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January 2, 2018
GO2-17-186

10 CFR 50.90

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

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATED TO LICENSE AMENDMENT REQUEST - REVISE TECHNICAL SPECIFICATION 5.5.12 FOR PERMANENT EXTENSION OF TYPE A TEST AND TYPE C LEAK RATE TEST FREQUENCIES**

References:

1. Letter, GO2-17-009, A.L. Javorik (Energy Northwest) to NRC, "License Amendment Request - Revise Technical Specification 5.5.12 for Permanent Extensions of Type A Test and Type C Leak Rate Test Frequencies," dated March 27, 2017 (ADAMS Accession Number ML17086A586)

2. Email, Klos (NRC) to L.L. Williams (Energy Northwest), "RAIs for MF9469, EPID: L-2017-LLA-0197, ILRT, LLRT LAR," dated October 13, 2017 (ADAMS Accession Number ML17286A575)

3. Email, Klos (NRC) to S.J. Christianson (Energy Northwest), "RAIs for MF9469, EPID: L-2017-LLA-0197, ILRT, LLRT LAR - new RAI response date," dated December 7, 2017 (ADAMS Accession Number ML17341B027)

Dear Sir or Madam:

By Reference 1 Energy Northwest submitted a License Amendment Request to revise the Columbia Generating Station (Columbia) Technical Specification (TS) 5.5.12 for permanent extension of Type A and Type C leak rate test frequencies. By Reference 2 the Nuclear Regulatory Commission (NRC) requested additional information related to the Energy Northwest submittal. The enclosure to this letter contains the requested information.

The No Significant Hazards Consideration Determination (NSHCD) provided in the original submittal is not altered by this submittal. No new commitments are being made by this letter or the enclosure.

If there are any questions or if additional information is needed, please contact Ms. L. L. Williams, Licensing Supervisor, at 509-377-8148.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 2nd day of January, 2018.

Respectfully,

A. L. Javorik
Vice President, Engineering

Enclosure: As stated

cc: NRC RIV Regional Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
CD Sonoda – BPA/1399 (email)
EFSECutc.wa.gov – EFSEC (email)
RR Cowley – WDOH (email)
WA Horin – Winston & Strawn

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

ENERCON

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Response to RAIs for the ILRT LAR

Columbia Generating Station

Revision 2

Prepared by: Joseph Lavelline Digitally signed by Joseph Lavelline DN: cn=Joseph Lavelline, o=Enercon Services, ou, email=jlavelline@enercon.com, c=US Date: 2017.12.27 16:10:03 -05'00'

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Eric Jorgenson, ENERCON Services

Reviewed by: 12/27/17

Habib Shtaih, Energy Northwest

Approval by: 12/27/17

Blake Smith, Energy Northwest

Response to RAIs for ILRT Permanent Surveillance Interval Extension

By letter dated March 27, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17086A586), Energy Northwest (the licensee) submitted a License Amendment Request (LAR) to revise the Columbia Generating Station (Columbia) Technical Specification (TS) 5.5.12 for permanent extension of Type A and Type C Leak Rate Test Frequencies to 15 years (180 months) and 75 months respectively.

The Probabilistic Risk Assessment Licensing Branch has reviewed the LAR and has identified areas where additional information is needed to complete its review. Responses to Request for Additional Information (RAI) 01 through 04 are provided below.

REFERENCES

1. Columbia PRA Peer Review Report Using American Society of Mechanical Engineers (ASME) PRA Standard Requirements, Boiling Water Reactor (BWR) Owners' Group, 2009
2. MSE-EJJ-11-02, "Columbia Generating Station Regulatory Guide 1.200 Compliance Database"
3. FPSA-1-RE-0001, "Columbia Generating Station Fire Probabilistic Safety Assessment Quantification and Results," Revision 2, dated December 1, 2006
4. SPSA-1-SE-0001, "Seismic Probabilistic Safety Assessment," Revision 1, dated February 1, 2007
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," dated February 2, 2009
6. RG 1.200, "An Approach for Determining the Technical Adequacy Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009
7. "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," Electrical Power Research Institute (EPRI), Palo Alto, CA: 2012, 1026511
8. Letter, GO2-15-045, D.A. Swank (Energy Northwest) to Nuclear Regulatory Commission (NRC), "Seismic Hazard and Screening Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(F) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2015
9. FRANX, R&R Workstation, EPRI, Version 4.2.0.4, 2013
10. NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," dated August 1983
11. IPEEE-2-RE-0001, "Re-examination of the external events evaluation in the IPEEE," Revision 0, dated 2017

12. 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

13. PSA-2-HR-0001, “Human Reliability Assessment,” Revision 3, dated July 6, 2016

14. “Flood Hazard Reevaluation Report, Response to the 50.54(f) information request regarding near-term task force recommendation 2.1: Flooding for the Columbia Generating Station,” ENERCON Services, Inc., Revision 0, dated September 27, 2016

15. “An Approach to Human Reliability Analysis for External Events with a Focus on Seismic,” Electrical Power Research Institute (EPRI), Palo Alto, CA: 2016, 3002008093

