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U. S. Nuclear Regulatory Commission
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Joseph M. Farley Nuclear Plant
Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications,
SNC Response to NRC Request for Additional Information (RAI)

Ladies and Gentlemen:

By letter dated July 27, 2018, (Agencywide Documents Access and Management System Accession No. ML18208A619), Southern Nuclear Operating Company, Inc. (SNC) submitted a License Amendment Request (LAR) for Joseph M. Farley Nuclear Plant, Units 1 and 2. The proposed amendment requested U.S. Nuclear Regulatory Commission (NRC) approval to modify the Technical Specifications to permit the use of Risk Informed Completion Times in accordance with NEI 06-09, Revision 0-A, Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines. By email dated March 12, 2019 the NRC staff notified SNC that additional information is needed for the staff to complete their review. Enclosure 1 provides the SNC response to NRC requests for additional information (RAIs). Enclosure 1 also includes an SNC supplemental request. The supplemental request and some RAIs required SNC to modify portions of the Operating license, including technical specification changes. The markups and clean type pages for the Operating License are included in Enclosure 2 and 3. SNC plans to supplement this LAR by providing corresponding Technical Specification Bases Markup changes related to the RAI responses, for information only, within 60 days of this letter.

SNC originally requested an implementation date of 120 days from the of issuance of the safety evaluation for this LAR. SNC would like to revise that request and proposes an implementation date of 180 days from the issuance of the safety evaluation.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 3rd day of May 2019.

Respectfully submitted,

Cheryl Gayheart
Director, Regulatory Affairs
Southern Nuclear Operating Company

CAG/PDB/SCM

Enclosures:

1. SNC Response to NRC Request for Additional Information (RAI)
2. Operating License and Technical Specifications Markups
3. Operating License and Technical Specifications Clean Type Pages

cc: Regional Administrator, Region II
NRR Project Manager – Farley
Senior Resident Inspector – Farley
Director, Alabama Office of Radiation Control
RTYPE: CFA04.054

Joseph M. Farley Nuclear Plant Units 1&2
Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications, SNC Response to NRC Request for Additional Information (RAI)

Enclosure 1

SNC Response to NRC Request for Additional Information (RAI)

NRC RAI 01 – Summary of Probabilistic Risk Assessment (PRA) Peer Reviews

Limitation and Condition 3 in the NRC staff's safety evaluation (SE) on NEI 06-09 states:

The LAR will provide a discussion of the results of peer reviews and self-assessments conducted for the plant-specific PRA models which support the RMTS, including the resolution or disposition of any identified deficiencies (i.e., findings and observations from peer reviews).

LAR Enclosure 2 stated that a full scope peer review for the internal events PRA (IEPRA) was performed in March 2010 against the 2009 PRA standard as endorsed by Regulatory Guide (RG) 1.200, Revision 2 (ADAMS Accession No. ML090410014), and using the process defined in NEI 05-04 (ADAMS Accession No. ML083430462). The LAR further stated that the fire PRA (FPRA) had undergone a peer review in accordance with NEI 07-12 (ADAMS Accession No. ML102230070) as endorsed by RG 1.200, Revision 2. The licensee provided the list of Facts and Observations (F&Os) and the current status of each one in Tables E2-2 and E2-4. However, the LAR did not provide any further peer review information or discussion of any efforts to resolve F&Os using NRC accepted processes.

For each PRA model (IEPRA, internal flooding PRA (IFPRA), and FPRA) provide a summary of all peer reviews and F&Os closures that were conducted after the most recent full-scope peer review. Include a summary description of scope, results and whether the resulting F&Os were addressed in the PRA models and associated documentation.

SNC Response to NRC RAI 01

For the Farley Internal Events including Internal Flooding PRA model (IEIFPRA), a F&O closure review was conducted by an Independent Assessment team. The scope was to review the F&O resolutions to determine if the issues identified in each F&O was addressed to meet the Capability Category II (CC-II) requirements that were applicable to each finding-level F&O. In addition to assessing the closure status, the Independent Assessment team reviewed F&O resolutions to determine whether the changes constituted a PRA upgrade.

