July 13, 2020
SBK-L-20092
10 CFR 50.90

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

RE: Seabrook Station
Docket 50-443
Renewed Facility Operating License No. NPF-86

Response to Request for Additional Information Regarding License Amendment Requests to Adopt TSTF-411 and TSTF-418

References:

1. SBK-L-18089, License Amendment Request 17-06, Change to the Technical Specification Requirements for Reactor Trip System Instrumentation and Engineered Safety Features Actuation System Instrumentation to Implement WCAP-14333 and WCAP-15376, November 1, 2019 (ADAMS Accession No. ML19310D804)

2. Request for Additional Information Regarding License Amendment Requests to Adopt TSTF-411 and TSTF-418, June 11, 2020 (ADAMS Accession No. ML2016A184)

In Reference 1, NextEra Energy Seabrook, LLC (NextEra) submitted a license amendment request (LAR) to adopt Technical Specification Task Force (TSTF) travelers: TSTF-411, Surveillance Test Interval Extension for Components of the Reactor Protection System (WCAP-15376)," and TSTF-418, "RPS and ESFAS Test Times and Completion Times (WCAP-14333)," for Seabrook Station, Unit No. 1.

In reviewing the submitted information, the U.S. Nuclear Regulatory Commission (NRC) staff has determined that additional information is necessary to complete its review. Reference 2 documents the requests for additional information (RAI).

The enclosure to this letter provides the responses to these RAIs.

There are no new or revised commitments made in this submittal.

Should you have any questions regarding this submission, please contact Mr. Kenneth Browne, Safety Assurance and Learning Site Director, at 603-773-7932.

Sincerely,

Eric McCartney

Mr. Eric McCartney
Site Director (VP)
NextEra Energy Seabrook, LLC

cc: USNRC Region I Administrator
USNRC Project Manager
USNRC Senior Resident Inspector

Director Homeland Security and Emergency Management
New Hampshire Department of Safety
Division of Homeland Security and Emergency Management
Bureau of Emergency Management
33 Hazen Drive
Concord, NH 03305

Katharine Cederberg, Lead Nuclear Planner
The Commonwealth of Massachusetts
Emergency Management Agency
400 Worcester Road
Framingham, MA 01702-5399

RAI 01 – PRA Maintenance and Upgrades

Table 3.2.1-1-f of the license amendment request (LAR) provides a list that summarizes the significant probabilistic risk assessment (PRA) model changes and core damage frequency (CDF) impacts for the Seabrook internal events PRA (IEPRA) that includes internal floods. The U.S. Nuclear Regulatory Commission (NRC) staff observes that there have been several model updates since the last peer reviews were performed (e.g., complete revision of latent human failure event analysis and complete revision of the internal floods analysis in 2011). It is not clear whether these PRA changes are considered PRA upgrades or PRA maintenance in accordance with the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard 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," as endorsed by Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090410014). Furthermore, it is not clear for the updates incorporated into the IEPRA (includes internal floods) whether these upgrades, if any, have been peer reviewed or align to the scope of a specific peer review(s) provided in Section 3.4 of the LAR.

Based on the observations provided above, address the following:

a. For the PRA model changes that occurred in 2011 and 2014 provided in LAR Table 3.2.1-1-f, provide sufficient justification to ascertain whether each of the model changes constitutes a PRA maintenance or upgrade (i.e., describe the change and include justification) as defined in the ASME/ANS RA-Sa-2009 PRA standard, as qualified by RG 1.200, Revision 2. For each change, indicate whether the change was determined to be PRA maintenance or a PRA upgrade consistent with the endorsed PRA standard.

b. For any PRA upgrades identified in Part (a) above, either:

   i. Identify any focused-scope (or full-scope) peer reviews that were performed in accordance with RG 1.200, Revision 2, to address each PRA upgrade. Provide the findings of these peer reviews and the associated dispositions as it pertains to the impact on this LAR if not already provided in the submittal dated November 1, 2019 (ADAMS Accession No. ML19310D804).

   ii. Alternatively, if a peer review was not performed to address the upgrade(s), provide sufficient information for the NRC staff to compare the technical adequacy of the upgrade to RG 1.200, Revision 2, or provide a bounding or sensitivity evaluation of its effect to demonstrate that it has no impact on the conclusions of this risk-informed application.