PRA RAI 1 – Internal Events PRA Technical Adequacy

The License Amendment Request (LAR) states that the Columbia Generating Station (CGS) internal events PRA last underwent a peer review in 2009 against the American Society of Mechanical Engineers (ASME)/ American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009, as clarified by Regulatory Guide (RG) 1.200, Revision 2. The NRC staff notes that the "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" (ADAMS Accession Number ML16313A573) states that the 2009 peer review was a full-scope peer review against ASME/ANS RA-Sa-2009 and RG 1.200, Revision 2, using the industry peer review process guidelines in NEI-05-04, Revision 2.

a. *Confirm that the 2009 peer review included a full-scope peer review of the internal flooding PRA model, otherwise provide the necessary information to assess the quality of the internal flooding PRA model against RG 1.200, Revision 2, including history of peer reviews, Facts and Observations (F&Os) and their resolution or disposition for the application.*
b. *Describe the changes made to the internal events and internal flooding PRA since the last full-scope peer review. This description should be of sufficient detail to assess whether these changes are PRA maintenance or PRA upgrades as defined in Section 1-5.4 of the PRA Standard. Since the following may indicate a PRA upgrade, include in your discussion: any new methodologies, changes in scope that impacts the significant accident sequences or the significant accident progression sequences, changes in capability that impacts the significant accident sequences or the significant accident progression sequences.*
c. *Indicate, and provide justification, whether the changes described in Part b are PRA maintenance or PRA upgrades as defined in Section 1-5.4 of the PRA Standard.*
d. *Indicate whether focused-scope peer review(s) has been performed for those PRA upgrades identified in Part 2. As applicable, provide a list of the F&Os from the peer review(s) that do not meet the appropriate Capability Category in accordance with EPRI TR 1021467-A, and explain how the F&Os were dispositioned for this application. If focused-scope peer review(s) have not been performed for these PRA upgrades, then provide a quantitative evaluation (e.g., sensitivity or bounding analysis) of its effect on the application until a focused-scope peer review can be completed.*

Response

The requested information is provided below.

a. The Columbia 2009 peer review included a full-scope review of the internal flooding Probabilistic Risk Assessment (PRA) model.
b. Following the Columbia 2009 full-scope peer review, three PRA updates have been performed of the Columbia internal events and internal flooding PRAs, PRA Revisions 7.1, 7.2, and minor PRA Revision 7.2.1. See Tables RAI01-1, RAI01-2, and RAI01-3 for descriptions of model changes for these PRA updates. Tables RAI01-1, RAI01-2, and RAI01-3 document, as applicable, the use of any new methodologies, changes in scope that impact the significant accident sequences or the significant accident progression sequences, and / or changes in capability that impact the significant accident sequences or the significant accident progression sequences.

c. See Tables RAI01-1, RAI01-2, and RAI01-3 for descriptions of model change classifications and rationale. All PRA model changes performed since the peer review constitute model maintenance.
d. All PRA model changes performed since the peer review constitute model maintenance; therefore, no focused-scope peer review was required for the Columbia PRA.

Table RAI01-1 Model Changes Performed for PRA Revision 7.1

Source	F&O[1]	SR[2]	Level	Model Change	Classification	Rationale
Peer Review 2009	1-11	IE-C14	Suggestion	The interface loss of coolant accident (ISLOCA) modeling was revised for PRA Rev. 7.1 to resolve this peer review suggestion. Cutsets containing multiple pre-initiator operator failures were determined to not be appropriate (they are demand-related, and do not represent frequencies). No cutsets contain multiple pre-initiator operator failures in the revised ISLOCA initiating event model. All cutsets now appropriately represent yearly frequencies.	PRA Maintenance	In response to a peer review suggestion, this change consisted of correcting a model error. This change has no significant impact on risk insights.
Peer Review 2009	1-14	IE-C14	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)	PRA Maintenance	In response to a peer review finding, this change corrected a model error.

[1] Finding & Observation (F&O)
[2] Supporting Requirement (SR)

Table RAI01-1 Model Changes Performed for PRA Revision 7.1

Source	F&O[1]	SR[2]	Level	Model Change	Classification	Rationale
				probability has been changed to a 0.1 based on documentation provided with the Columbia SPAR model. From the Columbia SPAR model documentation: 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		

Table RAI01-1 Model Changes Performed for PRA Revision 7.1

Source	F&O[1]	SR[2]	Level	Model Change	Classification	Rationale
				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 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.		
Peer Review 2009	1-15	AS-A5	Finding	To address this peer review finding, the TIA initiating event is now modeled to impact systems for only loss of non-safety air. The loss of non-safety and safety air is not relevant as an initiating event, since the loss of non-safety air itself causes a reactor trip. The potential loss of safety-related air is now modeled as part of the TIA accident sequence development. Depressurization is included in the updated TIA event tree, as safety-related air can support this function.	PRA Maintenance	In response to a peer review finding, this change corrected a model error.