Based on the results of the onsite closure review session, all the IEIFPRA F&Os were not determined to be closed. For the remaining open F&Os, the Independent Assessment team made recommendations to perform additional updates to the IEIFPRA documentation to ensure complete closure of the F&Os.

Following the on-site F&O closure review, the IEIFPRA documentation was revised by SNC to incorporate recommendations from the Independent Assessment team. The updated IEIFPRA documentation was provided to the Independent Assessment team. The Independent Assessment team concluded that all F&Os were closed and none of the changes constitute a PRA upgrade.

For the Farley FPRA, a focused-scope peer review was conducted in January 2018. The scope was to review the changes made during this most recent Farley FPRA update as part of NFPA 805 implementation, which constitute a change in methodology, that incorporated: guidance on incipient fire detection, updated spurious operation treatment, guidance for cable tray fires, and guidance for junction box fires. One finding-level F&O was generated from the review.

Following this focused-scope peer review in January 2018, a F&O closure review was conducted by an Independent Assessment team in April 2018. The scope of the F&O closure review was to determine if the resolution of F&Os, including the resolution for the one finding-level F&O from the January 2018 focused-scope peer review, meet CC-II requirements for finding-level F&Os, and if the changes constitute a PRA Upgrade. Changes made to the Farley FPRA to resolve the F&Os were considered by the Independent Assessment team review to not constitute a PRA upgrade with the exception of two F&O resolutions.

Another focused-scope peer review was conducted in July 2018. The scope was to review the two F&O resolutions, fire scenarios done with ignition sources characterized with one fire intensity and the treatment of secondary combustibles, that were determined to constitute a PRA upgrade. Two finding-level F&Os were generated from this review.

After the on-site F&O closure review and the focused-scope peer review in July 2018, updated FPRA documentation that incorporated recommendations for documentation enhancements were provided to the Independent Assessment team. Based on the documentation updates, all F&Os were determined to be closed including the two finding-level F&Os generated from the latest focused-scope peer review.

NRC RAI 02 – F&O Closure for the Internal Events, Internal Flooding, and the Internal Fire PRA

RG 1.200, Revision 2 provides guidance for addressing PRA acceptability. RG 1.200, Revision 2, describes a peer review process using 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 one acceptable approach for determining the technical acceptability of the PRA. The primary result of a peer review are the F&Os recorded by the peer review team and the subsequent resolution of these F&Os. A process to close finding-level F&Os is documented in Appendix X to the Nuclear Energy Institute (NEI) guidance documents NEI 05-04, NEI 07-12, and NEI 12-13, (ADAMS Accession No. ML12240A027), titled “NEI 05-04/07-12/12-06 Appendix X: Close-out of Facts and Observations (F&Os)” (ADAMS Package Accession No. ML17086A431), which was accepted by the NRC in a letter dated May 3, 2017 (ADAMS Accession No. ML17079A427).

An F&O closure review meeting was conducted in October 2018 at the SNC offices, which the staff observed, to close out the F&Os associated with the Farley IEPRA and IFPRA. Additionally, an F&O closure review was completed in September 2018 for the Farley FPRA model. For each F&O closure review address the following:

a. Provide the specific dates of the F&O closure, including when the on-site review was performed, any closures performed remotely after the on-site review, and when the final report was issued.

b. Summarize the results of the F&O closure reviews and, if applicable, update the dispositions for any finding-level F&Os, not closed by the F&O closure review, by explaining how the F&Os are resolved or dispositioned for this application.

c. Confirm that all finding-level F&Os from all the applicable peer reviews as described in RAI No. 1, including findings against PRA supporting requirements that were met at Capability Category (CC) II, were provided to the independent assessment (IA) team for the F&O

closure reviews.

d. Appendix X guidance states in part, "[a]dditionally, the team will review the [Supporting Requirements] SR to ensure that the aspects of the underlying SR that were previously not met, or met at CC I, are now met, or met at CC II".

