NEXTera® ENERGY SEABROOK

May 31, 2016
10 CFR 50.90

SBK-L-16082
Docket No. 50-443

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

Seabrook Station

Supplement to License Amendment Request 16-01
Request to Extend Containment Leakage Test Frequency

References:

1. NextEra Energy Seabrook, LLC letter SBK-L-16029, "License Amendment Request 16-01, Request to Extend Containment Leakage Test Frequency," March 31, 2016 (ML16095A278)

2. NRC letter "Seabrook Station, Unit No. 1 - Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Request to Extend Containment Leakage Test Frequency (CAC No. MF7565)," May 19, 2016 (ML16139A181)

In Reference 1, NextEra Energy Seabrook, LLC (NextEra) submitted a license amendment request (LAR) to revise Technical Specification (TS) 6.15, Containment Leakage Rate Testing Program. The proposed amendment would revise the TS to require a containment leakage rate testing program that is in accordance with Nuclear Energy Institute (NEI) topic report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J. This proposed change will allow extension of the Type A test interval up to one test in 15 years and extension of the Type C test interval up to 75 months, based on acceptable performance history as defined in NEI 94-01, Revision 3-A.

In Reference 2, the NRC staff determined that supplemental information is necessary to enable the staff to make an independent assessment regarding the acceptability of the proposed amendment and exemption in terms of regulatory requirements and the protection of public health and safety and the environment. The Enclosure to this letter provides the necessary supplemental information.

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This supplement to LAR 16-01 does not alter the conclusion in Reference 1 that the changes do not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with the changes.

No new or revised commitments are included in this letter.

Should you have any questions regarding this letter, please contact Mr. Michael Ossing, Licensing Manager, at (603) 773-7512.

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

Executed on May 31, 2016.

Sincerely,

Eric McCartney
Site Vice President
NextEra Energy Seabrook, LLC

Enclosure

cc: NRC Region I Administrator
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Enclosure to SBK-L-16082

Supplement to License Amendment Request 16-01
Request to Extend Containment Leakage Test Frequency

Supplement to License Amendment Request 16-01
Request to Extend Containment Leakage Test Frequency

Background

By letter dated March 31, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16095A278), NextEra Energy Seabrook, LLC (NextEra) submitted a license amendment request (LAR) to revise Technical Specification (TS) 6.15, Containment Leakage Rate Testing Program. The proposed amendment would revise the TS to require a containment leakage rate testing program that is in accordance with Nuclear Energy Institute (NEI) topic report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

The NRC staff has reviewed the LAR and concluded that the following information is necessary to enable the staff to make an independent assessment regarding the acceptability of the proposed amendment and exemption in terms of regulatory requirements and the protection of public health and safety and the environment.

EMCB-1

Section 3.2.1.2 of your submittal states that Seabrook has committed as part of license renewal to maintain the exterior surface of the Containment Structure, from elevation -30 feet to +20 feet, in a dewatered state. NRR Office Instruction LIC-109, "Acceptance Review Procedures," (ADAMS Accession No. ML091810088) notes that a requested licensing action should not be accepted if it is contingent upon another review. The license renewal review is not complete and the adequacy of associated commitments has not been determined by the NRC staff.

Please provide additional explanation of why the dewatering is relevant to this LAR, and if relevant, how this action is being tracked under the current licensing basis.

NextEra Response

The statement in Section 3.2.1.2 of the LAR regarding maintaining the exterior surface of the containment structure in a dewatered state is irrelevant to this application and is deleted from the LAR. Although high relative humidity is a condition for promoting alkali-silica reaction (ASR), evaluations have shown that ASR has not affected the structural integrity of the containment. Similarly, the license renewal commitments discussed in Section 3.2.7 of the LAR (License Renewal Commitments) regarding future actions are not relevant to this application and are deleted.

EMCB-2

NEI 94-01, Revision 3-A, Section 9.2, notes that a visual examination shall be conducted of accessible surfaces of the containment for structural problems that may affect either the containment leakage integrity of the performance of the Type A test. Section 3.2.1.2 of your submittal provides a very high-level summary of the Alkali-Silica Reaction (ASR) concrete degradation indications on the containment structure; however, no discussion is provided about the impacts of the degradation on the containment structure.

Please discuss the ASR degradation impact on the containment, including justification for extending the Type A test interval considering the previous test was conducted before ASR indications were identified on the containment building.

NextEra Response

Alkali Silica Reaction (ASR) involves a chemical reaction between alkalis in the cement paste (Portland cement) and reactive forms of silica in the aggregates. This reaction is dependent on several factors including the amount and form of reactive material in the aggregate (e.g., reactive forms of quartz), the amount of alkali in the cement (more alkali - faster reaction), temperature (higher temperature - higher reaction rate), and moisture content. The reaction forms an expansive gel in the affected material. As the reaction progresses and the gels expand, micro-cracks are formed in the aggregate extending into the cement paste. The main observable effect of ASR affected structures is expansion and cracking due to gel formation. As expansions increase, visible cracks begin to form on the exposed surfaces.

The containment structure is comprised of two major structural elements, the biological shield portion (concrete portion) and the gas barrier (steel liner). The biological shield is the structural element of the containment and is constructed of reinforced concrete. The structure is made up of three basic structural elements: (1) the base mat, (2) an upright cylindrical shell, and (3) a hemispherical dome. The gas barrier is constructed of carbon steel plate, referred to as the liner, which acts as a leak-tight barrier. The containment integrated leak rate test (ILRT or Type A test) is performed to validate leak tightness of the steel liner.

During walkdown assessments of the containment as part of the Structural Monitoring Program, four isolated locations were identified on the cylindrical shell of containment where the concrete had pattern cracking, which is typical of ASR. Three locations were at the lower elevations where water ingress into the containment enclosure building (CEB) led to direct exposure of the outer surface of containment to water. The fourth location was in the mechanical penetration area, which did include evidence of water intrusion at the seismic isolation joint. Evidence of pattern cracking was noted at the four locations. In all locations, the pattern cracking was limited to a small area of the concrete surface (i.e., was highly localized). Much less than 1% of the containment structural surface displays signs of ASR map cracking.

The areas of identified ASR are in the cylindrical shell of containment, which is 4'-6" thick and has steel reinforcement in all three directions (vertically, horizontally and thru wall thickness). ASR affects reinforced concrete structures differently than unreinforced concrete structures because reinforcement limits the ASR expansion and therefore cracking. The tensile stress created by the ASR expansion is resisted by the internal reinforcement; i.e., reinforcement provides beneficial confining stresses limiting ASR-induced expansion. The beneficial effect of confinement with regard to minimizing any adverse impact on structural performance is most pronounced if the concrete structure has reinforcement in all three directions. The presence of reinforcing bars in all three directions in the cylindrical shell of containment, provides a well-detailed and proportioned reinforcing bar cage that is effective in restraining ASR related expansions. Inhibiting ASR expansion ensures that that the anchorage system for the steel liner on the inside surface of containment is not adversely affected.

During the 16th refueling outage, the containment liner was ultrasonically inspected to determine if wall loss has occurred due to corrosion on the opposite surface (concrete side). Inspection areas included areas that were local to the four locations identified with ASR on the concrete surface of containment. The examination areas did not exhibit any signs of corrosion on the opposite (concrete) side of the steel liner.

In summary, the presence of ASR identified in containment concrete has no impact on the leak-tightness of the containment steel liner based on:

- Containment's three directional steel reinforcement arrangements, which inhibits ASR expansion,
- The very limited localized areas of ASR detected on the containment surface, and
- Previous UT inspections of the containment liner local to areas of ASR in which no anomalies or corrosion were identified.