Table RAI01-1 Model Changes Performed for PRA Revision 7.1

Source	F&O[1]	SR[2]	Level	Model Change	Classification	Rationale
Peer Review 2009	1-17	DA-A3	Suggestion	To address this peer review suggestion, the distribution or failure types were corrected for specific type codes: the running failure for compressor was corrected to be failure-to-run-related rather than demand-related; standby pump and air handling unit failures-to-run, as well as air-operated valve (AOV) spurious operation were corrected to be a gamma distribution; AOV failures to open / closed were changed to demand-related failures.	PRA Maintenance	In response to a peer review suggestion, this change corrected a data error.
Peer Review 2009	1-23	HR-D2	Finding	Procedure-specific pre-initiator human error probability (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 Columbia 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"	PRA Maintenance	In response to a peer review finding, this change enhances completeness.
Peer Review 2009	1-24	HR-G7	Suggestion	The Columbia PRA human reliability analysis (HRA) dependency analysis was performed/updated using the HRA calculator dependency module. This addresses the concern cited by this peer review suggestion regarding missing human failure events (HFE) combinations.	PRA Maintenance	In response to a peer review suggestion, this change enhances completeness.

Table RAI01-1 Model Changes Performed for PRA Revision 7.1

Source	F&O[1]	SR[2]	Level	Model Change	Classification	Rationale
Peer Review 2009	1-28	SY-A11	Finding	The need for gland seal cooling in the event of a loss of reactor core isolation cooling (RCIC) pump room cooling was added to the pump cooling dependency model. Miscalibration of RCIC leak detection temperature sensors could cause a spurious RCIC isolation, and miscalibration basic events are now modeled in the appropriate fault tree locations for the failing closed of RCIC valves 8 and 63. The modeling of temperature sensor miscalibration is not applicable to the RCIC pump cooling dependency logic, and was therefore removed.	PRA Maintenance	In response to a peer review finding, this change corrected an omission. This change has no significant impact on risk insights.
Peer Review 2009	1-32	QU-E4	Suggestion	For the ISLOCA initiating event development: 1) ISLOCA pathways up-to-and-including three valves were included in the ISLOCA evaluation, and 2) common cause internal leakage and rupture was included wherever applicable.	PRA Maintenance	In response to a peer review suggestion, this change corrected PRA model omissions. This change has no significant impact on risk insights.
Peer Review 2009	1-34	LE-E1	Suggestion	A basic event had two different labels in the PRA model. These two labels were consolidated into one consistent label for the basic event.	PRA Maintenance	In response to a peer review suggestion, this change corrected a model error. This change has no significant impact on risk insights.
Peer Review 2009	1-42	HR-I2	Finding	The Columbia PRA HRA dependency analysis now includes dependent HRA development for the Level II PRA. The Level I HEP dependency analysis was again updated for Rev. 7.2	PRA Maintenance	In response to a peer review finding, this change corrected a PRA model omission. The same modeling techniques as those in the Level I model were used.
Peer Review 2009	1-43	LE-D4	Finding	The residual heat removal (RHR)-V-8 and RHR-V-9 have interlocks that prevent valve opening at normal reactor pressure vessel 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.	PRA Maintenance	In response to a peer review finding, this change corrected a model error. This change has no significant impact on risk insights.
Peer Review 2009	1-45	SY-A10	Suggestion	The modeling for alternating current (AC) bus transfers was revised to model the direct current (DC) power dependency solely on associated battery in the fault tree logic, and not the associated battery charger.	PRA Maintenance	In response to a peer review suggestion, this change corrected a model error.

Table RAI01-1 Model Changes Performed for PRA Revision 7.1

Source	F&O[1]	SR[2]	Level	Model Change	Classification	Rationale
Peer Review 2009	2-12	SY-B5	Finding	Consequential loss of offsite power (LOOP) modeling, as a single basic event, has been added to the Columbia PRA model.	PRA Maintenance	In response to a peer review finding, this change corrects an omission. This change has no significant impact on risk insights.
Peer Review 2009	2-16	SY-B13	Finding	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.	PRA Maintenance	In response to a peer review finding, this change enhanced completeness. This change has no significant impact on risk insights.
Peer Review 2009	2-26	IFEV-A5	Finding	The following resolutions have been made: a. The formula in the spreadsheet cell for flood frequency calculation, D2*'Mean Pipe Failure Rates'!$D$110*'Mean Pipe Failure Rates'!$D$148, reflects the standby service water (SW) failure rate and the impact factor. This was noted to be incorrect since the flooding scenario is associated with the plant service water (TSW) piping break. This error has been corrected. b. The integrity management data used for IE-FLDC502TSW-U in the Initiating Events Frequency Development Report, PSA-2FL-0003, was incorrect. The equation in the flood frequency calculation spreadsheet and the value in PSA-2FL-0003 have been corrected. c. The walkdown sheet in Walkdown Summary Report, PSA-3-FL-0001, for area C502 was determined to be incomplete, which missed the 100', 3" TSW pipe listed in PSA-2-FL-0001, Table 4. Energy Northwest verified this TSW pipe length in the field and added it to the walkdown sheet for this zone. d. Further, all formulas in the flood initiating event spreadsheet were reviewed for correctness. A total of five frequencies were found to be incorrect and were revised. Only one of these initiators increased in frequency.	PRA Maintenance	In response to a peer review finding, this change corrected initiating event frequency errors. This change has no significant impact on risk insights.