   Confirm that closure of the F&Os was assessed to ensure that the capabilities of the PRA elements, or portions of the PRA within the elements, associated with the closed F&Os now meet all ASME/ANS RA-Sa-2009 SRs at CC-II.

e. Appendix X documents three stages to the F&O closure review: the pre-review activities, the on-site review, and closure after the on-site review, including the issuance of the final report. Appendix X also states that the scope of the review "may be expanded to include a concurrent focused-scope peer review to address changes to the PRA model that represent an upgrade per the definition of upgrade in the ASME/ANS PRA Standard."

   Regarding the pre-review activities, Appendix X states:

   The host utility should provide the complete and relevant review materials to the independent assessment team at least two weeks prior to the on-site review.

   Regarding the on-site review, Appendix X states:

   If the independent assessment team determines, during the course of the on-site review, that other PRA changes constitute an upgrade, they may decide to conduct the focused-scope review if time and expertise permit. If this is not possible, the team will indicate the associated findings as "Not Reviewed" with recommendation for a subsequent focused-scope review.

   Regarding the closure after the on-site review, Appendix X states:

   In some cases the host utility's resolution of the finding may be delayed based on questions from the independent assessment team, or other action being taken by the host utility in response to the team's questions.… The host utility may, in the time between the on-site review and the finalization of the independent assessment team report, demonstrate that the issue has been addressed, that a closed finding has been achieved, and that the documentation has been formally incorporated in the PRA Model of Record. The independent assessment team will then re-review the host utility's resolution and associated documentation and a separate consensus session will be conducted as described earlier in this procedure.

   Please address the following:

   i. Describe whether resolution to any F&O was found by the IA team to meet the definition of a PRA upgrade, consistent with the ASME/ANS RA-SA-2009 PRA standard as endorsed by RG 1.200 Revision 2.

   ii. Confirm that a focused-scope peer review was performed for any upgrade identified in part i above and describe its scope.

   iii. As described in Appendix X, closure after the on-site review is intended for resolution of

findings presented to the IA team prior to the on-site review.

Therefore if focused-scope peer review(s) were performed concurrently with or subsequent to the on-site F&O closure, provide all the F&Os that resulted from the focused-scope peer review(s) and their associated dispositions for the application.

SNC Response to NRC RAI 02

From October 29, 2018 through October 31, 2018, an on-site F&O closure review was conducted for the Farley Internal Events including Internal Flooding PRA model (IEIFPRA) by an Independent Assessment (IA) team in accordance with NEI 05-04/07-12/12-06 Appendix X process. All finding-level F&Os, including findings against PRA supporting requirements that were met at CC-II, were provided to the IA team for the F&O closure review.

Following the completion of the onsite closure review session, all the F&Os were not determined to be closed. For the remaining 16 open F&Os, the review team made recommendations to perform additional updates to the PRA models and documentation to ensure complete closure. Once updated documentation was provided, the Independent Assessment team concluded that all F&Os were closed and none of the changes constitute a PRA upgrade.

Closure of all finding-level F&Os was assessed to ensure that the capabilities of the PRA elements, or portions of the PRA within the elements, associated with the closed F&Os now meet all ASME/ANS RA-Sa-2009 SRs at CC-II. The IEIFPRA F&O closure report was sign and dated on November 29, 2018.

From April 24, 2018 through April 26, 2018, an onsite F&O closure review was conducted for the Farley Fire PRA (FPRA) model by an IA team in accordance with NEI 05-04/07-12/12-06 Appendix X process. All finding-level F&Os, including findings against PRA supporting requirements that were met at CC-II and the one finding-level F&O issued during the January 2018 focused-scope peer review, were provided to the IA team for the F&O closure review.

The completion of the onsite closure review session left 19 open F&Os. The IA team determined that 2 of the remaining 19 open F&Os were PRA upgrades. To close these F&Os, a subsequent focused-scope peer review was conducted over the period of July 5 through July 11, 2018, with additional consensus discussions later in July. Twelve affected SRs were reviewed. The review team reviewed the technical adequacy of the FPRA and compliance with each of the requirements as compared to current PRA practices in the industry. Two finding-level F&Os (Table RAI 02-1) were issued during this peer review.