APLA-1

The LAR specifies that the technical basis for the proposed change utilizes a risk impact evaluation yielding results within the limits set forth by EPRI Technical Report (TR) TR-1009325, Revision 2 (ADAMS Accession No. ML072970208). The NRC safety evaluation report for the EPRI TR (ADAMS Accession No. ML081140105), directs the licensee to submit documentation indicating that the technical adequacy of their Probabilistic Risk Assessment (PRA) is consistent with the requirements of RG 1.200 (ADAMS Accession No. ML090410014) relevant to the ILRT extension application. RG 1.200, Revision 2 states that to demonstrate the technical adequacy of the PRA used in an application is of sufficient quality a discussion of the resolution of pertinent PRA peer review findings and observations (F&Os) be included.

While Attachment 4, Appendix A, of your submittal provides a high-level summary of industry peer reviews and self-assessments, specific applicable Seabrook F&Os were not included. Thus, to demonstrate the technical adequacy of the Seabrook PRA against RG 1.200, Revision 2, submit a list of all F&Os cited as Findings from the latest full- and focused-scope peer reviews that were conducted for any hazard for which a PRA model exists and was peer reviewed including any self-assessments which were not closed by a subsequent peer review and for which the PRA failed to meet Capability Category I (CC I) of the applicable PRA standard supporting requirements, including any cited as Not Met. For each F&O include details regarding its disposition, and an explanation of why not meeting the corresponding CC I requirement has no impact on the application.

NextEra Response

A summary of the Seabrook Station PRA peer review findings and associated resolutions is contained in Attachment 1, Summary of Findings from PRA Peer Reviews. The findings in Attachment 1 are associated with peer reviews and self-assessments performed for ASME/ANS PRA Standard Part 2 (Internal Events) and Part 3 (Internal Flood Events). All findings have been resolved and closed and there are no open findings for which the PRA did not meet the ASME/ANS PRA Standard Capability Category (CC I) supporting requirements for internal events and internal flood events. Therefore, the PRA internal events and internal flood events models are fully capable to support the ILRT application. No other peer review or self-assessments have been performed for other hazard models including Part 4 (Internal Fire Events) or Part 5 (Seismic Events). As provided in Section 6.3 of Appendix A to Attachment 4 in the LAR, internal fire and seismic risks are qualitatively and quantitatively (via sensitivity) assessed to further demonstrate that the proposed ILRT Type A test extension will have a minimal risk impact on these risks.

ATTACHMENT 1

Seabrook Station PRA Peer Review Findings

Summary of Findings from PRA Peer Reviews.