Table RAI01-1 Model Changes Performed for PRA Revision 7.1

Source	F&O[1]	SR[2]	Level	Model Change	Classification	Rationale
Peer Review 2009	4-5	AS-A2	Suggestion	Credit for condensate booster injection was removed from the Medium loss of cooling accident (LOCA) event tree.	PRA Maintenance	In response to a peer review suggestion, this change corrected a model error. This change has no significant impact on risk insights.
Peer Review 2009	6-12	LE-C6	Suggestion	The exposure periods for containment isolation valves were reviewed and corrected as applicable.	PRA Maintenance	In response to a peer review suggestion, this change corrected a data error. This change has no significant impact on risk insights.
Peer Review 2009	6-14	LE-B2	Finding	The likelihood for wetwell failure was revised to be consistent for the Level 1 and Level 2 PRAs. The existing structure analysis (ME-02-91-77) is appropriate and applicable analysis. The Level 2 PRA had inappropriately credited the biological shield wall to reduce the likelihood for wetwell failure.	PRA Maintenance	In response to a peer review suggestion, this change corrected a modeling error.
Peer Review 2009	6-8	SY-B1	Finding	The process to develop common cause groups is now documented in PSA- 2-DA-0004. Columbia 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-0004, and the PRA was updated.	PRA Maintenance	In response to a peer review finding, this change corrected a data development error. This change involves no new data update methods.
Peer Review 2009	6-9	SY-B1	Finding	Columbia 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-0004, and the PRA was updated.	PRA Maintenance	In response to a peer review finding, this change corrected a data development error. This change involves no new data update methods and no change in scope or capability.
Self-Assessment	SA2-15	SY-A11	Self-Assessment	Logic for the condensate storage tank (CST) and failure modes of the tank isolation valves was corrected.	PRA Maintenance	This change corrected a model error. This change has no significant impact on risk insights.
Self-Assessment	SA2-29	AS-B3	Self-Assessment	Credit was removed for RCIC operation for medium and large reactor feedwater (RFW) LOCAs outside containment, based on insufficient steam pressure for operation of the RCIC pump turbine.	PRA Maintenance	This change corrected a model error. This change has no significant impact on risk insights.

Table RAI01-1 Model Changes Performed for PRA Revision 7.1

Source	F&O[1]	SR[2]	Level	Model Change	Classification	Rationale
Self-Assessment	SA2-36	AS-A7	Self-Assessment	The possibility for RPV depressurization was added to Sequence S14 of the loss of offsite power (LOOP) event tree.	PRA Maintenance	This change enhances completeness. This change has no significant impact on risk insights.
Self-Assessment	SA2-37	AS-A7	Self-Assessment	The PP event tree node (no stuck open safety relief valve (SRV) after reaching heat capacity temperature limit) was removed from station black out (SBO) sequences in which diesel generator (DG)-3 cross tie succeeds, as there is much less potential for a stuck-open safety relief valve (SORV). Success of the crosstie represents a controlled depressurization, with much less potential for a SORV.	PRA Maintenance	This change enhances completeness. This change has no significant impact on risk insights.
Self-Assessment	SA2-46	IE-A5	Self-Assessment	Partial LOOP was added to the modeling of switchgear E-SM-7 and E-SM-8. Partial LOOP represents loss of offsite power to one switchgear, E-SM-7 or E-SM-8.	PRA Maintenance	This change enhances completeness. This change has no significant impact on risk insights.
Self-Assessment	SA2-8	HR-D1	Self-Assessment	Backup cooling of the control rod drive (CRD) pumps from condensate (COND)-P-3A was added to the model.	PRA Maintenance	This change enhances completeness. This change has no significant impact on risk insights.
Self-Assessment	SA-10-QU-A1	QU-B1	Self-Assessment	The PRA was converted from a WinNUPRA solution to a CAFTA solution.	PRA Maintenance	The cutsets produced from the CAFTA solution were reconciled with the cutsets produce from the WinNUPRA model. This reconciliation constituted a thorough internal review and demonstrated that there was no significant impact on risk insights.
Peer Review 2009	3-9	SY-A6	Finding	Failure of makeup from the fuel oil makeup from the individual fuel oil storage tanks to the respective electrical diesel generator (EDG) fuel oil day tank was added as part of the EDG logic.	PRA Maintenance	In response to a peer review finding, this change enhanced completeness. This change has no significant impact on risk insights.

Table RAI01-2 Model Changes Performed for PRA Revision 7.2

Source	F&O	SR	Level	Model Change	Classification	Rationale
Peer Review 2009	1-3	HR-G3	Finding	The assignments of stress levels were reviewed for all Columbia post-initiator human failure events, and all post-initiator human failure events now utilize the stress levels recommended by the HRA Calculator. On a case-by-case basis the next higher stress level, above the stress level recommended by the HRA Calculator was selected, based on HRA analyst judgment.	PRA Maintenance	In response to a peer review finding, this change corrected a model error
Self-Assessment	217963	SY-B4	Self-Assessment	CCF probabilities were reviewed and corrected as applicable. Also, common cause failures-to-run and failures-to-start probabilities were also separated into separate basic events, where applicable, where these failures were previously modeled by single combined CCF basic events.	PRA Maintenance	This change corrected common cause failure modeling and data errors.
Self-Assessment	279126	SY-A6	Self-Assessment	The dependency of switchgear E-SM-4 on diesel building mixed-air air handling unit DMA-AH-31 was removed from the PRA model. DMA-AH-31 supports the HPCS diesel generator, but it doesn't not support E-SM-4.	PRA Maintenance	This change corrected a model error.
Self-Assessment	SA-14-QU-B3-1	QU-B3	Self-Assessment	For Rev. 7.2, a truncation sensitivity was performed for the core damage frequency (CDF) and large early release frequency (LERF) quantifications to validate the CDF and LERF PRA truncation levels. The sensitivity demonstrates appropriate convergence of the solution for the selected truncation level.	PRA Maintenance	This is a necessary step in the PRA maintenance process. The same modeling techniques were used.
Peer Review 2009	6-3	IE-C4	Finding	The Bayesian updating of the turbine trip, loss of feedwater and loss of condenser initiating event frequencies TT, TF and TC Bayesian updates were corrected.	PRA Maintenance	Based on a peer review finding, this change corrected a data development error. This change involves no new modeling techniques and no change in scope or capability.
Self-Assessment	8076	SC-A3	Self-Assessment	The success criteria for standby liquid control was changed from 2-of-2 pumps to 1-of-2 pumps based on a plant change.	PRA Maintenance	The PRA model was changed to reflect a plant design change.

