENT TECHNICAL ADEQUACY FOR LICENSE AMENDMENT REQUEST FOR ADOPTION OF TECHNICAL SPECIFICATION TASK FORCE TRAVELER (TSTF)-425, REVISION 3

Introduction

The NRC has previously reviewed the Columbia PSA for technical adequacy as part of the approval for License Renewal. By letter dated January 19, 2010, Energy Northwest submitted an application to the NRC to issue a renewed operating license for Columbia for an additional 20 years. As part of this submittal, Columbia submitted an assessment of Severe Accident Mitigation Alternatives (SAMAs), based on what was then the most recent Columbia PSA (Revision 6.2). During the NRC review of the SAMAs, the NRC issued requests for additional information (RAIs), and Columbia responded to the requests. Also during this review process, the PSA was updated from Revision 6.2 to Revision 7.1. As part of the responses to the RAIs, Columbia provided updated PSA information to the NRC based on PSA Revision 7.1. The PSA Revision 7.1 model incorporated the following:

- Resolution of Facts and Observations (F&Os) from the 2004 peer review;
- Resolution of areas of model incompleteness identified by Columbia internal technical reviews;
- Upgrades to meet NRC RG 1.200, Revision 2 and the associated ASME Standard RA-S-2008 for Level 1, LERF, and flooding modeling; and
- Other plant and procedure changes.

The above changes were first incorporated in the PSA Revision 7.0 model, for which a peer review was performed in 2009 on Level 1 and 2 internal events, which included internal flooding, and a report was issued in January 2010. The F&Os from this peer review that could significantly impact the model quantification were incorporated into the

Revision 7.1 model, and a review of the remaining F&Os associated with supporting requirements (SupRs) that were graded as CC-1 or not met identified none that would significantly impact the results of the SAMA analysis.

The NRC reviewed the information provided by Columbia concerning the internal events PSA model, seismic PSA model, Fire PSA model, and "other" external events PSA model. This review is documented in NUREG-1437, Supplement 47, Volume 1, "Generic Environmental Impact Statement for the License Renewal of Nuclear Plants Regarding Columbia Generating Station" (ADAMS Accession No. ML12096A334). The information regarding PSA revisions 6.2, 7.0, and 7.1 are included in NUREG-1437, Supplement 47, Volume 1, Section 5.3.2. In addition, the following statements of acceptability of the Columbia PSA model were made by the NRC in Section 5.3.2.

Internal Events

"The Columbia internal events model has been peer-reviewed, the peer findings were all resolved and their impact assessed in a sensitivity analysis using the updated PSA model, and Energy Northwest has satisfactorily addressed NRC staff questions regarding the PSA. Based on this information, the NRC staff concludes that the internal events Level 1 PSA model is of sufficient quality to support the SAMA evaluation."

Seismic

"The Columbia internal events modeling is an input to the seismic PSA model, the seismic PSA has been updated to a more recent external events PSA standard, the SAMA evaluation included a sensitivity analysis of the seismic CDF, and Energy Northwest has satisfactorily addressed NRC staff RAIs regarding the seismic PSA. Based on this information, the NRC staff concludes that the seismic PSA model in combination with the sensitivity analysis of the seismic CDF provides an acceptable basis for identifying and evaluating the benefits of SAMA."

Fire

"The Columbia internal events modeling is an input to the Fire PSA model, has been updated to incorporate industry fire data and NRC guidance, the fire PSA has been peer reviewed and the peer review findings were all addressed, and Energy Northwest has satisfactorily addressed NRC staff RAIs regarding the fire PSA. Based on this information, the NRC staff concludes that the fire PSA model provides an acceptable basis for identifying and evaluating the benefits of SAMAs."

"Other" External Events

"The NRC staff reviewed the Level 2 methodology and found that Energy Northwest adequately addressed NRC staff RAIs, the Level 2 PSA model was reviewed in more detail as part of the 1997 and 2004 peer reviews, and the findings from these peer

reviews have been resolved and their impact assessed in a sensitivity analysis using the updated PSA model. Based on this information, the NRC staff concludes that the level 2 PSA provides an acceptable basis for evaluating the benefits associated with various SAMAs."

Columbia has recently updated the PSA internal events model to Revision 7.2. Based on the previous NRC review and acceptability of the Columbia PSA Revision 7.1 model, the remainder of this section includes a general description of the PSA model and those changes from PSA Revision 7.1 to PSA Revision 7.2, and the basis for the technical adequacy of these changes.

General Description of the Columbia PSA Model

The Columbia PSA is performed in accordance with the Capability Category II requirements of ASME / ANS Standard, as clarified by Regulatory Guide 1.200. The PSA is a full scope PSA (the shutdown PSA and the external events including earthquakes, fires and winds are reported elsewhere and not discussed in this section) which uses a Level 1 analysis to model core damage frequencies and a Level 2 analysis to model containment performance characteristics during severe accidents. In the Level 1 analysis, a mission time of 24 hours was generally assumed for all equipment which plays a role in providing core protection functions after a transient and in the Level 2 analysis, a period of up to 40 hours was used to simulate and monitor predictions of containment performance following the occurrence of an accident sequence initiating event. The phenomenological uncertainties embodied in the Level 2 analysis were held to a minimum by using referenced assumptions wherever possible. The final results of the containment performance assessment are presented as "best point estimates."

The Columbia PSA uses the "large fault tree/small event tree" approach to identify and quantify individual accident sequences. With this approach, functionally descriptive event trees are used to model the possible accident sequences which result in core damage, and initiator-specific containment event trees are used to represent possible containment behavior following a core melt accident. Detailed system fault trees are merged to quantify individual event tree sequences in both the Level 1 and Level 2 analyses. The CAFTA and PRAQuant computer codes are used to perform the quantification and both the RETRAN and MAAP computer codes were used to characterize plant, containment, and fission product behavior.

The effects of common cause failures are included as basic events in the Columbia system fault trees and are quantified by the alpha factor method. This method provides a numerical estimate for the common cause failure probability which should be assigned to account for the likelihood that the plant would experience simultaneous failures of two or more identical components in a single train or multiple trains in a single system.

To ensure that the role of the operating staff is fully credited by the analysis, pre-accident and majority of the post-accident human actions were explicitly modeled in the detailed system fault trees, and recovery actions were linked to the functional headings in the event trees. A plant specific human reliability analysis (HRA) is used to predict pre-accident and post-accident human error probabilities for each of the modeled human actions. The potential dependencies between multiple operator actions performed in accident sequences modeled by the PSA have been evaluated and modeled. The HRA dependency evaluation examines potential influences on the failure likelihood for a task from the failure or success of the immediately preceding task.

The Level 1 and Level 2 PSA analyses were performed following the methodology and major tasks as described in NUREG/CR-2300, "PSA Procedures Guide" for Level 1 and Level 2 PSA. The major Level 1 tasks include information gathering of plant specific data, P&IDs, test, maintenance, operating procedures, and physical room, containment, and building information. The system analysis methodology utilizes the small event tree / large fault tree approach. The event trees combine the initiating event with system functional successes or failures to delineate the accident sequences. The fault trees are developed to the component, relay, and sensor level of detail. The system modeling includes human reliability and common cause failure events, utilizing plant specific failure data to the extent possible. The human reliability analysis (HRA) primarily utilizes the Cause-Based Decision Tree (CBDTM) methodology to estimate cognitive error probabilities for actions not constrained by timing, and the Human Cognitive Reliability/Operator Reliability Experiments (HCR/ORE) methodology for time-constrained actions, and Technique for Human Error rate Predication (THERP) to estimate execution error probabilities. Common cause modeling is performed with the alpha factor method utilizing data from the NRC Parameter Estimation database. The PSA is quantified using the CAFTA and PRAQuant computer codes.

The Level 2 tasks include grouping the Level 1 sequences into several bins or plant damage states, characterizing containment failure modes and locations, developing and quantifying logic trees and containment event trees, determining the magnitude of the radionuclide release, and performing sensitivity studies. The MAAP computer program was used to provide an integrated approach for modeling of plant thermal hydraulic response and fission product transport during severe accidents. In addition, research results in the open literature, Industry Degraded Core Rulemaking Group (IDCOR) task reports, the Shoreham and Limerick PRAs, NUREGs, and engineering judgment were used in understanding accident progression and quantifying event trees.