This table summarizes the Peer Review Findings. All of the findings have been addressed, as documented in the Resolution column, and are closed.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O IE-2 (IE-C10)	Closed	The frequencies of initiators L2CCA and L2CCB are under estimated due to the common cause model. The common cause term should include T=1 year (rather than 24 hours).	Changes were made to the CCF models in PCC and SWS initiators to use 1 year as the mission time.
F&O IE-6 (IE-C14)	Closed	The existing analyses for ISLOCA should be reviewed for consistency with a methodology for identification and quantification of ISLOCA pathways such as that provided in NUREG/CR-5744, and updated if appropriate.	Reviewed NUREG/CR-5744 for ISLOCA methodology and revised the ISLOCA assessment.
F&O AS-6 (DA-C15)	Closed	The emergency diesel generator recovery failure probability seems optimistic for the medium RCP seal LOCA event. The data for recovery of an EDG is based on data taken from LERs based on EDG failures. This data is used to develop a recovery curve. However, this recovery is applied in conditions very different than the conditions in the LER - common cause failure of both EDGs resulting in SBO conditions. The EDG recovery is based on generic data composed of EDG single failures during normal operation. This data needs to be reviewed to ensure applicability to CCF events, particularly events during more adverse SBO conditions (i.e., where stress, crew availability, and so forth, are more limiting). In addition, plant-specific evidence should be used to support this recovery probability.	Evaluated Seabrook Station EDG failure data. Of the four failures, two could easily be recovered within 4 hours. The other two failures were considered long-term failures. Based on SB data, a non-recovery probability of 0.5 was used for DG recovery.
F&O DA-4 (DA-D5)	Closed	The values for BETA2, GAMMA2, and DELTA2 are not derived as recommended in NUREG/CR-5485 as stated in the text. That document (p.76) recommends that "the values of α_2, α_3, and α_4 in Table 5-11 be reduced by a factor of 2 when applied to frequency of failure during operation." The effect of reducing theses values (and adding the difference to *1) is to reduce only the Beta factor - the gamma factors and delta factors are unchanged since the factor of one-half factors out. Contrary to this guidance, the MGL factors corresponding to the alpha factors in Table 5-11 were calculated, then the Beta factors were reduced by a factor of 2. Note these values were used in the PCC system and initiating event analyses, resulting in some factors being under-estimated by a factor of 4. The discussion in 6.3.3 regarding variable BETA1 is in error - 5 CCFs and 100 independent failures provides a beta factor of 5/105 if staggered testing is used, not the .05 indicated. A lognormal distribution is not appropriate for the GAMMA1 and DELTA1 - they should be modeled using beta distributions.	The values for GAMMA2 and DELTA2 were recalculated using the correct equations. Also beta distributions were developed for these generic distributions. With regard to the comment that BETA1 should be 5/105 rather than 0.05, these are essentially the same number.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O DA-6 (DA-C15)	Closed	Examine dependencies of HEPs embedded within recovery models with other human actions included in the plant model. Examine most recent component failure data to ensure recoverable failure fraction remains valid. Develop appropriate procedures for identifying and evaluating dependencies.	Operator dependencies were examined, resulting in changes made to the logic rules and HEP quantification.
F&O QU-3 (QU-B2)	Closed	A discussion of the limitations of using the saved sequences as a PRA model of the plant was not located. Although a very low cutoff is used to generate saved sequences, it is important that all analysts understand where limitations may exist so that they can be evaluated for specific applications	This issue of truncation has been addressed in the PRA documentation along with general guidance for setting the truncation level. Practically, this issue must be evaluated for each analysis. It is not possible to give general guidance that addresses every application.
F&O QU-9 (QU-E4)	Closed	At present no parametric uncertainty analysis exists based on the current plant model. While such studies were performed for earlier versions of the SSPSA, the results have significantly changed (internals are far less dominant) and the uncertainty distribution may no longer be valid. At present there is no formal analysis which addresses plant specific uncertainty or sensitivity issues. For example, cases where thermal hydraulic analyses predict only small margins for success in terms of the number of trains required, or the time available for operator actions, are prime candidates. Other examples might be cases where unique success criteria or modeling have been applied such as for feed and bleed and for RWST make up following LOCA. Perform a set of sensitivity runs and a qualitative or quantitative uncertainty analysis for the model. Risk achievement analyses may be used to focus the search for potentially significant cases.	Performed an uncertainty analysis to address this F&O. Ensured that all split fractions have an uncertainty distribution associated with them and quantified all event tree top events with Monte Carlo. Also quantified all system initiating events with Monte Carlo. Quantified uncertainty for dominant sequences for CDF and LERF.
F&O MU-2 (SY-A2)	Closed	During a review of plant design changes incorporated into the 1999 PRA models, it appeared that Design Change Request (DCR) 89-061 had not been incorporated into the service water fault tree. This DCR deleted the cooling tower fan auto-start feature. Therefore, a human error basic event was to be added to the service water fault tree. The service water fault tree did not appear to have been modified. Also, the PRA documentation still includes the cooling tower fans being actuated by a TA signal. It is believed that this is an isolated occurrence. However, the host utility should check for any others. Incorporate this DCR into the system fault tree / notebook.	A review of the PRA documentation (Service Water Notebook) indicated that this DCR had indeed been incorporated in the PRA model. In fact, the system notebook describes the modeling of the cooling tower and indicates that the operator must manually initiate CT operation and provides a justification for why this action is not modeled. The Service Water notebook was updated to ensure completeness. Also, a review of DCRs for the 1999 update was performed to ensure that all DCRs that impact the PRA model were addressed.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O AS-A9-1 (AS-A9)	Closed	The ASME Category II capability for this SR requires the use of realistic, applicable T/H analyses for accident sequence parameters. Category III requires use of realistic, plant specific T/H analyses. Although most of the SSPSS parameters have supporting calculations that are plant specific, it appears that some would benefit from more realistic analyses. In at least one case (i.e., CST depletion) more realistic analyses may impact sequence development (and are dependent on whether the EFW pump or SUFP is running). Expectation for future applications is more extensive use of realistic codes (e.g., MAAP), as applicable.	The SSPSS-2005 update effort used MAAP to provide substantial additional plant-specific, realistic support. In some cases such as the CST example noted above, hand calculations were considered to be appropriate and were reviewed to assure adequate realism. The actions below were taken to address realistic/plant-specific success criteria: Listed all current Level 1 success criteria, including impact of power uprate, RCPs, IA, etc. Identified current basis for success criteria. Ran series of MAAP runs where needed to provide basis.
F&O HR-E3-1 (HR-E3)	Closed	While simulator exercises were observed, there is no evidence of specific talk-throughs with Operations/Training. Interaction with Operations and/or Training is important regarding the assumptions used in the HRA, especially response times and performance shaping factors (PSFs), to confirm that the interpretation and implementation of the procedures are consistent with plant training and expected responses.	Walkthroughs / talk-through with Operations and/or Training were used to confirm modeling of operator actions and accident sequences. -.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O HR-G4-1 (HR-G4)	Closed	In general, the time available to complete actions is based on either generic T/H analyses for similar Westinghouse 4-loop plants or plant-specific analyses. Several issues were identified that may point to the need for establishing a more thorough and realistic basis. For example: The write-up for the operator action ODEP1 for SBO events states that 8.8 hours are available to perform this action, which is based on 9.8 hours to core damage from WCAP-16141, less one hour to restore equipment. However, WCAP-16141 states that without depressurization, core damage can occur as early as 2.7 hours. Therefore, the time available to perform this action should not exceed the time to core damage without credit for the action. It should be noted that WCAP-16141 does not specifically mention when depressurization must begin, but it seems to be assumed that depressurization will typically begin within 30 – 45 minutes. Since this action has a low F-V and RAW importance, SR HR-G4 is judged to be satisfied. WCAP-16141, which is used as a basis, assumes that the turbine-driven AFW pump supplies 1145 gpm, which seems to exceed the capacity of the Seabrook Station TD AFWP. The basis of the time available for operator action ODEP3 does not appear to be realistic. SSPSS-2004 credits post-LOCA cooldown and depressurization for MLOCA with high head injection (HHI) success. Operator Action timing (3.8 hours) is based on a small LOCA, not MLOCA. The success criteria indicates that only 42.8 minutes are available before reaching low-low level for MLOCA. While it is true that MLOCAs at the high end of the spectrum should not require this action and MLOCAs on the low end of the spectrum behave more like a small LOCA, the majority of MLOCAs will be in between.	Revised the HRA Calculator quantification using time windows from Seabrook Station-specific MAAP runs.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O HR-G4-1 (HR-G4) continued		Using the average timing between the high end (42.8 minutes) and low end (3.8 hours) would not leave enough time to successfully establish low pressure recirculation prior to reaching the RWST low-low level switchover set point. The time assumed to be available for feed and bleed using the Safety Injection (SI) pumps, which is based on the time until SG dryout, may not be realistic. It would seem that establishing feed and bleed with the charging pumps would have different timing than establishing feed and bleed with the SI pumps due to the lower shutoff head of the SI pumps. In particular, while waiting until SG dryout could allow successful feed and bleed cooling using the charging pumps, it isn't clear that waiting until SG dryout would allow successful feed and bleed cooling using the SI pumps. The time available for operator action HH.ORSGC2.FL is 2.3 hours, which is based on time to core damage. However, restoring secondary cooling at the time of core damage will not prevent core damage. In order to prevent core damage, secondary cooling must be completed earlier (e.g., core uncovery). With respect to the items identified: Re-evaluate the time available to perform RCS cooldown and depressurization following an SBO. Also evaluate the applicability of WCAP-14161 assumptions regarding flow from the turbine-driven AFW pump. Re-evaluate the time available used to quantify operator actions for depressurization and feed and bleed by performing sequence-specific MAAP (or other) thermal-hydraulic runs. In the case of operator action to perform depressurization for MLOCA sequences, T/H runs may need to be performed for an "average" MLOCA break size. Use MAAP or some other calculations to determine the latest time at which secondary cooling can be restored and still prevent core damage. More generally, complete the ongoing effort to establish appropriate timeframes using realistic codes (e.g. MAAP).	F&O HR-G4-1 (HR-G4) – continued. Refer above for resolution.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O HR-G7-1 (HR-G7)	Closed	Dependency between multiple human actions was considered, and the process for quantifying dependencies is described in SSPSS-2002. This appears to be a good approach. However, there is no guidance as to how to identify sequences with multiple operator actions for inclusion in the dependency analysis. Also, while the matrix showing dependency between two operator actions is good, it does not include new actions since the 2002 update. The review discovered at least two examples where dependencies appear to be inadequately addressed: (1) The dependency between operator actions ORSGC and OFB does not appear to be modeled, other than time consumed associated with responding to feed and bleed criteria. There is also some dependency in diagnosing the loss of secondary heat sink for these two actions. (2) The procedural guidance in Functional Restoration Procedure FR-H.1 for aligning fire water is contained in the RNO column of Step 14, which is predicated on not being able to open the PORVs. However, if the PORVs are opened too late, the procedure will not direct the operator to establish fire water to the SGs. This dependency is not modeled. Although significant progress has been made in this area since the 1999 peer review, it appears that there remains a need to develop an overall process for identifying multiple operator actions that need to be addressed in the dependency analysis.	The following actions were taken during the PRA update: 1. Identified all dynamic actions embedded in hardware top events. 2. Created new Operator Action top events, separate from hardware where appropriate. 3. For PCCW, redefined System split fractions to be conditional on Operator Action OPCC and added house events. 4. Added new top events to event trees 5. Modified logic rules to account for operator action dependency to system.
F&O 4-6 (IFQU-A7)	Closed	Appendix 12.1H describes assumptions and uncertainties. However, the level of detail with respect to sequence reviews and results, including integration are not judged to be of sufficient detail (e.g., QU-A3, D1, D5, D6, D7, E3). Integration into internal event QU notebook will resolve some of this.	IF analysis sequence review and results review were performed collectively, in an integrated fashion with all Level 1 results. No issues were identified with IF sequences. IF dominating sequences are described in Section 12 of the PRA report (Tier 2) and are reasonable and as expected for the model inputs. It is noted that the initial review of sequences identified the potential need to reduce flooding risk in the Control Building. This led to the proposed modification to install the flow reducing orifice in the FP piping upstream of the CB.
F&O 4-7 (IFQU-B1)	Closed	Self-Assessment points out areas of improvement in reviewing results and identifying significant contributors to CDF (and LERF), such as initiating events, accident sequences, and basic events (equipment unavailability and human failure events), shall be identified. In addition, the results shall be traceable to the inputs and assumptions made in the PRA.	IF analysis sequence review and results review were performed collectively, in an integrated fashion with all Level 1 results. No issues were identified with IF sequences. IF dominating sequences are identified and described in Section 12 of the PRA report (Tier 2). IF Sequences are reasonable and as expected for the model inputs. Tier 3 IF documentation of results includes initiating event, basic event and human failure event contributions to CDF. Internal flooding has been shown not to have a significant effect on Level 2 results. The IF results appear reasonable and as expected based on the model inputs and assumptions.
F&O 4-9 (IFQU-B3)	Closed	The completeness of assumptions and sources of uncertainty in the pipe failure data (e.g., error factor, applicability of data), failure probability of doors, generic data and modeling choices needs to be reviewed against other industry studies.	A check of the data and assumptions used in the internal flooding study was performed for reasonableness and for identification of additional uncertainties. PRA Appendix 12.1H (Tier 3), Uncertainties, was revised to clarify/ensure areas of uncertainty and important assumptions are adequately captured and characterized.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O 5-2 (IFSO-B3)	Closed	Appendix 12.1H acknowledges uncertainty in break flow rate. Need to expand uncertainty review to discuss other source related uncertainties such as maintenance-induced events and potential, if any, source pressure or temperature impacts. Also, discuss potential for breaks or human induced events greater than assigned (i.e., catastrophic CW expansion joint failure could far exceed 56,000 gpm). Potential for larger floods can represent key insights. Specifically, CW flood rates greater than 56,000 gpm could represent a more significant threat to the Essential Switchgear rooms due to the configuration at Seabrook.	As mentioned in the disposition for F&O 4-9, PRA Appendix 12.1H (Tier 3), Uncertainties, was revised to clarify/ensure areas of uncertainty and important assumptions are adequately captured and characterized. In addition, a sensitivity evaluation was performed to conservatively determine the risk significance of a postulated maximum CW flood event. The maximum CW break flow was estimated at approximately 300,000 gpm. A door failure evaluation was performed to estimate the capacity of the various door configurations at Seabrook. Doors C102, C101 and C100 provide an interface between the TB and ESWGR-A. The door evaluation indicates that the capacity of these types of doors loaded against the jam/frame is in excess of any credible flood height in the TB. In addition, other doors in the Turbine Building are expected to fail at considerably less water height - approximately 10 feet (or less) and there is an unlatched door on the east side near condensate polishing that opens out. The benefit of this door was not credited. Once a flood height of ~10 ft or less is achieved, failure of these other doors (which includes the rollup doors, glass sliding door, misc. double doors) is expected to vent the flood water to outdoors and result in a steady-state water level in the TB of ~4 feet. It is noted that this TB flooding scenario is likely to cause a loss of offsite power or fail non-essential electrical buses, resulting in a trip of the flooding source – the CW pumps long before there is propagation impact in the essential switchgear rooms. Based on the above, a conservative flood scenario was developed as sensitivity case F0TCWS. Based on this sensitivity case, the CDF from a postulated maximum CW break event in the TB is approximately 1E-09/yr. This scenario is screened from further detailed evaluation using criterion QN4a - Specific flood source in a flood area with CDF < ~1e-9 per yr based on flood-initiated accident sequences from a specific flood source in the flood area. This assessment is conservative. Realistic modeling would eliminate conservatisms and further reduce the impacts.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O 5-3 (IFSN-A2)	Closed	The assessment indicates that there are some "rugged" doors capable of withstanding a water-height of 6-7 feet. These were walked-down for the peer review and they are indeed rugged in appearance. However, there is limited basis for door capacity other than "Industry Sources" which include a PWR OG e-mail. The EPRI Flood Guideline says the following: If there are doors within the boundaries of the area then the following guidance can be applied: Water tight doors should be considered as failing only through human actions. If the door is alarmed its failure probability can be considered to be zero. If the door is not alarmed then assume the normal egress failure condition of a door opening out of the flood area if the water tight door opens out of the area. If the water tight door opens into the area then consider the failure probability to be zero. Normal egress and fire doors should be considered failed after 3 foot of flood level if the door opens into the area. Normal egress and fire doors should be considered failed after 1 foot of flood level if the door opens out of the flood area. The 1 and 3 foot EPRI Guideline should be used unless a higher value can be justified. While the doors are clearly rugged, some more detailed justification should be presented.	A structural evaluation of typical doors at Seabrook Station was performed and documented in a calculation, "Structural Evaluation of Door Capacity Under Flooding Loading Conditions". The evaluation was performed for 3 "typical"-type doors including: (1) rugged security door, (2) industrial 3 hour rated fire door, and (3) double-wide industrial door with and without a center locking pin. The evaluation addressed the difference in potential failure when each type of door is loaded against its frame/jamb (stronger door configuration) verses being loaded against its latch and hinges (weaker door configuration). It is noted that the door frames at Seabrook are embedded into the adjacent concrete and are not supported by installed anchor bolts. This represents a much stronger configuration than a conventionally installed frame with anchor bolts. Door capacity/failure insights from the structural evaluation are included in PRA Appendix 12.1A (Tier 3), Methodology. Door failures and the resultant propagation are assessed on an individual door/scenario basis. If the scenario's flood water height does not exceed the door's capacity, the door is not expected to fail, is assumed to remain intact with only gap leakage contributing to propagation. On the contrary, if the scenario's flood water height exceeds the door capacity, door failure is assumed and the resulting propagation is via the failed (open) door. No credit is given for failure of a barrier to limit the flood consequence without some assessment of the door failure potential.
F&O 5-5 (IFSN-A9)	Closed	Flood calculations are available in spreadsheets linked to the master summary for each area. However, these are dynamic and direct identification of key parameters can be difficult. For example, a key time for TB flood response is 11 minutes. In the spreadsheet related to this parameter (Turbine Buidling.xls) the flow rate was set to 15000 gpm versus the defined source value of 56,000 gpm. Therefore, the 11 minute time estimate was not depicted by the spreadsheet. In addition, some spreadsheet worksheets and tables are not used in the analysis and have confusing negative signs. This is all an unnecessary distraction in an already complicated analysis. Tractability between the flood area definition and HRA is required to justify the analysis.	A snapshot of spreadsheet calculations has not been provided. However, all spreadsheets have been cleaned up and all superfluous spreadsheets and information have been eliminated. Spreadsheets are also referenced in the text and master tables to improve retrieveability.
F&O 5-9 (IFPP-A2)	Closed	Specific rooms are discussed in the detailed analysis but there appears to be no specific definition at the room (or combined room level). There is no definition of what specifically constitutes a flood area other than the Building Level definitions presented in Appendix 12.1B Summary Table. Room level definition is left to be inferred based on discussions within the detailed analysis.	A new table is developed to define the Seabrook flood areas within the various buildings. The new table "Flood Area Definition" is contained in PRA Appendix 12.1B (Tier 3) and is discussed in Section 3.0 of PRA Appendix 12.1A, Internal Flooding Methodology (Tier 3). The flood areas are defined using the fire areas/rooms identified on the Seabrook Pre-Fire Strategy drawings.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O 5-12 (IFSO-A1) (IFSN-A8)	Closed	No review appears available relative to backflow through drains. Another plant recently had an NRC identified issue where a radwaste pipe tunnel floor drain emptied into an RHR Room Sump. This represented an identified pathway from one building to another. Consider a more detailed review of floor drain connections and interfaces. The self-assessment also questioned floor capacity for cases with significant water accumulation (i.e., rugged doors prevent propagation to other areas). More explicit review of the potential for floor drain backflow, including the potential for check valve failure, may yield some noteworthy insights. For example, there is currently no discussion of the potential for backflow from the RHR A sump to the RHR B sump.	Additional review of the floor drain systems in the major flooding areas of interest was performed including postulating possible floor drain backup, waste tank backup and qualitative assessment of check valve failure in sump pump discharge systems. There are no valves (AOVs/MOVs/check) in the floor drain system itself. Additional review is documented in PRA Appendix 12.1A "Methodology" (Tier 3), with pointers to other sections that provide more detailed discussion. No vulnerabilities were identified based on the review. In particular, check valves exist in the RHR sump pump discharge lines to prevent backflow from one RHR vault to the other.
F&O 5-13 (IFSO-A4) (IFSN-A12)	Closed	Limited evidence of review of potential for maintenance or operationally induced flooding events. Review of potential for maintenance induced floods is a specific requirement.	PRA Appendix 12.1A, Section 4, Flood Mechanisms and Maintenance-Induced Flood Events (Tier 3), documents the review of maintenance-induced flooding events. A review of the potential for maintenance-induced flooding events is performed for the major flood source systems. These systems included: CW, SW, FP and PCCW. The potential for maintenance-induced flood events from other sources, for example DW, PW, RW are judged to be less limiting because of their lower flooding flow rates. Development of maintenance-induced flooding events included performing a review of industry and plant-specific flood OE, performing a general survey of all Seabrook Work Orders (WO) performed on these systems between 1990 and 2009 to identify any potential flood related maintenance events, an assessment of water hammer potential, and a discussion with the respective system engineer. Based on the overall review performed, maintenance-induced actuation of fire protection deluge systems (inadvertent FP actuation) in the CSR, DG Fuel Tank Rooms and TB is specifically accounted for in the flooding risk assessment. Other maintenance activities including water hammer phenomena are judged not to have a significant potential to initiate flooding events; their likelihood to cause a flooding event is very low and judged to be adequately accounted for in the random flood initiating event frequencies.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O 5-14 (IFSO-A5)	Closed	Basis for flow rates is not specified (e.g., Page 31, Appendix 12.1F, says 24" SW discharge pipe is ~12,000gpm. No apparent basis for the 12,000gpm value is presented. The assignment of break flow rates is a key analysis input and specific, traceable basis for the values is required. Values reviewed seem reasonable but slightly more than just a documentation issue in that basis is not specified.	Additional flow calculations were performed to improve the basis for the selected scenario-specific break flow rates. Application of these flooding flow rates is described in Tier 3 PRA documentation Appendix 12.1A, Section 3.0, Scope, and summarized here. SW Maximum Break Flow Rate: Maximum SW pipe break flow rates were developed by specific calculations using the plant's SW flow model and ProtoFlow software. Specific SW calculations were performed for scenario development in the PAB, TB and yard. Maximum SW break flow rates at other locations (SWPH and CT) are based on inspection of the SW break calculations performed for PAB and discussion with the SW system design engineer. Other System Flood Scenario Maximum Break Flow Rate: The flood source maximum break flow rates used in this analysis for other systems (primarily for FP, CW, DM, PW and tank gravity drain) are based on either: (1) a specific break flow spreadsheet calculation based on actual/conservative system characteristics using flow equations/methodology in Crane Technical Paper 410 or, (2) the maximum break flow rate for the specific pipe size and pressure conditions as suggested in the Appendix C of the EPRI methodology. The maximum break flood scenarios developed in the master tables identify the scenario's maximum estimated break flow and its source reference (either specific spreadsheet calculation or EPRI). Flood Scenario Flow Rates Other Than Maximum: The flood source flow rates for scenarios other than maximum are based on the EPRI methodology for categorizing spray, large and major flood events depending on the flood source capacity. Spray-type events are assumed to be in the flow range of 0 to ~100gpm; large flood events are in the flow range of 100 to ~2000gpm and major flood events are in the range of 2000gpm to the maximum capacity. The flow rates used in the scenarios (other than the maximum flow rate) are also identified in the master tables with its EPRI source reference. It is noted that although a flow range is specified, the timing for each scenario (time to equipment damage, time for operator mitigative actions, etc) is conservatively based on the upper bound of each flow rate range.
F&O 5-16 (IFSO-A5)	Closed	Address item IFSO-A5(d) which requires consideration of pressure and temperature of the source of the rupture.	Tier 3 PRA documentation Appendix 12.1A, Table 3-1 provides a summary of the flood sources and the source parameters of temperature and pressure and overall volume capacity.
F&O 5-18 (IFSN-A14) (IFSN-A16)	Closed	Some additional, clarifying discussion would be beneficial within the master table. Consider the following example and address this and similar cases appropriately. Item 6 in the control Building master table addresses Initiator F1CFPS. The master table indicates that propagation to ESWGRB (correct to be ESWGRA) is screened based on highly reliable mitigation. A separate column provides a time window of 145 minutes. The reviewer infers from this that mitigation is highly reliable for this case because a long time is available for operator action (i.e., > 2 hours). However, this is not clearly stated. Clearly identifying not only the screening criteria but specific attributes that allow criteria to be met provides the most comprehensive and reviewable screening summary.	The timing basis for highly reliable operator actions was reviewed, and each is consistent with the screening criteria. The scenario screening description includes the timing basis for all operator actions including those credited as highly reliable actions.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O LE-C3-01 (SR LE-C3)	Closed	No credit for repair was taken in the analysis other than recovery of AC power. There was no review of the accident progression sequences for opportunities to credit equipment repair.	The total LERF contribution as determined in the current model of record (SB2011) is 9.2E-08/yr. The ASME standard defines the significant accident progression sequences as those sequences that sum to 95% LERF or with the individual sequence percentage of 1% LERF. The significant accident progression sequences are identified below, based on initiating events that contribute 95% (summed) and 1% (individual) to LERF include the following: **Initiator/ Description / LERF/yr / Contribution / Cumulative** SGTR/ SG Tube Rupture / 4.45E-08 / 48.3% / 43.8% LOC1VI / ISLOCA/ 1.80E-08 / 19.5% / 67.8% E25L / Seismic LOCA (2.5 g) / 7.29E-09 / 7.9% / 75.7% E18L / Seismic LOCA (2.5 g) / 4.36E-09 / 4.7% / 80.4% E18T / Seismic Transient (1.8g) / 3.83E-09 / 4.2% / 84.6% E14A / Seismic ATWS (1.4g) / 3.21E-09 / 3.5% / 88.1% E18A / Seismic ATWS (1.8g) / 2.83E-09 / 3.1% / 91.2% E25A / Seismic ATWS (2.5g) / 2.65E-09 / 2.9% / 94.1% E25T/ Seismic Transient (2.5g) / 1.80E-09 / 2.0% / 96.1% E10A / Seismic ATWS (1.0g) / 1.32E-09 / 1.4% / 97.5% Other <1E-09 / <1% The above LERF initiators and associated significant accident progression sequences are reviewed to determine if repair of equipment (beyond offsite power) could be credited during the accident progression or after containment failure to further reduce the LERF contribution.