Table RAI01-2 Model Changes Performed for PRA Revision 7.2

Source	F&O	SR	Level	Model Change	Classification	Rationale
Self-Assessment	8772	HR-G3	Self-Assessment	The human failure probability development for failure to open the reactor feedwater bypass loop, RFWHUMN-V109H3LL, was revised to conform to a plant design change. The switch for the valve was moved from a back panel to a front panel.	PRA Maintenance	One human error probability was changed to reflect a plant design change. There was no change in plant risk.
Self-Assessment	HPCS-1A	AS-A3	Self-Assessment	Credit for injection after containment failure was added to additional accident sequences.	PRA Maintenance	This change enhances completeness and the same modeling techniques were used.
Self-Assessment	HPCS-1B	AS-A5	Self-Assessment	Credit for late depressurization, and late emergency core cooling systems (ECCS) or alternate injection was added to accident sequences.	PRA Maintenance	This change enhances completeness and the same modeling techniques were used.
Self-Assessment	HPCS-2D	HR-G3	Self-Assessment	The HEP for failure to align the service water cross-tie, RHRHUMNSWCRTIELL, now uses the CBDTM, rather than a conservative screening value.	PRA Maintenance	This change enhances completeness and utilizes an existing HRA calculation option. No new modeling techniques were used.
Self-Assessment	SA14-AS-A1-05	AS-B1	Self-Assessment	Model logic was corrected to remove credit for injection post-containment-failure for sequences in which HPCS is unavailable.	PRA Maintenance	This change corrected a model error. This change has no significant impact on risk insights.
Self-Assessment	SA14-AS-A3-11	AS-A3	Self-Assessment	Model logic was corrected to remove credit HPCS after containment failure for the large LOCA sequence in which HPCS is unavailable.	PRA Maintenance	This change corrected a model error. There was no change to the PRA results.
Self-Assessment	SA14-AS-B1-14	AS-B1	Self-Assessment	Model logic was corrected to remove credit for power control system (PCS) for the stuck-open safety relief valve anticipated transient without scram (ATWS) event tree.	PRA Maintenance	This change corrected a model error. The same modeling techniques were used.
Self-Assessment	SA14-AS-B6-16	AS-B6	Self-Assessment	Completeness was enhanced for cutset post-processing for loss of offsite power loss cutsets in which RCIC fails due to failure of emergency diesel generator, (EDG)-1 to start or run. Where offsite power recovery has not be otherwise credited, an offsite power non-recovery term was added.	PRA Maintenance	This change enhances completeness. This change has no significant impact on risk insights.

Table RAI01-2 Model Changes Performed for PRA Revision 7.2

Source	F&O	SR	Level	Model Change	Classification	Rationale
Self-Assessment	SA14-HR-G1-29	HR-G3	Self-Assessment	The CBDTM HRA Calculator option was selected for 64 HFEs that had previously utilized the combination-sum (ASEP + CBDTM) option.	PRA Maintenance	The change removed the use of the HRA Calculator combination-sum option. There are no new techniques involved.
Self-Assessment	SA-14-HR-A1-1	HR-G3	Self-Assessment	The HCR/ORE HRA Calculator option was selected for 23 HFEs that had previously utilized the combination-sum (ASEP + CBDTM) calculation option.	PRA Maintenance	The change involved selecting the HRA Calculator HCR/ORE option for applicable human failure events. There are no new techniques involved.
Self-Assessment	SA14-IE-A1-41	IE-A1	Self-Assessment	Loss of HPCS charger was removed from the manual shutdown initiating event logic.	PRA Maintenance	This change corrected a modeling error. This change has no significant impact on risk insights.
Self-Assessment	SA14-IE-C2-42	IE-C2	Self-Assessment	The initiating event Bayesian update evaluations were updated.	PRA Maintenance	This change reflects new information on plant performance (new data).
Self-Assessment	SA14-IE-C8-52	IE-C8	Self-Assessment	Loss of E-PP-7AA was recategorized and modeled as causing a main steam isolation valve (MSIV) closure initiating event, rather than as a manual shutdown.	PRA Maintenance	This change corrected a model error. This change has no significant impact on risk insights.
Self-Assessment	SA14-IFSN-A10-59	IFSN-A10	Self-Assessment	Corrected the internal flooding logic to fail balance of plant motor control center E-MC-1B for an additional seven flood scenarios.	PRA Maintenance	This change corrected a model error. This change produced no change to the risk results.
Self-Assessment	SA14-IFSN-A10-60	IFSN-A10	Self-Assessment	Failure probabilities for reactor building basement floor drain isolation valves were changed to utilize appropriate failure data rather than failing with a probability of 1.0.	PRA Maintenance	This change was made to address a change of plant procedures. There are no new techniques involved.
Self-Assessment	SA14-IFSN-A10-61	IFSN-A10	Self-Assessment	The consequential loss of the CST was add to the internal flooding model if the flooding source involves a pipe connected to the CST.	PRA Maintenance	This change enhances completeness. This change has no significant impact on risk insights.
Self-Assessment	SA14-SY-A1-111	SY-A1	Self-Assessment	Power dependencies were added for reactor building heating ventilation and air conditioning (HVAC) isolation valves.	PRA Maintenance	This change corrected a model error. There was no change to the PRA results.
Self-Assessment	SA14-SY-A11-112	SY-A11	Self-Assessment	The main steam relief valve nitrogen accumulators were added to the model.	PRA Maintenance	This change enhances completeness. There was no change to the PRA results.