The Level 1 analysis is coupled with the Level 2 analysis through the binning of the multitude of the Level 1 sequences into several groups of plant damage states with similar Level 2 characteristics. Results of a structural analysis performed for Columbia are used to assess containment strength, failure size and location. MAAP calculations, analytical models, and widely accepted research results are employed in the accident progression analysis. Based on this information, containment event trees and logic trees were developed to provide a description of the containment damage states. The

containment event trees were quantified using the CAFTA and PRAQuant codes. The end states of the containment event trees were then grouped into a limited but complete set of unique release categories. Representative MAAP calculations include Large Early Release Frequency (LERF) were performed to provide an estimate of the fission product release to the environment. Finally, a sensitivity analysis was performed to investigate important parameters that could have large impact on the likelihood or time of containment failure and the magnitude of the source term. The results were used to identify the areas for which potential improvements of the plant might be considered.

Because of the consistency throughout the process and the linking and merging capabilities of the CAFTA and PRAQuant codes, dependencies, common cause effects, and system interaction are accounted for by the PSA.

CAFTA and PRAQuant, PC-based programs developed by Electric Power Research Institute (EPRI), are used to quantify the Level 1 and Level 2 accident sequences. CAFTA is used to convert the system-level fault trees into a single master fault tree that solves all event tree sequences from a single top gate (specifically, one top gate for the Level 1 solution, and one top gate for the Level 2 solution).

Verification of the CAFTA results was performed by installing the software on in-house PCs, its fault tree and sequence results were checked against a CAFTA sample quantification. They were found to be identical. CAFTA has been verified and validated in accordance with Energy Northwest's computer code Verification and Validation procedure.

Changes from PSA Revision 7.1 to PSA Revision 7.2

- PRA Update - Update to the current as-built, as-operated plant, this included incorporation of outstanding Engineering Changes, current procedure revisions, and new reliability information. Also included in the update of the PRA model to Revision 7.2, was Revision 3 of Emergency Operating Procedures/Severe Accident Guidelines that were approved and implemented for addressing Fukushima lessons learned.

- This update also involved conversion from WINUPRA computer code to CAFTA computer code. This involved reconciliation of all cutsets to Revision 7.1 event tree model above the Revision 7.1 truncation level, 5E-12 (55,000+ cutsets). By reconciling the cutsets this provides a high level of confidence in the conversion necessary for PRA applications and ensured continued support of Capability Category II.

- Resolved 109 of 131 self-assessment including peer review comments accumulated over the 5 years preceding submittal of the LAR for Adoption of Technical Specification Task Force Traveler (TSTF)–425, REVISION 3 in March 2015. There were two open findings from the 2009 peer review remain open with

this minor update and are discussed below. This effort addresses comments / suggestions judged to be medium-or-higher priority, with the highest priority applied to the items addressing modeling enhancements. Additional suggestions and self-assessment findings that are documentation-related were postponed based upon risk impact. Resolving these open comments / suggestions improved model realism to further support PSA Capability Category II.

- Discovered and corrected support system dependency modeling in motor control center heating and ventilation including containment venting.

- Credited and modeled a new operator action – Manual containment vent (MAN-VENTFAIL)

- Updated the HRA dependency analysis to utilize the HRA Calculator dependency module. This update was needed to assess dependence for the new HFE added to the model for manual containment vent, as well as include risk-significant dependent combinations missing in Rev. 7.1 (300 cutsets were reviewed for dependent HFE combinations in Rev. 7.1 versus 1.3 million cutsets in Rev. 7.2).

- Updated the HRA Calculator development to utilize the CBDTM or human cognitive reliability / operator reliability experiment methods to estimate cognitive human error probabilities, rather than the combination sum (CBDTM + ASEP) approach, which follows industry and NRC guidance consensus.

- These changes to the modeling were specifically focused on aligning the PSA model with the as built and as operated facility. It improved the realism of the PRA model and with the incorporation of the outstanding changes and addressing open suggestions and findings resolved issues that had been identified with the model

These changes to the modeling were specifically focused on aligning the PSA model with the as built and as operated facility. It improved the realism of the PRA model and with the incorporation of the outstanding changes and addressing open suggestions and findings resolved issues that had been identified with the model.

The two open findings from the 2009 peer review which were not addressed in the PSA model update from Revision 7.1 to Revision 7.2 are discussed as follows and the impacts identified:

F&O 2-2, Finding, SupR DA-C6:

Observation:

The number of plant-specific demands on standby components was mainly documented for the maintenance rule and MSPI components. Tier 3 PSA

documents for PSA-2-DA-0002 show the details. Estimates based on the surveillance tests and maintenance acts as described in this SR should be performed even though the major components have been included in the MSPI data.

Estimates based on the surveillance tests and maintenance acts as described in DA-C6 and DA-C7 should be performed for significant components whose data are not tracked in the MSPI data.

(This F&O originated from Supporting Requirement DA-C6)

Recommendation:

Update the estimates for significant events based on surveillance test and maintenance records.

Impact:

As a sensitivity, the base data for these failure modes were replaced by the following generic data (NUREG/CR-6928) and the model was quantified. This change in data produces a bounding delta CDF increase of 2%, which is a risk-significant difference, but relatively unlikely to change the conclusions of risk-informed decisions. However, until this F&O finding is resolved, PRA applications will perform this sensitivity analysis to examine whether the application is impacted.

- C---W2 = 9.2E-5/hr (compressor loss of function)
- C---W4 = 9.2E-5/hr (compressor loss of function)
- FN--R3 = 4.6E-3 (fan fails to start)
- FN--W4 = 1.1E-5/hr (fan fails to run)
- AHUSS3 = 1.2E-3 (standby air handling unit fails to start)
- AHURW4 = 1.4E-5/hr (running air handling unit fails to run)

F&O 2-14, Finding, SupR SY-A4:

Observation:

Interviews with plant system engineers or operators have not been documented and cannot be verified by the peer review team. Original interviews were performed for the original IPE.

System and operations change over time, and the system engineers and operators should be consulted with regard to the system models.

(This F&O originated from SR SY-A4)

Recommendation:

Perform and document the interviews with the system engineers and/or operators to confirm that the systems analysis correctly reflects the as-built, as-operated plant. It may be reasonable to develop a process that, following an initial interviews, the confirmation the model matches the as-built, as-operated plant is confirmed through review or discussion with the system engineers - typically through a periodic review of the notebook performed in accordance with plant procedures.

Impact:

SY-A4 only meets Capability Category I due to this open finding. Until this F&O finding is resolved, PRA applications will perform this sensitivity analysis to examine whether the application is impacted.

Conclusion

Based on the above information, Energy-Northwest believes that the quality of the current PSA is sufficient to support Adoption of TSTF-425 and the remaining open F&Os do not unduly impact the result.

3.0 SUPPLEMENTAL INFORMATION AS A RESULT OF REQUESTS FOR ADDITIONAL INFORMATION ON THE LAR FOR ADOPTION OF TECHNICAL SPECIFICATION TASK FORCE TRAVELER (TSTF)–425, REVISION 3

3.1 Response dated September 17, 2015 (ML15260A570) to August 12, 2015 Request for Additional Information (ML15224B646)

NRC Request

1. The LAR mentions a number of peer reviews and a self-assessment:

   - A peer review had been performed in 2004.
   - A peer review had been performed in 2009 and a report issued in January 2010. Findings and Observations (F&Os) included those graded as capability category I (CCI) or not met.
   - A self-assessment had been performed.
   - A Fire Probabilistic Risk Assessment (PRA) peer review had been performed.

   a. Please clarify which peer reviews (internal events PRA, Fire PRA, or other PRA) were full scope or focused scope, discuss the peer review guidance, standards, and regulatory guidance followed, and confirm the reviews were conducted consistent with applicable guidance and standards. Please clarify whether the

internal events PRA was reviewed to the Addenda to American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-S-2008 (i.e., ASME-ANS RA-Sa-2009). If reviews were not conducted consistent with applicable guidance and standards, please describe your plans to address any shortcomings in the review. With regard to the self-assessment, please describe when this was performed and the scope of the self-assessment, and whether it included a gap assessment between Regulatory Guide (RG) 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of a Probabilistic Risk Assessment Results for Risk-Informed Activities," (ADAMS Accession No. ML070240001) and RG 1.200, Revision 2 (ADAMS Accession No. ML090410014), for the internal events PRA. Please also provide additional information on the Fire PRA peer review describing when it was performed and what the peer review entailed.

b. Please provide the internal events PRA (including flooding) F&Os graded as CCI or not met and describe your disposition of these F&Os from the 2009 peer review and self-assessment.

c. If PRA models other than the internal events PRA model are used for detailed quantitative analysis versus for qualitative or bounding analyses, then please address the technical adequacy guidance of RG 1.200, Revision 2. If the LAR is requesting to use these PRA models as such, provide the F&Os graded as CCI or not met and describe your disposition of these F&Os from the peer reviews.