For 17 of the remaining 19 open F&Os, the review team made recommendations to perform additional updates to the PRA models and documentation to ensure complete closure. Once updated documentation was provided, the IA team, with additional reviews and consensus sessions conducted remotely in late April through September 2018, concluded that all 17 F&Os were closed and none of the changes constitute a PRA upgrade. In addition, the IA team reviewed the resolution to the two finding-level F&Os issued during the July focused-scope peer review. These too were determined to be closed.

Closure of all finding-level F&Os was assessed to ensure that the capabilities of the PRA elements, or portions of the PRA within the elements, associated with the closed F&Os now meet all ASME/ANS RA-Sa-2009 SRs at CC-II. The FPRA F&O closure report was sign and dated on September 24, 2018.

Table RAI 02-1					
F&O	Review Element	Other Affected SRs	Topic	Finding Description	Disposition for 4b
19-3	FSS-H9	UNC-A2	No discussion of newly-added modeling treatments and the discussion of uncertainties related to HRRs appears to be out of date.	No discussion of sources of uncertainty is included for secondary ignition or multi-point intensity in the 014 (FPRA QU) report. Qualitative discussion appears to rely on outdated information as well for other FSS tasks. In Appendix D and other locations in the FPRA QU report (014), Task 8 and 11 have a lot of disposition for uncertainty being bounded by use of NUREG/CR-6850 HRRs. Yet most of the fire modeling study has been updated to use NUREG-2178 HRR data which is typically a significantly lower HRR than the 6850 data. This should be updated in the QU report's assessment of HRR bounding uncertainty since the HRRs are now potentially significantly lower than when these assessments were originally written – including the multi-point work. While the secondary scenarios still use 6850 HRRs, the secondary scenarios are not discussed in any of the Task uncertainty contribution writeups, include their contribution to overall uncertainty. (This F&O originated from SR FSS-H9) <u>Possible Resolution</u> Update the discussion of sources of uncertainty to reflect the updates made to fire modeling, including the use of updated HRRs.	This finding has been closed by Appendix X F&O Closure Review and it does not impact the RICT calculations as the F&O pertains to a documentation issue.

Table RAI 02-1					
F&O	Review Element	Other Affected SRs	Topic	Finding Description	Disposition for 4b
19-4	FSS-H1	N/A	Appendix E.5 documentation is no longer consistent with the actual PRA models and scenarios.	Appendix E.5 of the 008A report will be out of date after next MOR QU, is the plan to keep this appendix up to date each time or to include secondary cases in the QU results documentation? One would expect the secondary cases to just be part of the normal QU documentation, suggest deleting the entire section that compares old/new results for secondary cases only. If retained, the following notes should be updated: 008A appendix E.5 Section 4.2 and Section 5 notes: “5. The flame spread rate for thermoplastic cables was changed from 3 mm/s to 0.9 mm/s as recommended in NUREG/CR-6850 (Ref. 5) and NUREG/CR-7010, Volume 1 (Refs. 9, 10, 12).” Is thermoset intended since that is the assumed cable type in the Farley FPRA? If so, ensure appropriate rates are noted. “The closest example that is a PRA upgrade to the FNP secondary combustible update is Example 13” #13 is not presented in the materials “but does not include new methods originally contained in the GFMT” reword, new methods originally included is illogical (This F&O originated from SR FSS-H1) <u>Possible Resolution</u> Update Appendix E.5 of the 0008A report to remove the justification for the treatment of secondary cases	This finding has been closed by Appendix X F&O Closure Review and it does not impact the RICT calculations as the F&O pertains to a documentation issue.

NRC RAI 03 – Incorporation of Internal Events PRA F&O Resolutions into the Internal Fire PRA

RG 1.200, Revision 2 provides guidance for addressing PRA acceptability. RG 1.200 describes a peer review process using an endorsed ASME/ANS PRA standard (currently ASME/ANS-RA-Sa-2009) as one acceptable approach for determining the technical acceptability of the PRA.