Table RAI01-2 Model Changes Performed for PRA Revision 7.2

Source	F&O	SR	Level	Model Change	Classification	Rationale
Self-Assessment	SA14-SY-A1-93	SY-A1	Self-Assessment	The detail of the drywell cooling model was enhanced.	PRA Maintenance	This change enhances completeness. This change has no significant impact on risk insights.
Self-Assessment	SA14-SY-A1-94	SY-A1	Self-Assessment	The dependency of CRD on the CST was added to the model.	PRA Maintenance	This change enhances completeness. This change has no significant impact on risk insights.
Self-Assessment	SA14-SY-A1-99	SY-A1	Self-Assessment	The conditional probability (0.5) that plant service water pump A or B is running was added to the model.	PRA Maintenance	This change enhances completeness. There was no change to core damage frequency (CDF) and LERF.
Self-Assessment	SA14-SY-A3-120	SY-A3	Self-Assessment	The possibility for plant service water (TSW) backflow through a failed TSW pump was added to the model.	PRA Maintenance	This change enhances completeness. This change has no significant impact on risk insights.
Self-Assessment	SA14-SY-A3-122	SY-A3	Self-Assessment	HVAC dependencies were added for E-MC-7BB and E-MC-8BB.	PRA Maintenance	This change corrected a model error. This change has no significant impact on risk insights.
Self-Assessment	SA14-SY-A6-126	SY-A6	Self-Assessment	Modeling of the RCIC lube oil system was removed. The lube oil system is now part of the RCIC pump system boundary.	PRA Maintenance	This change corrected a model error. This change has no significant impact on risk insights.
Self-Assessment	SA14-SY-A17-1	SY-A17	Self-Assessment	Credit for local alignment of containment venting was added to the model.	PRA Maintenance	This change addresses a plant procedure change. There were no new methods involved.
Self-Assessment	SA14-SY-B1-127	SY-B1	Self-Assessment	Failure modes were corrected for the plant service water (balance of plant) discharge valves.	PRA Maintenance	This change corrected a model error. This change has no significant impact on risk insights.
Self-Assessment	SA14-SY-C1-128	SY-C1	Self-Assessment	The HPCS model was corrected to credit pump suction from the CST for LOCAs.	PRA Maintenance	This change corrected a model error. This change has no significant impact on risk insights.
Self-Assessment	SWC-HVAC	SY-A1	Self-Assessment	The dependency of HPCS-P-2 on pump house cooling was removed.	PRA Maintenance	This change corrected a model error. This change has no significant impact on risk insights.

Table RAI01-3 Model Change Performed for PRA Revision 7.2.1

Source	F&O	SR	Level	Model Change	Classification	Rationale
Self-Assessment	SA-16-QU-A1		Self-Assessment	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.	PRA Maintenance	The changes associated with this model release had no impact on the CDF or LERF (only on non-LERF Level 2 release terms) modeling or quantitative solutions.

PRA RAI 2 – Internal Events F&Os

A summary of the F&Os from the 2009 peer review were provided in Table A-1 of Enclosure 3 to the LAR. Address the following:

a. F&O 1-42 related to SR HR-I2 found that Human Reliability Analysis (HRA) Dependency Analysis has not been conducted for the Level 2 model. In resolution to this F&O, the licensee stated that the Level 2 HRA dependency analysis has been performed and documented. *Confirm that the Level 2 dependent Human Error Probabilities (HEPs) meet the HR supporting requirements in the PRA Standard ASME/ANS RA-Sa-2009 and justify why no focused-scope peer review is required.*
b. F&O 2-17 related to SY-A14 identified inadequate consideration of system operational history in the PRA system models. In resolution to this F&O the licensee stated that operational history has been collected and will be added to the PRA documentation. *Confirm that the system operational history has been reflected in the PRA models, or alternatively, justify why exclusion of operational history has no impact on the application.*