Energy Northwest Response:

1.a:

The following discussion clarifies which Columbia Generating Station (CGS) peer reviews (internal events PRA, fire PRA, and seismic PRA) were full scope or focused scope and discusses the peer review guidance, standards, and regulatory guidance followed. All reviews were conducted consistent with applicable guidance and standards.

Internal Events PRA

In 2004 the CGS Revision 5.0 internal events PRA received a full scope peer review against the Capability Category II (CC-II) requirements of the ASME/ANS PRA Standard, ASME RA-Sa-2003, as clarified by Regulatory Guide (RG) 1.200 (DRAFT), using the industry peer review process guidelines described in Nuclear Energy Institute (NEI) NEI-00-02, Revision A-3, "Probabilistic Risk Assessment Peer Review Process Guidance."

In 2009, the CGS Revision 7.0 internal events PRA received a full scope peer review from the Boiling Water Reactor Owners' Group (BWROG) against the ASME/ANS PRA

Standard ASME/ANS RA-Sa-2009, as clarified by Regulatory Guide (RG) 1.200, Revision 2, using the industry peer review process guidelines in NEI-05-04, Revision 2, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard."

<u>CGS Fire PRA</u>

The fire PRA has not been upgraded to meet CC-II for the supporting requirements (SRs) of the combined ASME/ANS Standard, ASME/ANS RA-Sa-2009. There are no plans to perform a peer review of the current version of the fire PRA.

In 2004, the CGS Revision 1 fire PRA (FPRA) received a full scope peer review as part of the internal events peer review. At the time, the fire PRA peer review process had limited industry and regulatory guidance available. The following is an excerpt from the 2004 Peer Review report on the guidance used:

> *"…a technical checklist for conducting a FPRA Peer Review was developed based on the available references for use in the PRA Peer Review of the CGS Fire PRA.*
>
> *In general, the objectives of the checklists are to provide a structured process to confirm:*
> 1. *Use of acceptable methodology in comparison with current state of the art and industry practices.*
> 2. *Appropriateness of methodology and analysis scope given intended application.*
> 3. *Analysis is free of obvious errors or misapplications*
> 4. *Acceptable level of detail to support the specific application under review – S/G AOT Extension*
>
> *The Fire PRA Peer Review checklists were developed based in ERIN's standard practices for FPRAs, the guidance provided in NEI 00-02, The [Electrical Power Research Institute] EPRI Fire PRA Implementation Guide, the industry responses to the NRC Generic RAIs, and Regulatory Guide (RG) 1.200. ERIN's standard project approach for conducting a FPRA formed the basic framework and structure for the checklist. The NEI 00-02 document provided guidance for the depth of review while the RG 1.200 and EPRI documents provided specific issues for review. Based on this Information, the checklists are structured using six topical areas:*
>
> 1. *Fire Areas and Fire Compartments (FC)*
> 2. *Cable and Equipment Location Data (CE)*
> 3. *Developments of Fire Ignition Frequencies (FI)*
> 4. *FPRA Model Development – Plant Response (FM)*

5. *Fire Scenario Development (FS)*
6. *FPRA Model Quantification (MQ)"*

CGS Seismic PRA

The CGS seismic PRA (SPRA) has not been peer reviewed. The SPRA has not been upgraded to meet CC-II for the supporting requirements of the combined ASME/ANS Standard, ASME/ANS RA-Sa-2009. There are no plans to perform a peer review of the current version of the seismic PRA.

PRA Self-Assessment Process
No gap assessment between RG 1.200 Revision 1 and Revision 2 has been performed on the CGS PRA. CGS Revision 7.0 internal events PRA received a full scope peer review from the BWR Owners' Group against the ASME/ANS PRA Standard, ASME/ANS RA-Sa-2009, as clarified by Regulatory Guide 1.200, Revision 2, using the industry peer review process guidelines in NEI 05-04, Revision 2.

A Self-Assessment as described in the submittal refers to the process in which PRA modeling self-identified facts and observations (F&O) are entered into the CGS F&O database for inclusion in the next modeling update.

1.b:

The 2009 internal events peer review and self-assessment findings assigned to SRs that were graded as CC-I or not met are presented in Table 1. Dispositions of findings are provided in Table 2.

Table 1
F&Os for SRs Graded as CC-I or Not Met by the 2009 Peer Review and Self-Assessment

SR	2009 Peer Review Assessment	F&Os
IE-C14	Not Met	1-10, 1-14
SY-A4	CC-I	2-14
SY-A14	Not Met	2-16, 2-17
SY-A24	Not Met	3-11
SY-B1	Not Met	6-10, 6-8, 6-9
SY-C2	Not Met	2-14, 2-16, 2-17
HR-D2	CC-I	1-23
HR-D3	CC-I	1-23
HR-D4	Not Met	1-23
HR-G3	CC-I	1-3
DA-D1	CC-I	3-1
LE-C7	Not Met	1-3, 1-33, 1-42, 1-43

Table 1 F&Os for SRs Graded as CC-I or Not Met by the 2009 Peer Review and Self-Assessment		
SR	**2009 Peer Review Assessment**	**F&Os**
LE-D4	Not Met	1-10, 1-14, 1-43
LE-G6	Not Met	2-18
IFPP-B3	Not Met	2-28
IFQU-A6	Not Met	1-3, SA-IF-E5a-1
MU-C1	Not Met	4-7

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 1-10 SupRs: IE-C14 LE-D4	High Pressure Core Spray (HPCS) notebook does not include the Surveillance Test for HPCS-V-4. Additionally, the screening for HPCS in the ISLOCA Section of the Initiating Event Notebook (appendix E) does not account for the possibility of ISLOCA during testing of this valve. A possible scenario is, prior to V-4 test, the injection Check Valve V-5 has failed open, and once the test has completed, V-4 does not close and V-24 fails (may be CC). The result is ISLOCA to low pressure piping. OSP-HPCS/IST-Q701 lists V-4 as being tested quarterly. V-23 is open during the test, given reactor is at pressure.	Consider the quarterly test of V-4 in the ISLOCA analysis. Update the HPCS notebook to include IST for HPCS valves, including V-4.	Peer Review	Resolved: **YES** To resolve this finding, an ISLOCA assessment was performed for the HPCS system, including consideration of HPCS-V-23, HPCS-V24, and the inservice testing of HPCS-V-4 and as discussed in the finding. Documentation of this assessment was added to Appendix E, ISLOCA Initiating Event Frequency, of the initiating events notebook.
F&O 1-14 SupRs: IE-C14 LE-D4	ISLOCA analysis applies a conditional probability of valve closure from NSAC-154, but applies this to NUREG/CR-6928 data. See IE notebook, Table E-1. Application of The NSAC data does not appear appropriate when applied to NUREG/CR-6928. The new valve failure data, such as check valve internal rupture, is not applicable to the older factors for valve re-closes. For example, CV closes following pipe failure (0.01) is based on older data, while the NUREG/CR-6928 data for check valve rupture would typically have screened failures where the check valve stuck open (fails to close), but later reclosed. ISLOCA is risk significant, and the factors affect multiple sequences.	ISLOCA analysis should be revised to remove NSAC-154 factors, and apply factors based on the latest valve failures in NUREG/CR-6928.	Peer Review	Resolved: **YES** The ISLOCA event tree was modified to address this peer review finding. The ISLOCA analysis was revised to replace older valve failure data published by NSAC-154 with the latest valve failure data published in NUREG/CR-6928, wherever newer data was available. A review of NUREG/CR-5124 revealed that the conditional probability of check valve closure (event tree node CV) is applicable only to scenarios in which a testable check valve is held open due to reverse air flow to the controller. The 0.01 credit had minimal impact on the ISLOCA sequence cutsets, as the significant check valve failures are leakage and rupture. Therefore, event tree node CV was removed from the ISLOCA event tree. Event tree node SML (small leak through the high/low pressure interface) is moved to an earlier position in the event tree, as leaks through the high/low pressure boundary are judged to be isolable (the 900-pound MOVs that are present at the boundaries are judged to be capable of closing against a leak). The model retains the assumption that MOVs would not likely close if a piping rupture and interface rupture were to occur, per NUREG/CR-5124 guidelines. Also, the event tree node ISOVAV (early isolation of the ISLOCA) probability has been changed to a 0.1 based on documentation provided with the CGS SPAR model. From the CGS SPAR model documentation:

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 1-14 SupRs: IE-C14 LE-D4				Without doing detailed pressure capacity calculations and detailed modeling of the expected internal pressures and temperatures expected in the connected systems, it is impossible to predict the location of potential ruptures. Even with detailed calculations and modeling, precise rupture locations are impossible to identify. Nevertheless, some general observations can be made based on the GI-105 research. For most situations the RHR heat exchanger, and pump suction pipe are the components with the lowest pressure capacities. Generally, these components are positioned within the systems such that one or more valves are available to isolate a rupture, should an ISLOCA occur at these locations. However, it is possible that if the pressure isolation interface were to fail, that either the available valves would not successfully isolate the rupture, or the rupture could occur in a location that cannot be isolated. To account for these possibilities, a generic 10% probability is assumed that if a rupture were to occur, it cannot be isolated. This 10% probability for the rupture being non-isolable can be considered to be a reasonable estimate for a number of reasons. First, virtually every rupture location examined as part of the GI-105 research program was found to be potentially isolable. The pipe and other components (e.g., pump suction pipe and RHR heat exchangers) that are most susceptible to over-pressure induced rupture, are located “deeper” within the connected system such that a number of valves are typically available for isolating the rupture. Further, the typical failure mode postulated in the ISLOCA analysis for motor operated valves is spurious operation. The few actual instances of this observed in the operating experience were all recoverable from the control room. However, one factor that affects the ability to isolate the rupture is local accessibility. If a rupture were to occur, the resulting local environment would likely preclude access to the immediate vicinity. Therefore, if local access was necessary, and if the potential isolation valves were located close to the rupture then isolation would be unlikely. Again, the research performed to support resolution of GI-105 included an assessment of induced flooding and the resultant environment. That work concluded these effects

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 1-14 SupRs: IE-C14 LE-D4				would not significantly affect the ISLOCA risk. Therefore, the non-isolable ruptures are assumed to compose 10% of the potential ISLOCA ruptures. The accident sequence notebook was updated to reflect all of these changes to the ISLOCA model.
F&O 1-23 SupRs: HR-D2 HR-D3 HR-D4	The quantification of pre-initiating events using 3 surrogate events to represent all pre-initiating events does not provide an accurate assessment of each HEP, taking into account plant specific or component specific attributes. Appendix A.4 for example, provides a "generic" assessment of miscalibration with dependencies, without actually including the specific attributes for the procedure affecting events it is applied. For pre-initiator failure to restore events, the use of the surrogate event does not account for post-maintenance testing, operations walkdowns, and when the system may be operated again. For pre-initiator miscalibration, the surrogate event does not represent the plant specific calibration procedures. In Appendix A.4, the following is provided in the generic analysis "The results of these evaluations indicate that, depending on the methods used to verify the adequacy of the calibration and the assumptions used in the quantitative evaluation, there can be substantial variation in the common cause miscalibration error probability." This statement is true, which is why plant-specific attributes are needed for this analysis. Numerous Pre-Initiating Events in all three categories are risk-significant, based on table E-2 of the QU notebook. Comparison with other PRAs shows that variation between component/action specific HEPs is expected, and the range of differences from one component to the next can be several orders of magnitude. For mis-calibrations, for example, depending on whether the mis-calibration can be quickly recognized or whether there is a second check on the calibration can greatly affect the results. Variation in dependent failures and failure to restore is equally large within a PRA. CGS provides discussion on this F&O suggesting the important pre-initiating events were represented by the "representative" actions, and the procedures for each action	Re-Analyze pre-initiating events for all significant HEPs, taking into account the component specific testing, maintenance and operations attributes affecting the HEP. Use of surrogate or generic analysis should be limited to non-significant events, with some justification that the surrogate is representative or bounding for the HEPs it is applied.	Peer Review	Resolved: **YES** Procedure-specific pre-initiator HEP calculations were developed for the top three pre-initiator human failure events when sorted by RAW and the top three pre-initiator human failure events when sorted by Fussell-Vesely based on the CGS Rev. 7.0 Level 1 PRA model results. This resulted in a total of five pre-initiator human failure events for which to perform procedure-specific HEP calculations. (Note - Five are evaluated, rather than six, because the top pre-initiator human failure event when sorted by RAW was also the top pre-initiator action when sorted by F-V). Section 3.0a and Appendix A.0 were added to the HRA Notebook to document the updated procedure specific pre-initiator HEP calculations. The five revised pre-initiator HEPs were subsequently incorporated into the final HRA Calculator file "CGS 2008 HRA.HRA"

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 1-23 SupRs: HR-D2 HR-D3 HR-D4	were similar. The following procedures were reviewed for this follow-up: Representative Procedures: SOP-CN-FILL (for CN-HUMNTK--1X3XX), ISP-LPCS/RHR-X301 (for LPSHUMNFIS-4XLL) Significant HEP Procedures: Calibration: ISP-RCIC-Q901 (HEP: RCIHUMNPS13AX3LL), 10.27.86 (HEP: RCIHUMNPS--6X3LL) Restoration: OSP-SW/IST-Q702 (HEP: SWB-XHE-RE-RHRSW), OSP-SW/ISP-Q701 (HEP: SWA-XHE-RE-RHRSW), SOP-COLDWEATHER-OPS (HEP: SW-HUMNV218-X3LL), OSP-HPCS/IST-Q701 (HEP: HPC-XHE-RE-MAINT) Based on a review and comparison of the representative procedures versus the procedures for significant HEPs, it is clear the procedures, associated critical steps, and verification steps are significantly different.			
F&O 1-3 SupRs: HR-G3 LE-C7 IFQU-A6	Analysis for HEPs apply low stress to post-accident situations for almost all HEPs, even though the criteria in NUREG-/CR-1278 recommends High Stress especially when the time window is short. See HPSHUMN-SP-H3LL, OP-SW-PMP, SLC-XHE-FO-LLVCT and others. Supplemental guidance was provided by CGS personnel as a result of this issue. The CGS argument included an argument that there was not extensive use of simulator training prior to the development of NUREG/CR-1278, and that this training affects the application of stress to simple actions where training occurs. A second set of justification was provided, with discussion that basically justified moderate or low stress would be appropriate, given enough training for the operators. Review of this new guidance does not provide sufficient justification for the revised application of NUREG/Cr-1278 Guidance. Two points are important to this finding: a) stress will affect actions occurring during an accident in comparison to non-accident actions making the actions less reliability, and b) Stress is based on an "overall sense of being pressured and/or threatened in some way	Apply high stress factors per Table 17-1 of NUREG/CR-1278 to HEPs where time pressure is present during an accident situation. Benchmarking against other PRAs performed by alternate vendors to determine how the Stress Factors from 1278 were applied can be useful for this issue.	Peer Review	Resolved: **YES** After reviewing the peer review F&Os, no changes to the calculated post-initiator HEPs were judged necessary to address the post-initiator stress levels. Section 4.4 of the HRA documentation has been enhanced to further justify the position that the post-initiator stress levels are treated appropriately in the CGS PRA. The use of high stress for the manipulation errors in the post-initiator HEPs is performed on a case-by-case basis. The blanket requirement that all manipulations in the plant for the first hour of an event are high stress does not agree with the intent of NUREG/CR-1278 nor common practice within the industry as confirmed with the developers of the HRA Calculator and the Chairman of the EPRI committee on HRA. Results from the CGS inquiry of the industry and alternative vendors, the time consideration for the stress factor for Internal Events PRA is NOT typically treated as interpreted by the peer review team. HRA leaders in the industry do not agree with the position on time pressure