NextEra Response

a

Table 3.2.1-1-f (Note f above) Summary of Significant Model Changes and CDF Impacts		
Year Update	Internal Event CDF	Summary of Significant Model Changes
2014	6.6E-06	- Update to the plant-specific data distributions. - Update of the Internal Flooding initiating event data consistent with EPRI-2010. - Assessment of Level 1 HRA Success Criteria with MAAP408. - Updates to HRA with review of EOPs. - Update of Reactor Coolant Pump (RCP) seal LOCA model.
2011	7.1E-06	- Complete revision of Internal Flood analysis. - Complete revision of latent human failure event analysis.

2014 -
- Update to plant specific data - Data update is considered to be model maintenance and not an upgrade.
- Update of the Internal Flood IE - Integration of IE model with the 2011 IF model. Data is from the 2011 IF model upgrade that was peer reviewed.
- Assessment of the L1 HRA – Updated success criteria utilizing information from the MUG (MAAP Users Group). This was a data update utilizing more conservative information and is maintenance.
- Updates of the HRA with review of EOPs - Reviewed and updated plant-specific procedure / steps identified for each operator action in the human action documentation against the current procedures. This was a documentation change only and as such not an upgrade.
- Update of the RCP seal LOCA model is discussed in the response to RAI 03. The changes were maintenance and not an upgrade.

2011 -
- The IF model revision - Upgraded and peer reviewed in 2009. Following the peer review updates were made to the model to address the findings. These finding resolutions were independently reviewed and none of the resolutions were determined to be upgrades. Two findings were deemed still open. These findings were resolved in SSPSS 2014. The resolutions were reviewed and closed as part of the current MOR's peer review. The two findings with a statement of impact to this application is provided in Table 3.4.1 of the LAR submittal.
- Latent Human Failure Event analysis revision -Updated to resolve F&Os HR-A3, HR-C2, and HR-D2. These finding resolutions were reviewed and closed via independent review in 2018. They were determined to be updates by NEE with independent review concurrence.

b. All open findings and dispositions are provided in the LAR, Table 3.4.1.

RAI 02 – Open Peer Review Findings and Self-Assessment Items

Tables 3.4.1 and 3.4.2 of the LAR provide the dispositions and resolutions for the Facts and Observations (F&Os) and self-assessment items that remain open following the peer reviews and Independent Assessment performed for the IEPRA. The disposition/resolution for F&O HR-E3-1 does not provide sufficient justification for the staff to discern that there is no impact for the requested technical specification (TS) change or that the licensee has resolved the F&O to meet

capability category (CC) II for the associated supporting requirement (SR) of ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2.

For the disposition of F&O HR-E3-1, the licensee stated that a comprehensive review of all post-initiator dynamic operator actions associated with scenarios initiated at-power was performed by a "former" Seabrook Station operations shift manager and that only a documentation change is needed to close this finding and not expected to have any impact on the human error probabilities (HEPs) or PRA results for this risk-informed application.

Furthermore, the independent review team concluded that the action taken (i.e., comprehensive review performed by the former personnel) to address this finding does not fully meet the intent of the SR. Procedures and operational practices continue to be revised and modified, therefore, the NRC staff interprets the intent of SR HR-E3 is to be performed by a "current" plant personnel to confirm that the interpretation of the procedures remains valid and is consistent with any revised or new plant observations and training procedures.

EITHER:

i. Provide justification (e.g., results and description of a sensitivity study performed) to confirm that this F&O has no impact on the conclusions of this risk-informed application.

OR

ii. Perform the review and talk throughs with current plant operations and training personnel and update the human reliability analysis and IEPRA as necessary to meet the intent of SR HR-E3 at CC II.

NextEra Response

Background - Finding HR-E3 states:

While simulator exercises were observed, there is no evidence of specific talk-throughs with Operations/Training, nor is there is reference for the need for this activity in the HRA Guidance. 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.

POSSIBLE RESOLUTION:

Include talk-throughs with, and/or review by Operations or Training as part of the process for finalizing the HRA.

NEEs complete Resolution:

In 2009 Update, a comprehensive review of modeled operator actions was performed by a retired shift manager, and training instructor. This individual has 20+ years of experience at Seabrook Station, including participation in the development of the Operations department processes and procedures and early Westinghouse Owners Group Emergency Response Guidelines (ERGs) as well as the first 10+ years of

operation. His experience in both operations and training gives him a broad perspective to review the operator actions as modeled in the PRA. His 2008 assessment and re-development of time-critical aspects of station Appendix R procedures provided insight into the most recent operator protocol for strict three-way communication and rigorous place keeping standards.