F&O ID	Status	Peer Review Finding	Resolution
F&O LE-C3-01 (Continued)			SGTR Sequences: - SGTR sequences contribute 4.45E-08/yr to LERF. A typical warning time (time between core damage and start of the release) for SGTR scenarios is short (approximately 4 hours). SGTR initiator models steam generator tube rupture occurring randomly. SGTR with subsequent failures can result in core damage with containment bypass. The top ranking SGTR sequences involve a tube rupture with successful SG cooling and primary system makeup but with failure of the main steam safety valve(s) to reseat, failure of the operator to terminate SI, and failure of decay heat removal via long-term primary inventory makeup including credit for decay heat removal using feed and bleed cooling. Also modeled is failure of the operator to establish feed to the faulted SG after core damage but prior to a significant release (scrubbing). A possible SGTR repair action to further reduce LERF is repair of MS Safety Valve. Possible actions would be to either repair the safety valve or install some type of manual gagging device to get the valve reclosed. Re-closure would limit the safety valve discharge; this would reduce or terminate the release. However, actual repair of the valve is judged not practical. Also, given the location of the safety valve, equipment needed, limited timing, personnel habitability concerns, installation of a gagging device cannot reasonably be justified. Based on this SGTR sequence review, there are no practical repair candidates that can be reasonably credited to further reduce LERF from SGTR events. LOC1VI Sequences: - LOC1VI sequences contribute 1.8E-08/yr to LERF. The typical warning time (time between core damage and start of the release) for an ISLOVA scenario is very short (approximately 0.1 hours). LOC1VI models an interfacing system LOCA (ISLOCA) resulting in containment bypass sequences. The LOC1VI event occurs in the low pressure RHR system due to failure of dual normally closed discharge check valves in one of four injection lines. LOC1VI sequences involve failure of the RHR pipe/heat exchanger or failure of the RHR pump mechanical seal. Sequences that involve RHR pipe or heat exchanger failure are given minimal credit for mitigation because injection water is lost out of the break (located outside containment) and long term sump recirculation is not available. Sufficient time is not available to refill the RWST for long term injection. This break location is assumed to be high in the RHR Equipment Vault, thus the release path is outside containment without scrubbing through a pool of water in the equipment vault.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O LE-C3-01 (Continued)			Operator actions to depressurize the RCS (using SG depressurization and SG makeup with EFW) to limit the rate of RCS inventory loss, and to refill the RWST for long term RCS injection capability are modeled in the RHR pump seal failure event. Based on this ISLOCA sequence review, there are no practical repair candidates that can be reasonably credited to further reduce LERF from LOC1VI events. With the break location at the RHR pump mechanical seal, the release path is also outside of containment but with the benefit of scrubbing credited through a pool of water in the equipment vault. The RHR pumps are assumed to not survive the harsh environment given their location relative to the ISLOCA event. However, the Charging pumps are credited (located in a different building) along with the SI pumps (minimal credit due to their location upper elevation of the RHR vaults) for injection. Seismic Initiating Event Sequences: - Seismic event LERF sequences contribute a total of 2.73E-08/yr to LERF. The typical warning time (time between core damage and start of the release) for seismic-induced large LOCA is relatively long (approximately 20 hours). The seismic initiators, E25L through E10A, represent extreme seismic events (1.0g and larger). These events are assumed to impair/delay effective evacuation and thus contribute to LERF. Without the assumption of impaired evacuation, these seismic event sequences would bin to "late" release bins. Sequences associated with these seismic events include large LOCA seismic initiators with failure of RCS inventory injection (accumulators and low-head injection), small LOCA seismic initiator with failure of RCS inventory makeup (high-head injection), ATWS seismic initiator during unfavorable exposure time with resultant overpressure condition that ruptures the RPV, and transient seismic events with failure of decay heat removal equipment. For all of these extreme seismic events, the dominant failure mode of plant equipment is seismic-induced failure. Based on this review and given the severity of these seismic events, there are no practical repair candidates that can be reasonably credited to further reduce LERF from these large seismically-induced events.