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 1-3 SupRs: HR-G3 LE-C7 IFQU-A6	with respect to what they are trying to accomplish." Training cannot fully remove the treat or pressure during an actual event. Stress factors assumed to be nominal are also in the Level II model and flooding model for short duration HEPs. NUREG/CR-1278 recommends applying optimum stress to activities such as maintenance and calibration activities, reading an annunciator light, or scheduled readings in the control room. On the other hand, Page 17-7 states that 'In general, situations that impose time pressure on the performers are classified as heavy task load situations.' In almost all of the Post-Initiator HEPs where optimal stress is assumed, time is a factor with Core Damage occurring between 30 minutes and an hours. This time stress is typically modeled as high stress in HRA using NUREG/CR-1278. Numerous HEPs where this is applied are risk-significant. Discussions with CGS staff indicated that the low stress was applied to actions that are relatively simple. However, since this simple actions are already low if failure rate, the stress factor is still applicable in order to differentiate between a simple action performed as a routine action, and a simple action performed in order to avoid core damage. Nominal stress is also applied for level II actions, such as failure to provide injection after the control rods fail (10 minute window) just prior to core damage. The general rules for this as applied by CGS do not appear consistent with NUREG/CR-1278 or other PRAs reviewed for this issue.			(less than 1 hr = high stress) associated with Finding 1-3. The Columbia upgrade project specifically used the most current methodology from EPRI (HRA Calculator) available. The general consensus is there have been improvements in knowledge and methodology since NUREG/CR-1278 was issued over 30 years ago that support execution error evaluations to be assessed considering current training regimes and the simple nature of such actions.

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 1-33 SupRs: LE-C7	HEPs are calculated using the HEP calculator, and are realistically treated in most cases consistent to the Applicable procedures. However, documentation on several actions was not found in the HRA notebook: 1) L2-HUMN-MUPHNOWS 2) L2-HUMN-RCVR-SYS Note this is just a sampling of the notebook, so others may also be missing. L2-HUMN-MUPHNOWS is mentioned in Section C.16 of the LERF notebook, but not in the HRA notebook. Based on discussion with CGS, this HEP is set to 1.0 based on engineering judgment. L2-HUMN-RCVR-SYS is listed as using the HRA calculator, but is set to 0.9 based on engineering judgment. L2-HUMN-RCVR-SYS is risk-significant. Based on the Internal Events and Flooding HRA review, other HEPs are likely missing in the documentation (HRA notebook) that are credited in the analysis).	Add all missing HEPs from the Level II analysis into the HRA section, including events set to 1.0. Additionally, provide basis for HEP L2-HUMN-RCVR-SYS equal to 0.9 (since it is risk-significant), including information on timing, cues, procedures or other aspects causing little credit for the HEP.	Peer Review	Resolved: **YES** This finding was resolved by adding Level II human failure events to the HRA Calculator and the summary Table 5.1-2 of the HRA Notebook. Also, as documented in Table C.4.7-2 of the CGS Level 2 notebook, the basis for the screening HEP for L2-HUMN-RCVR-SYS is engineering judgement.
F&O 1-42 SupRs: LE-C7	HEP Dependency Analysis included in Appendix D of the HRA notebook does not included in the Level II Analysis. Level II modeling includes new HEPs not in the Level I HRA. LERF may be underestimated if dependent HEPs are not included along with the independent HEPs.	Complete a dependency analysis for Level II similar to the Level I analysis in HRA appendix D.	Peer Review	Resolved: **YES** To resolve this peer review finding, a Level II HEP dependency analysis was performed and documented in Section 5 and Appendix D of the HRA Notebook.
F&O 1-43 SupRs: LE-C7 LE-D4	HRA events RHRHUMN-V--803XX and 903XX are included in the ISLOCA analysis but do not include dependency considerations. See the top cutset in Appendix D, dependency analysis. Based on discussion with CGS, the cutset with 2 operator failures is not valid, since the valves are interlocked. Since these are ISLOCA cutsets, and significant for both CDF and LERF, dependency will be significant.	Either: a) Add a dependency analysis for these events, or b) Provide justification for deletion of the cutset, and add the combination to the mutually exclusive event file.	Peer Review	Resolved: **YES** This peer review finding has been resolved. The RHR-V-8 and RHR-V-9 have interlocks that prevent valve opening at normal RPV pressures above about 125 psig. Therefore human failure events RHRHUMN-V--8O3XX and RHRHUMN-V--9O3XX have been removed from the ISLOCA initiating event fault tree (that is, even in operators mistakenly select valve OPEN from the control board for either or both valves, the valves will not open).

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 2-14 SupRs: SY-A4 SY-C2	Interviews with plant system engineers or operators have not been documented and cannot be verified by the peer review team. Original interviews were performed for the original IPE. System and operations change over time, and the system engineers and operators should be consulted with regard to the system models.	Perform and document the interviews with the system engineers and/or operators to confirm that the systems analysis correctly reflects the as-built, as-operated plant. It may be reasonable to develop a process that, following an initial interviews, the confirmation the model matches the as-built, as-operated plant is confirmed through review or discussion with the system engineers - typically through a periodic review of the notebook performed in accordance with plant procedures.	Peer Review	Note: Resolution provided here is from Energy Northwest response was submitted under Letter GO2-15-145, dated October 29, 2015 (ML1530A492) Resolved: **YES** The open F&O, 2-14, for SR SY-A4 has been resolved to meet CC-II for SY-A4. Finding 2-14 was resolved as part of the risk-informed technical specification initiative 5b license amendment request. System reviews with the system engineers were completed for all PRA systems with a focus on confirming that the PRA system analyses correctly reflect the as-built, as-operated plant, as well as to discuss recent operating history and any problems in system operation. The interviews and reviews were documented, and this documentation will be added to the system notebooks in the next PRA update. Discrepancies identified by the system engineer interviews had no impact on the PRA modeling and were documentation-related only, with the exception of two modeling conservatisms related to the Reactor Feedwater system model and the Standby Liquid Control (SLC) model. The first model conservatism is the current Reactor Feedwater system PRA model only has two paths modeled for possible injection into the vessel when there are three possible paths. The second model conservatism is the current SLC system PRA model contains a component failure event that, based on system design, is conservative. Both of these modeling conservatisms do not have a risk-significant impact on the PRA results.

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 2-16 SupRs: SY-A14 SY-C2	Failure modes have been considered in development of the system models adequately. However, the exclusion of some failure modes was not adequately documented. Some failure modes could be important to system models. A sample review of the system models show that the following failure modes may not have been fully investigated: (d) failure of a closed component to remain closed especially for standby components where the failure would not be identified quickly - say months or greater). (f) failure of an open component to remain open (see above) (g) active component spurious operation (h) plugging of an active or passive component (i) leakage of an active or passive component (j) rupture of an active or passive component (k) internal leakage of a component (l) internal rupture of a component (m) failure to provide signal/operate (e.g., instrumentation) (n) spurious signal/operation For example, in the [Residual Heat Removal] RHR system model, valves RHR-V54A/B could have a failure mode for fail to remain closed, which is not included in the model. The exposure time on these valves would rely on the surveillance tests.	Consider adding documentation for both the inclusion and exclusion of the failure modes with justification for components included the system boundary. Consider adding more failure modes into the system models if the exclusion requires additional quantitative evaluations.	Peer Review	Resolved: **YES** This peer review finding was resolved. A Tier 3 calculation was prepared to ensure that failure modes have been considered in development of the system models adequately, with evaluation for both the inclusion and exclusion of the failure modes for components within the system boundary, including justification. Failure modes were added or corrected in the system models and system notebooks were updated based on this evaluation.