Prior to the IEPRA and IFPRA F&O closure review, all open finding-level FPRA F&Os were closed in a FPRA F&O closure review that was initiated in April 2018 and finalized in September 2018. The IEPRA model forms the basis for the FPRA plant response model, and therefore, resolutions to findings in the IEPRA can also impact the adequacy of the FPRA model. In light of these observations, address the following:

a. Summarize any IEPRA modeling updates that were not incorporated into the FPRA, and provide justification regarding why these updates do not apply to the FPRA or do not have an impact on the application.

b. As an alternative to part (a) above, propose a mechanism that ensures all applicable IEPRA modeling updates that were performed to resolve F&Os for the F&O closure review are incorporated into the FPRA prior to implementation of the RICT program.

SNC Response to NRC RAI 03

The updates to the IEIFPRA model from the F&O resolutions include the incorporation of the periodic data update, dual unit loss of service water via loss of service water dam and river water initiator model, new pre-initiator human failure events (HFEs) for the Diesel Fuel Oil system, detailed human reliability analysis (HRA) performed on post-initiator HFEs that used screening values, and updated HRA dependency analysis. SNC proposes a licensee condition that these updates from the F&O resolutions of the IEIFPRA model shall be incorporated into the FPRA, per our normal PRA configuration process, prior to implementation of the RICT program. This is included in Enclosures 2 and 3 of this letter.

NRC RAI 04 – Potential Credit for FLEX Equipment or Actions

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16 06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC's staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision-making in accordance with the guidance of RG 1.200, Revision 2. Though implementation of FLEX procedures is cited in the LAR as possible Risk Management Action (RMAs), the LAR and other docketed information do not indicate that SNC has credited FLEX equipment or actions into their internal events or FPRA models. As such, please address the following:

a. State whether FLEX equipment and strategies have been credited in the PRA. If not incorporated or their inclusion is not expected to impact the PRA results used in the RICT program, no additional response is requested.

b. If the equipment or strategies have been credited, and their inclusion is expected to impact the PRA results used in the RICT program, please provide the following information separately for IEPRA, external hazards PRA, and external hazards screening as appropriate:

i. A discussion detailing the extent of incorporation, i.e. summarize the supplemental equipment and compensatory actions, including FLEX strategies that have been quantitatively credited for each of the PRA models used to support this application.

ii. A discussion detailing the methodology used to assess the failure probabilities of any modeled equipment credited in the licensee's mitigating strategies (i.e., FLEX). The discussion should include a justification explaining the rational for parameter values, and whether the uncertainties associated with the parameter values are considered in accordance with ASME/ANS RA-Sa-2009 as endorsed by RG 1.200, Revision 2.

iii. A discussion detailing the methodology used to assess operator actions related to FLEX equipment and the licensee personnel that perform these actions. The discussion should include:

1. A summary of how the licensee evaluated the impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of ASME/ANS RA-Sa-2009.

2. Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and if the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of ASME/ANS RA-Sa-2009.

3. If the licensee's procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

c. ASME/ANS RA-Sa-2009 defines PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 in Part 1 of ASME/ANS RA-Sa-2009 states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this Standard.

i. Provide an evaluation of the model changes associated with incorporating mitigating strategies, which demonstrates that none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences, OR

ii. Propose a mechanism to ensure that a focused-scope peer review is performed on the model changes associated with incorporating mitigating strategies, and associated F&Os are resolved to Capability Category II prior to implementation of the RICT program.

<u>SNC Response to NRC RAI 04</u>

FLEX equipment and strategies have not been credited in the IEPRA (including internal flooding) or FPRA for this application. Equipment and operator actions that were part of the as-built, as-operated plant and were previously credited in the PRA prior to FLEX implementation remain credited.