F&O ID	Status	Peer Review Finding	Resolution
F&O LE-C5-01 (SR LE-C5)	Closed	The only relevant system looked at for this SR was AFW (for SGTR scrubbing). No basis for the AFW success criteria, as documented in the SSPSS, Section10.4.3.3, is given. It is assumed that these success criteria are based on design calculations, not realistic analyses.	All "small-early" and "large-late" release bins were reviewed to identify if success of a particular system (for example EFW) is credited in the Level 2 analysis, which then allows binning of the sequences as "small" and instead of "large" or "late" instead of "early". Such system credit can be thought of as being significant to defining LERF/non-LERF sequences. Of all the release category bins, only bin SE1 - "Small Early Containment Bypass - SGTR with Scrubbed Release" credits a specific system (EFW) for release reduction (scrubbing). SE1 sequences are SGTR sequences that credit use of the EFW system to maintain/re-establish SG water level in the faulted SG thus scrubbing (reducing) the release. SE-1 sequences are summarized below along with a realistic EFW "Level 2" success criterion. It is noted that SE2 - Interfacing LOCA also credits scrubbing for release reduction. However, SE2 scenarios include success of Level 1 injection of HPI until the RWST inventory is depleted. Break flow/HPI causes flooding/submergence of the break (RHR pump seal) located in the lower elevation of the RHR vault. There are no specific systems credited in Level 2 for achieving the flood conditions needed for scrubbing/release reduction. There are no other systems specifically credited in the remaining "small" or "late" release bins that require established Level 2 success criteria. Sequences in these small and late bins are there because of the Level 1 plant damage state and/or containment response. SE1 - Small Early Containment Bypass - SGTR with Scrubbed Release The Level 2 PRA evaluates fission product scrubbing for SG tube rupture events that lead to core damage if water inventory can be maintained/re-established in the faulted SG. With successful SG inventory and scrubbing, the sequence release is a "small" early release (MAAP case #103k). The small-early Level 2 sequences depend on success of EFW/SUFP. SAMG guidance in SAG-1 and SAG-5 provide the TSC and plant operators with guidance for restoring SG level (SAG-1) and reducing fission product release (SAG-5) post core damage. SAG-1 guidance considers assessment and alignment (if necessary) of many system options to restore SG inventory including the use of EFW or SUFP pumps. The Level 2 EFW/SUFP success criteria for restoring SG level after core damage are the same as Level 1 sequences. That is, either one of the EFW pumps or the SUFP is capable to provide the required flow to restore level in the effected SG. This is consistent with SAG-1 guidance. This has been documented/summarized in PRA Report (Tier 2) Section 10.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O LE-C10-01 (SRs LE-C10 & C12)	Closed	LE-C10 and C12 Category II/III require the REVIEW of significant accident progression sequences to determine whether there is a possibility of continued equipment operation or operator actions in adverse environments of post containment failure. No documentation was found to address this requirement and it is acknowledged that meeting Category I is conservative.	The total LERF contribution as determined in the current model of record (SB2011) is 9.2E-08/yr. The ASME standard defines the significant accident progression sequences as those sequences that sum to 95% LERF or with the individual sequence percentage of 1% LERF. The significant accident progression sequences are identified below, based on initiating events that contribute 95% (summed) and 1% (individual) to LERF include the following: **Initiator/ Description / LERF/yr / Contribution / Cumulative** SGTR/ SG Tube Rupture / 4.45E-08 / 48.3% / 43.8% LOC1VI / ISLOCA/ 1.80E-08 / 19.5% / 67.8% E25L / Seismic LOCA (2.5 g) / 7.29E-09 / 7.9% / 75.7% E18L / Seismic LOCA (2.5 g) / 4.36E-09 / 4.7% / 80.4% E18T / Seismic Transient (1.8g) / 3.83E-09 / 4.2% / 84.6% E14A / Seismic ATWS (1.4g) / 3.21E-09 / 3.5% / 88.1% E18A / Seismic ATWS (1.8g) / 2.83E-09 / 3.1% / 91.2% E25A / Seismic ATWS (2.5g) / 2.65E-09 / 2.9% / 94.1% E25T/ Seismic Transient (2.5g) / 1.80E-09 / 2.0% / 96.1% E10A / Seismic ATWS (1.0g) / 1.32E-09 / 1.4% / 97.5% Other <1E-09 / <1% The above LERF initiators and associated significant accident progression sequences are reviewed for possible additional credit of equipment or operator actions that could be applied during the accident progression or after containment failure to further reduce the LERF contribution. SGTR Sequences: SGTR sequences contribute 4.45E-08/yr to LERF. SGTR initiator models steam generator tube rupture occurring randomly. SGTR with subsequent failures can result in core damage with containment bypass. The top ranking SGTR sequences involve a tube rupture with successful SG cooling and primary system makeup but with failure of the main steam safety valve(s) to reseat, failure of the operator to terminate SI, and failure of decay heat removal via long-term primary inventory makeup including credit for decay heat removal using feed and bleed cooling. Also modeled is failure of the operator to establish feed to the faulted SG after core damage but prior to a significant release (scrubbing).