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 2-17 SupRs: S Y-A14 SY-C2	The following requirements associated with system modeling are determined to be not adequate: (e) actual operational history (such as [Significant Operating Experience Reports] SOERs) indicating any past problems in the system operation [not documented], (l) the components and failure modes included in the model and justification for any exclusion of components and failure modes [also see F&O 2-16], (q) the sources of the above information (e.g., completed checklist from walkdowns, notes from discussions with plant personnel) [also see F&O 2-14]. The following should be enhanced: (f) system success criteria and relationship to accident sequence models (k) assumptions or simplifications made in development of the system models. The inadequate documentation limits the review of the completeness of the system models.	Consider adding the documentation associated with items (e), (l) and (q). Also consider enhancing the documentation for the following two items: (f) system success criteria and relationship to accident sequence models (k) assumptions or simplifications made in development of the system models (i.e., provide a bulleted list of major assumptions in each system notebook).	Peer Review	Resolved: **YES** This peer review finding was resolved by enhancing system modeling documentation items associated with items (e), (l), (q), (f) and (k). For item (e), significant operating experience has been collected, but this information has not yet been incorporated into the system notebooks. This information will be incorporated into the system notebooks during future PRA update. (e) actual operational history (such as SOERs) indicating any past problems in the system operation was collected, and will be added to the system notebooks in a future PRA update, (f) documentation of system success criteria and relationship to accident sequence models was enhanced in the system notebooks, (k) documentation of assumptions or simplifications made in development of the system models was enhanced in the system notebooks, (l) the components and failure modes included in the model and justification for any exclusion of components and failure modes, and (q) the sources of the above information (e.g., notes from discussions with plant personnel) were documented.
F&O 2-18 SupR: LE-G6	The quantitative definition used for significant accident progression sequence was not documented.	Add the quantitative definition used for significant accident progression sequence in the LE notebooks.	Peer Review	Resolved: **YES** This peer review finding was resolved. Section 6.3 of the Level 2 Notebook was enhanced to add the quantitative definition used for significant accident progression sequence.

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 2-28 SupR: IFPP-B3	Sources of model uncertainty and related assumptions for flood plant partitioning were not documented in the flooding notebooks. Neither do the Internal Flooding items included in PSA-2-QU-0001 Tables 5-2 and 5-3. Sources of model uncertainty and related assumptions for flood plant partitioning are not documented.	Investigate and document the sources of model uncertainty and related assumptions for flood plant partitioning. Perform sensitivity studies if deemed necessary.	Peer Review	Resolved: **YES** The PRA Quantification Notebook Table 5-3, documents uncertainties and related assumptions for the overall PRA, including uncertainties and related assumptions for flood plant partitioning. This information was added to the IF notebooks for completeness. Sensitivity studies, other than the standard set of sensitivities, i.e., CCFs set to 5th and 95th percentile and HEPs set to 5th and 95th percentile, are not directed for the base PRA model per the EPRI Uncertainties/Assumptions methodology, 1016737, and were not performed.

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 3-1 SupRs: DA-D1	The process of developing plant-specific parameter data updates based on plant-specific experience and generic data is described in PSA-2-DA-0002, Bayesian Update of CGS PSA Data. The process is focused on significant basic events. However, the process used to determine which events are significant and should be Bayesian updated appears to be faulted. A set of screening criteria based on risk achievement worth, F-V, and Birnbaum importance measures is applied, but the criteria for determination of significant (i.e., per PSA-2-DA-0002, "RAW value of 3 is typically used in risk ranking initiatives to identify risk important components.") differs from those applied in risk-informed applications (e.g., Maintenance Rule, where RAW of 2 and F-V of 0.005 are normally used). In the DA screening, a RAW of 3 is used as the criterion, so it is possible that some significant basic events could be screened from consideration using the process in PSA-2-DA-0002 (i.e., in Table 4 there would be a lower value for the CDF change for which a failure would be a candidate for Bayesian updating). Capability Category 2 for SR DA-D1 requires that realistic parameter estimates be calculated for all significant basic events. The process used to define which events are significant is not adequately objective and uses criteria that are inconsistent with selection criteria used for risk-informed applications (e.g., MR), and may result in underestimating the set of significant events. After discussion with CGS PRA personnel, they performed a sensitivity evaluation to estimate the impact of using more appropriate screening criteria. This exercise identified one additional component type (RV/SV) for which there were no EPIX failures.	Consider using screening criteria more consistent with criteria used for important plant applications of the base PRA (e.g., Maintenance Rule) to ensure that potentially risk significant failures reflect plant experience.	Peer Review	Resolved: **YES** This peer review finding has been resolved. Screening criteria consistent with important plant applications of the base PRA are now used, per RG 1.200, Rev. 2 (footnote page 10: Significant basic event/contributor: The basic events (i.e., equipment unavailabilities and human failure events) that have a Fussell-Vesely (FV)importance greater than 0.005 or a risk-achievement worth greater than 2): - The identification of plant-specific parameter data updates now uses a risk-achievement worth (RAW) value greater than 2 as one screening criterion, and - A FV of greater than 5E-3 is typically used in risk ranking initiatives to identify risk-significant components. The identification of plant-specific parameter data updates now uses a FV of 1E-4, because it was practical to do so.

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 3-11 SupR: SY-A24	The accident sequence model includes top event EAC, recovery of onsite AC power. The text of the AS notebook refers to Appendix D of that notebook for the onsite power recovery calculation. Appendix D is a portion of ERIN letter C1069805-3919 "Completion of CGS PSA Model Modifications to Address PSA Certification Comments (P.O.00303454)", August 3, 1999. titled Emergency AC Power Recovery. That analysis uses as its basis a 1993 regulatory analysis, SECY-93-190. There are 2 issues with this. First, the basis for the onsite AC power recovery is a set of emergency diesel generator (EDG) repair data from before 1993, i.e., significantly more than 16 years old, and originally based on relatively few data points. Second, this represents credit for repair of failed equipment without checking against plant-specific experience. Credit should not be taken for repair of failed equipment, particularly EDGs, without sound plant-specific basis. Significance may be increased if tied to the F&O on consequential LOOP.	Provide a sound basis for the present-day validity of the repair probabilities, or remove the credit from the model.	Peer Review	Resolved: **YES** This peer review finding was resolved by enhancing Appendix D of the accident sequence notebook to further support that the existing EDG non-recovery values based on SECY-93-190 are judged to be appropriate for CGS. Further, the Emergency AC Power recovery values from SECY-93-190 used in the CGS PRA were compared with more recent data from NUREG/CR-6890. The SECY-93-190 data produces non-recovery probabilities that are higher relative to the NUREG/CR-6890 data. The following comparison is made: Recovery Time (hr),SECY-93-190,NUREG/CR-6890 0.5, 0.89, 0.86 2, 0.86, 0.65 3, 0.78, 0.56 4, 0.71, 0.48 6, 0.59, 0.37 10, 0.4, 0.24 15, 0.29, 0.14 24, 0.23, 0.24 The SECY-93-190 data provides higher non-recovery values for the entire range of EDG recovery times. Although the NUREG/CR-6890 diesel non-recovery data development is more recent, there is one prominent concern about the NUREG/CR-6890 data for purposes of the Columbia PSA modeling. The NUREG/CR-6890 non-recovery values assume that the diesel generator that is most straightforward to repair will be chosen for recovery. What is not clear from the NUREG/CR-6890 treatment is how the SPAR models address the possibility that the DG that is easiest to repair actually won't be restored because the associated SW pump or RHR pump is out of service. The Columbia model assumes a 50-50 percent likelihood for recovery of DG1 or DG2, unless the SW or RHR pump on one of the diesel generators is out of service. Due to this potential conflict, the NUREG-6890 data was not used. The SECY-93-190 data is judged to be the most applicable and realistic.