NRC RAI 05 – Modeling of the Reactor Coolant Pump (RCP) Shutdown Seals

Enclosure 7 states that the plant has been modified to install the RCP shutdown seals and that the shutdown seal is modeled consistent with WCAP-17100-NP, "PRA Model for the Westinghouse Shutdown Seal," Revision 1 (ADAMS Accession No. ML101020568), which applies to Generation I and II. In the licensee's October 12, 2017 response to NRC RAIs for the Containment Integrated Leakage Rate Testing Program (ADAMS Accession No. ML17285B308), the licensee stated that the PRA includes credit for the Westinghouse Generation III RCP seals.

The PRA model for the Generation III Seals was approved by the NRC in the final safety evaluation of Topical Report (TR) PWROG-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," dated the August 23, 2017 (ADAMS Accession No. ML17200C875).

Consistent with the guidance in RG 1.174, Revision 3, that the PRA scope, level of detail and technical acceptability be based on the as-built and as-operated and maintained plant, and reflect operating experience at the plant, please address the following:

a. Clarify what kind of seals are installed in each RCP in Farley Unit 1 and 2 and whether the current internal events and fire PRA models include credit for the Westinghouse Generation III ("SHIELD") RCP seals.

b. If the internal events or the internal fire PRA models include credit for the Westinghouse Generation III RCP seals, address the following:

   i. Confirm that the limitations and conditions in the NRC safety evaluation for PWROG-14001-P, Revision 1, are met.

   ii. If exceptions to the limitations and conditions exist, identify all the exceptions and justify their impact on the application.

   iii. Clarify whether the Generation III Westinghouse RCP seal model has been peer-reviewed as part of the internal events PRA and fire PRA peer-reviews.

   iv. If this RCP seal model has not been peer reviewed, justify why the addition of this model is not considered a PRA upgrade requiring a focused-scope peer review.

   v. If the addition of RCP seal model qualifies as a PRA upgrade, provide the results from the focused-scope peer review including the associated F&Os and their resolutions.

SNC Response to NRC RAI 05

a. Westinghouse Generation III Shut Down Seals (Gen. III SDSs), model 93A, are installed in all RCPs in Farley Unit 1 and 2. Farley internal events including internal flood (IEIF) and internal fire PRA (FPRA) models credit the SDSs.

b. Credit for the SDS is included, and

i. The limitations and conditions in the NRC safety evaluation for PWROG-14001-P, Revision 1, are met as follows for the internal events with internal flooding PRA model and the fire PRA model.

The limitations and conditions were addressed as follows:

1. Limitation/Condition 1 is not applicable to this application, as the seals have already been installed.

2. A condition resulting in cold leg temperature exceeding the PWROG-14001-P limiting value was included in the models as a failure mode of the SDS, and the impact of asymmetric cooling have been explicitly modeled.

3. SNC may choose to perform Bayesian updating once the appropriate testing and data analysis has taken place.

4. The SDS bypass failure mode has been explicitly modeled with the appropriate failure probabilities.

5. Limitation/Condition 5 has been addressed by modeling plant-specific operator actions with corresponding human error probabilities in the models as described in PWROG-14001-P, Rev. 1.

6. Related to future testing/monitoring.

7. Related to future testing/monitoring.

8. Related to future testing/monitoring.

9. Related to future testing/monitoring.

10. Not applicable to this application.

ii. SNC takes no exceptions to the limitations and conditions.

iii. Incorporation of the Generation III Westinghouse RCP seal model has not been peer-reviewed as part of the IEIF and FPRA peer reviews.

iv. Incorporation of the RCP shutdown seals into the Farley IEIF and FPRA models is PRA maintenance as defined in ASME/ANS RA-Sa-2009 and qualified by RG 1.200, Revision 2.

The peer-reviewed IEIF and FPRA models did not include the Westinghouse Generation III low-leakage SDSs. However, the peer-reviewed PRA models did include an RCP seal leakage model (WOG 2000) to assess the plant response to events that result from a total loss of cooling to the RCP seals. Implementation of the new low-leakage RCP seal model into the IEIF and FPRA was performed consistent with the PRA method, modeling, and framework that had already been peer-reviewed.