Response

a. This change corrected a PRA model omission and employs no new methodologies. To resolve this peer review finding for PRA update 7.1, a Level II HEP dependency analysis was performed and documented in Section 5 and Appendix D of the HRA Notebook, PSA-2-HR-0001. The HEP dependency analysis utilizes the same methodology to develop joint HEPs, NUREG/CR-1278, utilized for the Rev. 7.0 PRA, which was peer reviewed. As the level II HEP dependency analysis, utilized the same methodologies in the level I HEP dependency analysis, the level II HEP dependency analysis meets the HR supporting requirements in the PRA Standard ASME/ANS RA-Sa-2009 and no focused-scope peer review is required or recommended.
b. Actual system operational history, including industry significant operating experience and plant-specific operational history was collected and has been added to the system notebooks as part of the 7.1 PRA update. No PRA model changes were necessary to resolve F&O 2-17.

PRA RAI 3 – Seismic

In Section 3.2.4.2, "Scope of the PRA," of the SE for EPRI TR-1009325, Revision 2, the NRC staff stated that:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: "Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals." This section also states that: "If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this document [(i.e., EPRI Report No. 1009325, Revision 2)], the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed." This assessment can be taken from

existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

In Section 5.4.2 of Enclosure 4 to the LAR, the licensee performed an analysis of the external events contributions. The licensee stated that the seismic CDF and LERF risk estimates were taken from the Individual Plant Examination for External Events (IPEEE), with periodic updates to reflect the as-built and as-operated plant. This estimate does not appear to take into account the re-evaluated seismic hazard for Columbia, obtained in response to the Near-Term Task Force (NTTF) recommendation 2.1. *Justify, preferably quantitatively, why the seismic estimate provided in the LAR is a bounding estimate of seismic risk given the re-evaluated seismic hazard, or, alternatively, provide with justification, a conservative or bounding estimate of seismic risk that takes into account the new re-evaluated seismic hazard, and assess its impact on the application.*

<u>Response</u>

Columbia is upgrading its seismic PRA (SPRA) in response to Fukushima Near Term Task Force Recommendation 2.1. The SPRA upgrade is underway and has performed the tasks below. The tasks that are not complete are the detailed fragility development and the detailed human reliability assessment.

1. Incorporation of the updated hazard curve

   The re-evaluated seismic hazard has been developed in response to NTTF recommendation 2.1 and incorporated into the SPRA. This information is complete with respect to the SPRA development; therefore, the current state of knowledge will not change between now and the time the finalized SPRA is released.

2. Seismic equipment list

   The seismic equipment list has been developed and added to the plant response model. This task is complete pending final review and acceptance.

3. SPRA walkdown

   Seismic PRA walkdowns have been performed. This task is complete pending final review and acceptance.

4. Draft representative fragilities

   The SPRA model uses draft representative fragilities for structures, components and contact chatter. The SPRA representative fragilities are developed based on the new Columbia seismic hazard, available plant specific information, and plant walk downs. These representative fragilities have a conservative bias. Detailed fragility analyses still need to be performed. Refinements to the draft representative fragilities will produce higher seismic capacities. Therefore, the seismic CDF and LERF are not expected to increase beyond the magnitudes listed in this RAI response as a result of further refinements to seismic fragilities.

5. Contact chatter assessment

   Contact chatter assessment has been performed and added to the plant response model. This task is complete pending final review and acceptance.

6. Draft human reliability assessment

   The seismic HRA uses conservative screening human error probabilities (HEPs) for all post-initiator human failure events (HFEs) modeled in the SPRA, including the HFEs carried over from the internal events PRA. A post-initiator HEP dependency analysis has been performed to assess the degree of dependency between the HFEs and these dependencies are addressed in the SPRA quantification. Detailed HEP analyses still need to be performed. The HEPs for HFEs included in the SPRA are expected to decrease when the HRA guidance is applied (EPRI 3002008093). The seismic CDF and LERF are not expected to increase as a result of further refinements to the HEPs.

7. Seismic –induced fire and flooding

   The SPRA includes modeling of seismic-induced fire and seismic-induced flooding.

8. Plant response model

   The plant response model is based upon a robust, peer reviewed internal events PRA model. The modeling changes necessary to introduce the seismic hazard, fragility values, contact chatter, and the human reliability assessment have been integrated into the seismic plant response model. Correlated failures of like components during the earthquake are modeled. Therefore, the seismic CDF and LERF are not expected to increase beyond the magnitudes listed in this RAI response as a result of further refinements to the seismic plant response model.

The quantitative estimates provided in response to this RAI are based upon the current working SPRA model being produced as a result of that project. The seismic CDF and LERF are not expected to increase beyond the magnitudes listed in this RAI response as a result of further refinements to the SPRA model due to the conservatism in the current model.

The quantified CDF and LERF has been updated for revision 2 of this document to reflect work that has been completed on the SPRA since revision 0. The quantified CDF for the interim SPRA model is 1.88E-5/yr. This value is used to calculate an estimate for the LERF increase associated with the integrated leak rate test (ILRT) frequency extension. Equation RAI3-1 below (Equation 5-3 in the risk assessment for the regulatory submittal) is used as the basis for the calculation of the LERF increase.

$$F_{Class\ 3b} = (CDF_{total} - CDF_{AONL}) * PROB_{Class\ 3b} \qquad \text{(Equation RAI3-1)}$$

Where,

$F_{Class\ 3b}$ is the Class 3b release frequency.