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 4-7 SupR: MU-C1	While consideration of pending model changes has occurred for important applications (e.g. DGAOT), there was no identified Columbia process that requires that pending changes be considered for applications. Having such a process is an explicit requirement of the standard.	Consider revising SYS-4-34 or other procedure/instruction to address this issue. A industry model process for performing Maintenance and Update is provided in the BWROG document TP-09-012 Living PRA Configuration Control and Model Maintenance, however use of this process is not a requirement of the ASME Standard.	Peer Review	Resolved: **<u>YES</u>** The peer review finding was resolved. A new Section 4.4 was added to the model maintenance and update procedure, SYS-4-34, that documents the Columbia process that requires pending model changes to be considered for applications. Also, a list of current applications that require a consideration of pending model changes in applications was added in SYS-4-34, Attachment 8.2.
F&O 6-10 SupR: SY-B1	A variety of common cause basic events exist in the model without documentation provided to substantiate their basis. For example, Table 1 of PSA-2-DA-0004 lists 13 CCF events with probabilities of zero related to relays, vacuum breakers, and nitrogen bottles. Table 1 also lists two events that are termed CCF place keepers but not true CCF events. These are not further discussed. Table 1 lists CCF events with apparently generic values (1E-6), but no discussion or justification is provided to substantiate the value. Table 1 lists CCF events for 5 of 7 ADS valves and 5 of 11 SRVs, with values of 1.24E-6, but no discussion / justification is provided. Table 2 lists an event (CRDSV--24567C8LL) for "8 of 8", but uses the NRC data for "2 of 2" instead of the closer "6 of 6" value. Upon review, CGS noted that this event is not needed in the model and will be removed in the next update.	Provide a basis for all CCF values, including events set to 0.0, based on the CCF analysis process.	Peer Review	Resolved: **<u>YES</u>** This peer review finding has been resolved. The documentation of CCF events has been refined and improved. The bases for all CCF values are provided, based on the CCF analysis process.

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
F&O 6-8 SupR: SY-B1	Several issues were identified in the review of CCF: 1) The process to develop common cause groups is not documented in the CCF NB and therefore justification is not provided for the grouping scheme. Per CGS, selection of CCF groups was performed following the guidance of NUREG/CR-5485 based on similarities in service conditions, environment, design or manufacturer, and maintenance. NUREG/CR-5485 is not currently referenced in PSA-2-DA-0004. 2) There are limited formulae documented in the CCF data package PSA-2-DA-0004. As a result, the bases for the calculations embedded in the spreadsheets are not fully documented. Discussions with CGS staff indicates the Alpha CCF calculations were performed incorrectly. Moreover, based on the formulae in NUREG/CR-5485, the CCF basic event probabilities may be slightly more conservative. 3) As a result of the above, CCF basic events are incorrectly calculated.	Add CCF grouping methodology to PSA-2-DA-0004, and add formulae for the CCF basic event probability calculations. Add reference in the data notebook to NUREG/CR-5485. Re-evaluated CCF values, based on the revised methods.	Peer Review	Resolved: **YES** This peer review finding has been resolved. The process to develop common cause groups is now documented in PSA-2-DA-004. CGS employs staggered testing of redundant trains modeled in the PRA. Therefore, CCF probabilities were recalculated using the CCF equations for staggered testing, and these equations are documented in PSA-2-DA-004, and the PRA was updated.
F&O 6-9 SupR: SY-B1	Per Section 2 of PSA-2-DA-0004, the common cause calculations are based on a non-staggered testing scheme based on the prior use of this scheme for the original PRA and the fact that current CGS test intervals / test information has not been developed for the current data update. PSA-2-DA-0004 notes that the non-staggered approach is conservative. The staggered approach should be utilized where appropriate to reflect plant practices.	Redevelop CCF approach using staggered approach where appropriate.	Peer Review	Resolved: **YES** This peer review finding has been resolved. CGS employs staggered testing of redundant trains modeled in the PRA. Therefore, CCF probabilities were recalculated using the CCF equations for staggered testing, and these equations are documented in PSA-2-DA-004, and the PRA was updated.

Table 2 Disposition of Findings from the 2009 Peer Review and Self-Assessment for Supporting Requirements Graded as CC-I or Not Met

Finding	Observations	Recommendations	Source	Resolution
SA-IF-E5a-1 SupR: IFQU-A6	No documentation was found of how scenario-specific impacts were addressed for operator actions taken in response to an internal flooding event. Supporting requirement IFQU-A6 requires the following: For all human failure events in the internal flood scenarios, INCLUDE the following scenario-specific impacts on PSFs for control room and ex-control room actions as appropriate to the HRA methodology being used: (a) additional workload and stress (above that for similar sequences not caused by internal floods) (b) cue availability (c) effect of flood on mitigation, required response, timing, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) (d) flooding-specific job aids and training (e.g., procedures, training exercises)	Review all HEPs for actions that could be taken following an internal flood and address how they would be impacted by the flood. Document this review and modify the analysis accordingly.	Self-Assessment	Resolved: **YES** This self-assessment finding has been resolved. The internal flooding HRA assesses scenario-specific impacts on PSFs, including additional workload and stress, cue availability, effect of flood on mitigation, required response, timing, and recovery activities flooding-specific procedures and training. These enhancements to the HRA are documented in the HRA Calculator and in the HRA Notebook.

1.c:

No PRA models other than the internal events PRA will be used for detailed quantitative analysis.

NRC Request

2. The LAR indicates that PRA models other than the internal events PRA model may be used. Please confirm that these PRA models reflect the current plant configuration and operation. If this is not the case, please explain how the PRA models support the application, using Nuclear Energy Institute (NEI) 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed method for Control of Surveillance Frequencies," April 2007 (ADAMS Accession No. ML071360456) guidance, and whether current plant configuration and operation is considered in their use.

Energy Northwest Response:

No PRA models other than the internal events PRA will be used for detailed quantitative analysis.

The qualitative assessment of fire risk and other external event risk will include a review of applicability to the current plant configuration and operation. For example, some STI change evaluations, per Step 10b qualitative reasoning and very low ΔCDF and ΔLERF results from the internal events analysis may be sufficient to support the STI change evaluation where Step 10b reads in part:

> *"Alternative evaluations for the impact from external events and shutdown events are also deemed acceptable at this point. For example, if the ΔCDF and ΔLERF values have been demonstrated to be very small from an internal events perspective based on detailed analysis of the impact of the SSC being evaluated for the STI change, and if it is known that the CDF or LERF impact from external events (or shutdown events as applicable) is not specifically sensitive to the SSC being evaluated (by qualitative reasoning), then the detailed internal events evaluations and associated required sensitivity cases (as described in Step 14) can be used to bound the potential impact from external events and shutdown PRA model contributors."*

Therefore, by following the NEI 04-10, Rev. 1 guidance, the evaluation of fire risk and other external events supporting this application will qualitatively reflect and consider the current plant configuration and operation.

NRC Request

3. The impact of the open F&O for supporting requirement (SR) SY-A4 states that sensitivity analysis will be performed. It is not clear how a sensitivity analysis could be defined to address the lack of documented interviews that confirm that system analyses represent the as-built, as-operated plant. The TSTF-425 program considers capability category II for the internal events PRA model; therefore, please address this F&O to meet capability category II and provide the disposition of the F&O.

[1]Energy Northwest Response:

Open F&O 2-14 as documented in the August 2009 peer review is as follows:

> *"Interviews with plant system engineers or operators have not been documented and cannot be verified by the peer review team. Original interviews were performed for the original IPE.*
>
> *System and operations change over time, and the system engineers and operators should be consulted with regard to system models.*
>
> *(This F&O originated from [supporting requirement] SR SY-A4)"*

The open F&O, 2-14, for SR SY-A4 has been resolved to meet capability category (CC) CC-II for SY-A4. Finding 2-14 was resolved as part of the risk-informed technical specification initiative 5b license amendment request. System reviews with the system engineers were completed for all PRA systems with a focus on confirming that the PRA system analyses correctly reflect the as-built, as-operated plant, as well as to discuss recent operating history and any problems in system operation. The interviews and reviews were documented, and this documentation will be added to the system notebooks in the next PRA update. Discrepancies identified by the system engineer interviews had no impact on the PRA modeling and were documentation-related only, with the exception of two modeling conservatisms related to the Reactor Feedwater system model and the Standby Liquid Control (SLC) model. The first model conservatism is the current Reactor Feedwater system PRA model only has two paths modeled for possible injection into the vessel when there are three possible paths. The second model conservatism is the current SLC system PRA model contains a component failure event that, based on system design, is conservative. Both of these modeling conservatisms do not have a risk-significant impact on the PRA results.