The 2009 analysis consisted of reviewing all the post-initiator dynamic operator actions associated with scenarios initiated at-power. The retired SEA operations/training person reviewed each action as it is documented in the HRA Calculator Version 4 and made revisions based on his research and knowledge and, where needed, the support of current operations and training personnel. To check the time required for actions, he performed detailed talk-throughs of the actions using current procedures and images of MCB panels as needed. These talkthroughs were with SEA active operators.

The retired shift supervisor is the individual leading the talkthroughs with operations staff. Not the individual whom PRA held the talkthroughs with.

The finding resolution as documented in the LAR submittal was incomplete in identifying the entire action taken to resolve this finding. The updated resolution documents that the talkthroughs were completed as needed for this SR and no further action is needed. The model documentation was updated to more clearly note this for future reviews.

RAI 03 – Reactor Coolant Pump Seal Modeling

Section 2.5.1.2 of RG 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (ADAMS Accession No. ML17317A256) states, "[t]he use of models for specific events or phenomena supports the development of the PRA model. In many cases, the industry's state of knowledge is incomplete, and opinions may vary on how the models should be formulated. Examples include approaches to modeling human performance, CCFs [common cause failures], and reactor coolant pump seal behavior upon a loss of seal cooling. This gives rise to model uncertainty." Section 2.5.1.2 of RG 1.174, Revision 2, states, "[t]he impact of using alternative assumptions or models may be addressed by performing appropriate sensitivity studies or by using qualitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models." In addition, Section 2.5.2 states, "[i]n general, the results of the sensitivity studies should confirm that the guidelines are still met even under the alternative assumptions (i.e., change generally remains in the appropriate region)."

In Table 3.2.1-1-f of the LAR, NextEra provides a summary of significant model changes and CDF impacts that include an update of the Reactor Coolant Pump (RCP) seal loss of coolant accident (LOCA) model that is credited in the IEPRA. Provide the following information to validate and confirm the PRA acceptability of the RCP seal LOCA model used in the risk evaluation performed to support the requested risk-informed application.

a. Provide a summary of the update to the RCP seal LOCA model, addressing the following aspects:

    i. Describe any RCP seal modifications at Seabrook Station (e.g., installation of Westinghouse Generation III shutdown seals) associated with the updated RCP seal LOCA model.

ii. Describe the PRA update of the RCP seal LOCA model.

iii. Discuss the credit taken for any RCP seal modifications and the technical basis for that credit (e.g., technical report approved by the NRC). Demonstrate how any limitations and conditions delineated in the applicable NRC-approved guidance are being met [e.g., if the Westinghouse Generation III shutdown seals were installed and credited, those limitations and conditions in Section 3 of Topical Report PWROG-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," and Section 5 of NRC safety evaluation (SE) for PWROG-14001-P (ADAMS Accession No. ML17200A116)].

iv. Indicate, and provide justification, whether the updated RCP seal LOCA model is PRA maintenance or PRA upgrade, as defined in Section 1-5.4 of ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2. This discussion should be of sufficient detail to allow NRC staff to independently assess whether this change is a PRA maintenance or PRA upgrade (e.g., summarize the original method in the PRA and the new method, summarize the impact that this change has on significant accident sequences or the significant accident progression sequences).

b. If the PRA update of the RCP seal LOCA model is considered a PRA upgrade and a peer review(s) was performed for this upgrade, then discuss this peer review(s). In this discussion, describe the peer review process applied to the model; identify the guidance used to perform this peer review(s) (e.g., ASME/ANS RA-Sa-2009, NEI 05-04, RG 1.200, Revision 2); include any necessary gap- or self-assessments if current endorsed guidance/standards were not used in the peer review(s); provide all F&Os characterized as findings from the peer review(s) and the associated dispositions as it pertains to this application.

c. If the PRA update of the RCP seal LOCA model is considered a PRA upgrade and a peer review was not performed for this upgrade, then perform an appropriate sensitivity and/or bounding analysis (e.g., remove credit for RCP shutdown seals) that assesses the contribution of risk for the risk-informed application. Discuss this sensitivity/bounding analysis and provide updated risk values from the LAR to assess the risk impact. Confirm that the results of this analysis still meet the acceptance guidelines in RG 1.174, Revision 2, WCAP-14333-P-A, WCAP-15376-P-A, and RG 1.177, Revision 1, “An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications” (ADAMS Accession No. ML100910008). If the acceptance guidelines are exceeded, then provide qualitative or quantitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models, to support the conclusions of the LAR. This discussion should include which metrics are exceeded and the conservatisms in the analysis and the risk significance of these conservatisms.