CDF_{total} is the total seismic hazard CDF.

CDF_{AONL} is the portion of the CDF that is “always or never LERF”.

$PROB_{Class\ 3b}$ is the Class 3b failure probability.

Based on the Jeffrey’s Non-Informative Prior, the Class 3b failure probability is equal to 0.0023 for ILRT testing frequency of 3 in 10 years.

To simplify the evaluation and to ensure that there is a conservative bias in the treatment of the seismic hazard, CDF_{AONL} is set to zero and Equation RAI3-1 simplifies to:

$$F_{Class\ 3b} = CDF_{total} * PROB_{Class\ 3b}. \qquad \text{(Equation RAI3-2)}$$

Numerically,

$$F_{Class\ 3b} = 1.88E\text{-}5/yr * 0.0023 = 4.32E\text{-}8/yr.$$

Per the methodology in the risk evaluation for the regulatory submittal, the Class 3b failure probability for ILRT testing frequency of one in 15 years increases by a factor of 5.0. The Class 3b release frequency for the ILRT testing interval of once per fifteen years is 2.16E-7/yr. The increase in LERF from the base interval frequency to the once in 15 year frequency, therefore, is

2.16E-7/yr – 4.32E-8/yr = 1.73E-7/yr.

The baseline seismic LERF is quantified as 8.41E-6/yr.

The combined results for internal events, fire, and seismic hazards are summarized in Table RAI03-1. The 1E-6/yr ΔLERF acceptance threshold (based upon the baseline LERF value being between 1.0E-6/yr to 1.0E-5/yr) is met with large margins to that threshold.

Table RAI03-1: External Events LERF Summary

Hazard	LERF (yr^{-1})	ΔLERF (yr^{-1}) (3)
Internal Events	3.28E-07	3.07E-08
Fire	2.69E-07	6.79E-08
Seismic	8.41E-06	1.73E-07
Total	9.01E-06	2.72E-07
Threshold	1.00E-05 (1)	1.00E-06 (2)

(1) Threshold based upon the RG 1.174 (Reference 12) maximum LERF for Region II as illustrated in Figure 5-1 of the risk evaluation for the submittal.
(2) Threshold based upon the RG 1.174 (Reference 12) maximum ΔLERF for Region II as illustrated in Figure 5-1 of the risk evaluation for the submittal.

(3) Columbia internal fire and seismic model results are presented to provide qualitative insights into the risk associated with these hazards, as well as to provide an order of magnitude estimate for the increase in LERF attributable to these hazards.

PRA RAI 4

Section 5.4.2 of Enclosure 3 to the LAR states that external hazards other than fire and seismic (e.g., high winds and tornadoes, external floods, transportation accidents, and nearby facility accidents) were not considered because of their negligible contribution to overall plant risk. This conclusion was reached based on the IPEEE.

a. Since the IPEEE studies were performed in 1994 and have not been updated, *discuss, in the context of the current plant and its environs, the applicability of the IPEEE conclusions for the current LAR*.
b. In light of recent external flooding re-evaluation performed in response to the NTTF recommendations, *provide technical justification for why the risk from external flooding is negligible, or provide, with justification, a conservative or bounding estimate of the impact of external flooding risk for the current application.*

Response

a. A re-examination of the external events evaluation in the internal plant examination of external events (IPEEE) was performed in early 2017 and is documented in Reference 11. This examination assessed the applicability of the information presented in Section 5 of the IPEEE (which addresses external events excluding seismic and internal fire hazard) to the current as-built, as-operated plant. Updated information (where applicable) was presented in the analysis for the hazards discussed in Section 5 of the IPEEE. This evaluation looked at the following areas to identify hazards whose IPEEE disposition might need to be updated:

   1) Results of new research on the topics in question.
   2) Data revisions.
   3) New measures for risk management.
   4) Revised procedures for addressing external hazards.

   The re-examination concluded that the risk associated with each of the external hazards was qualitatively similar or lower in magnitude to that portrayed in the IPEEE. Based upon this evaluation, the conclusion for the ILRT that these hazards pose a negligible contribution to risk remains appropriate.

b. The Flood Hazard Reevaluation Report reevaluated all appropriate external flooding sources at Columbia Generating Station, including the effects of local intense precipitation (LIP) on the site, probable maximum flood (PMF) on streams and rivers, storm surges, seiches, tsunamis, and dam failures (as applicable).

   The reevaluated maximum water surface elevations validate the current flood mitigation strategy of the current license basis, which states that the site can be maintained in a safe condition for water levels up to 441 ft mean sea level (MSL). The reevaluated LIP maximum water surface elevations at Columbia vary between 435.14 ft and 443.27 ft. The calculated maximum water depths vary between 0.03 ft and 0.79 ft. Although the water surface elevations exceed 441 ft, MSL, the results indicate no flooding of site safety-related SSCs.

The addition of a site procedure, ABN-FLOODING, several years after the issuance of the IPEEE and the continued refinement of that procedure have enhanced the site's ability to mitigate this hazard.

Based upon this evaluation, the conclusion for the ILRT that the external flooding hazard poses a negligible contribution to risk remains appropriate.