F&O ID	Status	Peer Review Finding	Resolution
F&O LE-C10-01 (continued)			Possible additional actions to further reduce LERF are evaluated below: - Use the fire suppression system as an additional means of injecting water into the ruptured SG to maintain level above the tube breach (effect scrubbing). However, this could only be accomplished if the pressure in the affected SG was low (below 100 psi) which is unlikely to occur in the short term. Therefore, credit for this action cannot be justified. - Use the fire suppression system to spray water above the safety valve discharge; this would tend to reduce the release as a result of scrubbing. However, given the location of the safety valve discharge, equipment needed, limited timing, personnel habitability concerns, and limited scrubbing effectiveness, credit for this action cannot be justified. Conclusion: The SGTR event sufficiently credits equipment and operator actions to realistically mitigate the SGTR LERF during the accident progression and containment bypass failure. There are no additional, practical mitigative actions or equipment that warrant further specific consideration and justification to further reduce the LERF contribution to less than 4.45E-08/yr. The existing SGTR LERF sequences and modeling are judged adequate and no further modeling changes are necessary to reduce SGTR LERF. LOC1VI Sequences: LOC1VI sequences contribute 1.8E-08/yr to LERF. LOC1VI models an interfacing system LOCA (ISLOCA) resulting in containment bypass sequences. The LOC1VI event occurs in the low pressure RHR system due to failure of dual normally closed discharge check valves in one of four injection lines. LOC1VI sequences involve failure of the RHR pipe/heat exchanger or failure of the RHR pump mechanical seal. Sequences that involve RHR pipe or heat exchanger failure are given minimal credit for mitigation because injection water is lost out of the break (located outside containment) and long term sump recirculation is not available. Sufficient time is not available to refill the RWST for long term injection. This break location is assumed to be high in the RHR Equipment Vault, thus the release path is outside containment without scrubbing through a pool of water in the equipment vault. With the break location at the RHR pump mechanical seal, the release path is also outside of containment but with the benefit of scrubbing credited through a pool of water in the equipment vault. The RHR pumps are assumed to not survive the harsh environment given their location relative to the ISLOCA event.