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[1] The response to question 3 was sent under letter GO2-15-145, Dated October 29, 2015 (ML1530A492)

NRC Request

4. The peer review F&O on SR DA-C6 is related to meeting the data requirements for standby components (SR DA-C6) as well as for surveillance requirements (SR DA-C7). The F&O states: "Estimates based on the surveillance tests and maintenance acts as described in DA-C6 and DA-C7 should be performed for significant components whose data are not tracked in the MSPI data." SR DA-C6 and SR DA-C7 include consideration of plant-specific data. Please explain the basis for concluding that the proposed sensitivity analyses, which are based on generic data, are considered bounding if these two SRs are graded at not met or capability category I. If use of plant-specific data consistent with SR DA-C6 and SR DA-C7 cannot be demonstrated to be bounding with respect to the proposed method to perform sensitivity analyses for relevant components, then please complete the work to meet SR DA-C6 and SR DA-C7 provide the disposition of the F&O.

Energy Northwest Response:

The 2009 peer review graded supporting requirements DA-C6 and DA-C7 as met for CC-II:

- DA-C6 Assessment: MET for Capability Categories I-III
- DA-C7 Assessment: MET for Capability Categories I-III

The SRs were met because the major components' maintenance activities were based mainly on MSPI data. Finding 2-2 applies to non-MSPI components that were based on estimates.

A sensitivity study was performed by replacing the base data for these failure modes with generic data from NUREG/CR-6928. It was determined that the finding is unlikely to change the conclusions of risk-informed decisions.

Although Finding 2-2 is open, the SRs are graded at met for CC-II and the impact is unlikely to affect the results.

NRC Request

5. Do the failure probabilities of structures, systems, and components modeled in the CGS internal events PRA include a standby time-related contribution and a cyclic demand-related contribution? If not, please describe how standby time-related contribution is addressed for extended intervals.

Energy Northwest Response:

The failure probabilities of structures, systems, and components modeled in the CGS internal events PRA include either a standby time-related contribution or a demand-related contribution.

The standby time-related failures will be evaluated in accordance with NEI-04-10, Revision 1 by direct change in the test interval for those SSCs that include a standby periodically tested failure mode along with the appropriate adjustments to common cause failure events. For demand-related events an appropriate time-related failure contribution will be determined for each component that is uniquely impacted by the proposed STI change to obtain the maximum test-limited risk contribution.

3.2 Response dated April 7, 2016 (ML160984A387) to March 9, 2016 Request for Additional Information (ML16069A359)

NRC Request PRA RAI 1.1 (Follow up to PRA RAI 1)

In response to PRA RAI 1, the licensee provided the internal events PRA 2009 peer review facts and observations (F&Os), self-assessment findings, and their dispositions. Please address the following regarding F&O 1-3.

a. The peer review F&O 1-3 states "In almost all of the post-initiator Human Error Probabilities (HEPs) where optimal stress is assumed, time is a factor with core damage occurring between 30 minutes and an hours."

   Clarify if these post-initiator HEPs were assumed to be optimal stress in all cases involving this time frame or if such HEPs were determined on a case-by-case basis to represent optimal stress or high stress as appropriate. Please describe your process to make this determination.

b. The peer review also observed, "A second set of justification was provided, with discussion that basically justified moderate or low stress would be appropriate, given enough training for the operators." The peer review recommendation stated: "Apply high stress factors per Table 17-1 of NUREG/CR-1278 to HEPs where time pressure is present during an accident situation."

   Please clarify if high stress factors were used on a case-by-case basis from this table, and, if not provide justification. Please discuss whether time pressure was used to determine whether an execution operator action justified the application of a high stress factor.

Energy Northwest Response:

As part of the Rev. 7.2 PRA update performed in 2014, the assignments of stress levels were reviewed for all Columbia Generating Station (CGS) post-initiator human failure events, and all post-initiator human failure events now utilize the stress levels recommended by the Human Reliability Analysis (HRA) Calculator. The stress level recommended by the HRA calculator was reviewed by the HRA analyst and in some cases increased to a higher stress level. In this manner, the full intent of F&O 1-3 has been resolved.

By utilizing the HRA Calculator, the stress levels assigned to the CGS HFEs follow the NUREG/CR-1278 guidelines, including addressing the impact to stress levels from time pressure.

NRC Request PRA RAI 4.1 (Follow up to PRA RAI 4)

The NRC staff reviewed the response to PRA RAI 4 and found the response did not address the issue. The peer review observation in the LAR states:

"Estimates based on the surveillance tests and maintenance acts as described in DAC6 and DA-C7 should be performed for significant components whose data are not tracked in the MSPI [Mitigating Systems Performance Index] data."

The peer review recommendations states:

"Update the estimates for significant events based on surveillance test and maintenance records."

The response to PRA RAI 4 states:

"A sensitivity study was performed by replacing the base data for these failure modes with generic data from NUREG/CR-6928. It was determined that the finding is unlikely to change the conclusions of risk-informed decisions."

Supporting requirements (SRs) DA-C6 and DA-C7 are not limited to the MSPI systems. SR DA-C6 and SR DA-C7 include consideration of plant-specific data, and it is not clear how generic data has been demonstrated to be bounding for the non-MSPI components. That is, if the peer review recommendation were to be performed it is not clear that the generic data would necessarily be bounding.

Please provide justification why the use of generic data for performing sensitivity analyses is bounding without updating applicable component estimates with plant specific data consistent with SR DA-C6 and SR DA-C7, or compete the work to meet

SR DA-C6 and SR DA-C7 for applicable components and provide the disposition of the F&O.

Energy Northwest Response:

Finding F&O 2-2 against SR DA-C6 and SR DA-C7 from the 2009 peer review has been resolved. The disposition of the F&O is as follows:

Estimates for significant events not tracked in the Mitigating Performance System Index (MSPI) data are now based on plant-specific surveillance test and maintenance records. Resolution of F&O 2-2 impacts the following component failure modes:

- C---W2 (compressor fails to start),
- C---W4 (compressor fails to run),
- FN--R3 (fan fails to start),
- FR--W4 (fan fails to run),
- AHUS---S3 (air handling unit fails to start),
- AHUR---W4 (air handling unit fails to run).

The resolution to this F&O has been incorporated into a working model. This working model will be used for all Risk informed Technical Specification (RITS) 5B assessments until a model of record revision is released which contains the resolution to this F&O.

3.3 Response dated June 22, 2016 (ML16174A432) to May 31, 2016 Request for Additional Information (ML16152A737)

NRC Request:

The licensee is asked to resubmit the response to RAI question PRA RAI 1.1 a) submitted to the licensee on March 9, 2017 which states;

a. The peer review F&O 1-3 states “In almost all of the post-initiator Human Error Probabilities (HEPs) where optimal stress is assumed, time is a factor with core damage occurring between 30 minutes and an hours.”

Clarify if these post-initiator HEPs were assumed to be optimal stress in all cases involving this time frame or if such HEPs were determined on a case-by-case basis to represent optimal stress or high stress as appropriate. Please describe your process to make this determination.

Energy Northwest Response:

These post-initiator HEP’s were not assumed to be optimal stress in all cases, including the time frame of 30 minutes and an hour.

Post-initiators HEP stress levels were determined on a case-by-case basis to represent optimal (Low), moderate stress or high stress as appropriate, including those where time is a factor with core damage occurring between 30 minutes and an hour. The determination of stress was made using the Execution Stress decision tree in the Human Reliability Analysis (HRA) Calculator.