NextEra Response

a. The 2011 Seabrook PRA update reflected a planned installation of RCP shutdown seals. Since the seals were not yet installed however, a 'switch' was programmed into the fault tree to inactivate use of the new logic. Therefore, no credit was taken for the RCP seal modifications in the SSPSS2014 MOR. This is conservative with respect to the current configuration of the plant, where RCP shutdown seals are installed.

b. N/A

c. N/A

RAI 04 – Treatment of Key Assumptions and Key Sources of Uncertainty

Section 3.3.2 of RG 1.200, Revision 2, provide guidance on how to identify, characterize and treat key sources of uncertainty relevant to a risk-informed application. Additionally, Section 3.3.2 of RG 1.200, Revision 2, defines *key* assumptions and *key* sources of uncertainty to be considered for "relevance" to the risk-informed application.

The licensee further states in Section 3.4 of the LAR, "there are no sources of uncertainty that have an impact on the internal events and internal flooding quantitative risk model and thus there is no impact on the proposed application..." The NRC staff requests the following information to confirm the key assumptions and key sources of uncertainty were properly assessed from the base IEPRA (includes internal floods) model that has received peer reviews:

a. Provide a description of the process used to determine the *key* sources of uncertainty and *key* assumptions for the IEPRA (includes internal floods) model used to support this application.

    i. A description of how *key* assumptions and *key* sources of uncertainty were assessed from the initial comprehensive list of PRA model(s) (i.e., base model) sources of uncertainty and assumptions, including those associated with plant-specific features, modeling choices, and generic industry concerns determined to be applicable for this risk-informed application.

    ii. A discussion on how the process and the criteria used to identify an assumption or source of uncertainty as "key" is consistent with RG 1.200, Revision 2, or other NRC-accepted methods.

b. If the process of identifying the *key* assumptions or *key* sources of uncertainty for the IEPRA model used to support this application cannot be justified for use in this risk-informed application, provide the results of an updated assessment that includes a description of each *key* source of uncertainty or *key* assumption identified along with the disposition (e.g., sensitivity study, etc.) to address the impact of each on this risk-informed application. Also, provide a description of the process used to determine the *key* sources of uncertainty and *key* assumptions in the *updated* assessment and how this process is consistent with RG 1.200, Revision 2, or other NRC-accepted methods.

NextEra Response

a. The following reports were used in developing the list of uncertainties and assumptions for the Seabrook PRA. WCAP-16432 *Process for Identifying Assumptions within a PRA* (May 2005) expands the term *plant risk profile* to mean different insights into plant operation or maintenance or a relatively large change (e.g., factor of 2 increase) in CDF or LERF. This WCAP provides an approach to collecting and categorizing assumptions. WCAP-16282 *WOG Guidelines for PRA Key Assumptions* (June 2004) provides a list of candidate assumptions. EPRI TR-1009652 *Guideline for the Treatment of Uncertainty in Risk-Informed Applications* (Dec 2004) gives a list of generic causes of uncertainty. This information was used to develop Seabrook's process described below. This process was validated to meet NRC accepted methods by an independent review and self-assessment.

The process for identifying sources of uncertainty and related assumptions, based on the available industry reference documents and regulatory guidance described above is as follows:

Step 1 - Identify and evaluate EPRI generic sources of uncertainty for applicability to Seabrook.

Step 2 - Identify explicit plant-specific assumptions by searching existing site PRA documentation.

Step 3 - Identify implicit plant-specific assumptions by reviewing the High Level Requirements for each element in the ASME PRA Standard.

Step 4 - For each assumption, identify the related source of uncertainty that the assumption addresses.

Step 5 - Categorize assumptions & sources of uncertainty by the related Technical Element from the ASME PRA Standard.

This process is applied to the Seabrook PRA and the uncertainties and assumptions are documented in site documentation.

Plant-specific lists of assumptions and uncertainties for each of the PRA technical areas were developed based on a previous version of the PRA. In some cases, changes made to the PRA since the time of the initial evaluation have changed or eliminated the assumption or uncertainty. These were either deleted or noted that they are not applicable in the list of assumptions and uncertainties. If model changes have introduced any new assumptions or uncertainties, these are included in the plant-specific listings as well.

b. N/A