F&O ID	Status	Peer Review Finding	Resolution
F&O LE-C10-01 (continued)			However, the Charging pumps are credited (located in a different building) along with the SI pumps (minimal credit due to their location upper elevation of the RHR vaults) for injection. Operator actions to depressurize the RCS (using SG depressurization and SG makeup with EFW) to limit the rate of RCS inventory loss, and to refill the RWST for long term RCS injection capability are modeled in the RHR pump seal failure event. Possible additional action to further reduce LERF is evaluated below: - Use the fire suppression system to spray water above the RHR pipe rupture (release scrubbing) or to add pool inventory to the RHR vault so as to submerge the pipe break opening to provide release scrubbing. However, given the location of the RHR vault and numerous possible locations for the pipe break within the vault, and personnel habitability concerns, it would be impractical to access the area to implement fire water spray or to flood the equipment vault. Therefore, credit for this action cannot be justified. Conclusion: The ISLOCA LOC1VI event sufficiently credits equipment and operator actions to realistically mitigate the ISLOCA LERF during the accident progression and containment bypass failure. There are no additional, practical mitigative actions or equipment that warrant further consideration and justification to reduce LERF. It is noted that ISLOCA events are of very low frequency and subject to considerable uncertainty as a result of the state-of-knowledge correlations of common or similar components. Given these uncertainties, the existing ISLOCA LERF sequences and modeling are judged adequate and no further modeling changes to reduce LERF are needed.
F&O LE-C10-01 (continued)			Seismic Initiating Event Sequences: Seismic event LERF sequences contribute a total of 2.73E-08/yr to LERF. The seismic initiators, E25L through E10A, represent extreme seismic events (1.0g and larger). These events are assumed to impair/delay effective evacuation and thus contribute to LERF. Without the assumption of impaired evacuation, these seismic event sequences would bin to "late" release bins. Sequences associated with these seismic events include large LOCA seismic initiators with failure of RCS inventory injection (accumulators and low-head injection), small LOCA seismic initiator with failure RCS inventory makeup (high-head injection), ATWS seismic initiator during unfavorable exposure time with resultant overpressure condition that ruptures the RPV, and transient seismic events with failure of decay heat removal equipment. For all of these extreme seismic events, the dominant failure mode of plant equipment is seismic-induced failure. Possible additional action to further reduce LERF is evaluated below: - Given containment failure, use the fire suppression system to spray water at the containment failure location to provide release scrubbing. However, given the severity of these seismic events, the fire protection system is also likely to fail. Therefore, credit for this action cannot be justified. Conclusion: Seismic event sequences sufficiently credit plant equipment and operator actions to realistically mitigate the seismic-induced LERF during the accident progression and containment failure. Given the severity of the seismic events represented by these sequences, there are no additional, practical mitigative actions or equipment that warrant further consideration and justification to reduce LERF.

F&O ID	Status	Peer Review Finding	Resolution
F&O LE-D6-01 (SR LE-D6)	Closed	The analysis does not consider an increased probability of thermally-induced steam generator tube rupture due to depressurized steam generators that may occur due to secondary side conditions as mentioned in item (b) of the SR. In addition, because thermally-induced tube rupture follows hot leg integrity in the event tree, proper consideration of the conditional probabilities should be re-addressed to ensure that it is not receiving a lower probability than it should. As the plant ages, the analysis should also be cognizant that at some point the tubes should no longer be considered 'pristine.'	EPRI TR-107623 Rev. 1 and NUREG-1570 were reviewed. It is noted that to a large extent, the EPRI report is based on insights gained from NUREG-1570. Section 8 of the TR was reviewed for applicability and to ensure that the top event modeling of SGTI (pressure-induced failure before core damage) and XSGTI (temperature-induced SG tube failure after core damage) is reasonably consistent with the EPRI report relative to LERF (section 8 of the report). It is noted that the EPRI Report provides a detailed method to risk-inform application of alternate repair criteria and/or operate with degraded tubing. This risk methodology goes beyond the existing resolution of the Seabrook PRA for modeling of SGTI and XSGTI. However, the existing modeling of SGTI and XSGTI is judged to be robust and adequate to account for all LERF contributors driven by tube failure during severe accident conditions. Further refinement of the model will not produce additional LERF insights. NUREG-1570 report was reviewed for SGTI and XSGTI modeling insights with emphasis on Figure 2.3, Table 5.1a, Table 5.2 and Table 5.6. The NUREG analysis evaluates LERF sequences involving both pressure-induced failures and temperature-induced failures of SG tubes. The sequences evaluated generally consisted of transients (with RCS intact), RCP seal LOCA (with RCS loop seal clearing), and Stuck open PORV. In summary, the analysis suggests that: (1) pressure-induced and/or temperature-induced tube failures are unlikely to occur in sequences where the SGs remain intact and pressurized (2) pressure-induced and/or temperature-induced tube failures are more likely to occur when the SGs are de-pressurized and dry; a factor that can contribute to the timing of a temperature-induced failure is whether the RCS loop seal is cleared causing full circulation of hot gases to the SG tubes, etc. (3) temperature-induced SG tube failure requires RCS pressure; it was shown that it is more likely to have a breach in the RCS due to heating of the hot leg and/or surge line. However, there are considerable uncertainties in the modeling and hot leg creep rupture causing RCS depressurization cannot be a guarantee to avoid a temperature-induced SG tube rupture. (4) SG tube failure probabilities used in the LERF analyses range from approximately 0.05 to 0.1 for pressure-induced failure and 0.05 to 1.0 for temperature-induced failure probability depending on the sequence of events. The EPRI TR values for temperature-induced failure range from 0.003 to 0.85 depending on the sequence (whether it involves loop seal clearing, etc.

Seabrook Station PRA – Summary of Peer Review Findings			
F&O ID	**Status**	**Peer Review Finding**	**Resolution**
F&O LE-D6-01 (continued)			This F&O finding essentially identifies three issues, all of which were evaluated to be minor issues, requiring improved documentation. ISSUE (1): Pressure-induced SG tube failures: The Seabrook analysis does not consider an increased probability due to depressurized steam generators that may occur due to secondary side conditions as mentioned in item (b) of LE-D6. EVALUATION: In fact, the SBK PRA (PRA Report (Tier 2) Section 14.3.3.1) documents the failure probabilities for induced SGTR given secondary side depressurization due to failure of MSSVs (see split fraction SGTI2). MSSV failure would clearly bound the impact of an ASDV failure. ISSUE (2): Temperature-induced SG tube failures: Because thermally-induced tube rupture follows hot leg integrity in the event tree, proper consideration of the conditional probabilities should be re-addressed. EVALUATION: Thermally-induced SG tube rupture that occurs prior to core melt (e.g., induced tube rupture due to steam line break) is modeled in top event SGTI. This includes conditional tube rupture probabilities as high as 0.1 (PRA Report (Tier 2) Section 14.3.3.1). In addition, thermally-induced SG tube rupture that occurs following core melt is modeled in top event XSGTI. This failure mode has low probabilities due to the strength of the SG tubes and the more likely failure of hot leg creep rupture. This approach appears consistent with the insights gained from ERPRI TR 107623 and NUREG-1570. However, even two orders of magnitude increase in this probability (1e-3 to 0.1) would increase LERF less than 1%. This suggests that possible uncertainty and dependence between hot leg creep rupture and XSGTI is not significant from a LERF perspective. ISSUE (3): SGTI and XSGTI failure probabilities: Seabrook should evaluate the health of the SG tubes and determine if any adjustment in the SGTI and XSGTI failure probabilities is warranted due to tube wear or degradation. EVALUATION: Seabrook is designed with Westinghouse Model F SGs with Alloy 600TT tubes. As of OR16 (April 2014) a total of 185 tubes have been plugged (185/22504 = 0.8%). Degradation issues identified through OR16 include: (1) ODSCC (two tubes in OR15 and one tube in OR13), (2) Flow-Induced Vibration Fatigue (some tubes might require dampening (and plugging) to prevent a flow-induced failure, (3) Anti-Vibration Bar Wear (3 tubes were plugged in OR16 due to AVB wear in SG-C), and (4) Secondary Side Deposit Loading (need to limit loading to prevent tube support plate fouling).

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F&O LE-D6-01 (continued)			Overall, the life expectancy of the steam generators is good with some uncertainty based on the number of small cracks identified to date. Also, the high hot leg temperature of 621F (nominal) is a contributing factor to acceleration of the aging/cracking process. Bridging strategies have been developed to address these issues including evaluating the possibility of a hot leg temperature reduction to ensure high confidence of achieving a full 40 year life and beyond. In general, as of April 2014 (OR16), the overall health of the SG tubes is characterized as very good. However, there are signs of slow-progressing degradation after 24 years of operation. As a result and based on judgment, the baseline severe accident SG tube rupture probability for temperature-induced (XSGTI) is increased from 0.001 to 0.1. This is consistent with discussion in WCAP-16600, Table 15 regarding tube degradation over time and possible need to adjust the XSGTI failure probability. In addition, this is consistent with the F&O suggestion to adjust the tube rupture probability given the probability of hot leg remaining intact. The pressure-induced failure probabilities (0.05 and 0.1) are judged to remain appropriate.
F&O LE-E4-01 (SRs LE-E4 & E1)	Closed	The LERF result reported in Section 2.4.2 appears to be a point-estimate result rather than the mean of the uncertainty distribution. Most Level 2 events do not have uncertainty distributions and therefore do not propagate through the uncertainty analysis. State of knowledge uncertainty does not appear to be addressed throughout the model. In order to meet the Capability Category II QU requirements, the mean result from the LERF uncertainty should be reported, including consideration of any state-of-knowledge correlation.	(1) Uncertainty distributions were developed for the remaining Level 2 basic events (split fractions). This was done by modifying the equations for the point estimate split fractions to include multiplication of the point estimate by a LOG Normal data variable (LOGN10). LOGN10 provides a range factor of 10 between the 5th and the 95 and this range factor is judged reasonable for the point estimates, which range in value between approximately 0.5 to 1E-03. This range factor is also consistent with that used in the Level 2 HEP basic event modeling. The Level 2 top events that did not already include uncertainty and were modified under this MCDB action include: XHLI, XNH2E, XNH2V, XRACE, XRACL, XRPV, XSGTI and XSUMP. (2) The Level 1 and Level 2 sequences were reviewed to identify where the state-of-knowledge correlation might be important. It is noted that the SOKC is explicitly accounted for in the ISLOCA evaluation when determining the mean failure rate and uncertainty associated with failure of similar valves that use the same data variables. Based on review of the sequences, it is judged other sequences would not benefit from application of SOKC corrections. (3) The LERF uncertainty distribution and associated mean value are quantified using model SB2014 as part of the 2014 PRA update process and final quantification. The LERF uncertainty results are provided in PRA Report (Tier 2) Section 2.4.2.
F&O LE-G6-01 (SR LE-G6)	Closed	No documentation of the quantitative definition used for a significant accident progression sequence was found. The Standard definition for a significant accident progression sequence was used for the LERF results, but this fact was not documented. Since this is an HLR-G SR which deals with documentation only, this lack of documentation is categorized as a Finding.	The definition of "significant accident progression sequences" used in the LERF analysis is consistent with the definition provided in the ASME PRA Standard, Section 1-2 (Definitions) and QU-F6 (Quantification). Refer to PRA Report (Tier 2) Section 2.5 "Model Review" and Section 2.5.3 "review of Significant Contributors to LERF) for improved documentation.