ASME/ANS RA-Sa-2009 defines a PRA upgrade as a new methodology, or a change in scope or change in capability, that impacts the significant accident sequences or the significant accident progression sequences. PRA maintenance is defined as changes within the framework of an existing model structure. The change in the seal leakage model is not a new methodology because the new seal leakage model is simply an expansion of the current peer-reviewed model with different failure probabilities and associated human actions. There is no change in the model scope because the equipment, dependencies, and types of accident sequences remain the same. Finally, there is no change in PRA modeling capability, i.e., the peer reviewed PRA model can still evaluate the risk associated with station blackout and total loss of cooling events related to RCP seal failures. Therefore, implementation of the new seal leakage model is a change implemented within the framework of the existing peer-reviewed PRA model structure.

The seal leakage model change is only a change in the expected seal leakages associated with the new seals. The framework of the model remains essentially the same, and the High Level and Supporting Requirements (HLRs) in the PRA Standard for the Technical Elements associated with RCP seal modeling (e.g., those within the Accident Sequence Analysis, Data Analysis, Human Reliability Analysis, and Quantification technical elements) will continue to be Met or Not Met regardless of implementation of the change from the WOG2000 RCP seal model to the shutdown seal model. The five HLRs of interest associated with quantification (HLR-QU-A, -B, -C, -D and -E) continue to be met. Although the lower seal failure rates affect the ordering of the associated accident sequences and reduce CDF and LERF overall, the associated sequences were not significantly changed and new sequences that were not already modeled in the PRA and peer-reviewed were not generated. Consequently, this change in the internal events PRA does not constitute a PRA upgrade and does not require a focused scope peer review.

Generation III RCP seal model is not considered a PRA upgrade as defined in the ASME/ANS PRA standard because the differences between WOG2000 RCP seal model and Generation III RCP SDS PRA model are not methodology changes.

v. A focused-scope peer review has not been performed.

<u>NRC RAI 06 – Supplanted and Updated Fire PRA Guidance</u>

Since the safety evaluation was issued to Farley for implementation of its National Fire Protection Association (NFPA) 805 program (ADAMS Accession No. ML14308A048), NRC has issued updated guidance for aspects of fire PRA that supplant earlier guidance issued by NRC. Recently, as part of NRC's review of SNC's request to revise the Farley integrated leak rate test (ILRT) program, NRC staff requested information about the impact of such guidance on the ILRT application in a letter dated March 15, 2017 (ADAMS Accession No. ML17058A113). Specifically, NRC requested information about the impact of the following NRC fire PRA guidance documents.

- NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE)" (ADAMS Accession No. ML16343A058) regarding the updated approach to credit incipient fire detections systems;

- NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database" (ADAMS Accession No. ML15016A069) regarding changes in fire ignition frequencies and non-suppression probabilities;

- NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," Volume 2, "Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure" (ADAMS Accession No. ML14141A129) regarding possible increases in spurious operation probabilities.

In response to NRC's request, SNC provided the results of a sensitivity study in a letter dated October 12, 2017 (ADAMS Accession No. ML17285B308) demonstrating that the aggregate impact from applying the new guidance was an increase in total core damage frequency (CDF) of about 32 percent for Unit 1 and an increase in CDF of about 19 percent for Unit 2. LAR Enclosure 4 shows that fire risk is the dominant risk contribution to total risk by a significant margin (e.g., the fire CDF for Unit 1 is about a ten times greater than the internal events CDF for unit 1). Accordingly, the sensitivity study results demonstrate that the aggregate impact of using the updated NRC guidance cited above could impact the RICT calculations.

In light of the observations above and given the impact that the new fire PRA guidance can have on the application, provide the following:

a. Clarify whether or not guidance from the cited guidance documents (i.e., NUREG-2180, NUREG-2169, and NUREG/CR-7150) has been incorporated into the Farley FPRA that will be used to support the RICT program.

b. If the guidance from the guidance documents cited above (i.e., NUREG-2180, NUREG-2169, and NUREG/CR-7150) has not been incorporated into the fire PRA, then propose a mechanism that ensures that the guidance from NUREG-2180, NUREG-2169, and NUREG/CR-7150 is incorporated into the fire PRA prior to implementation of the RICT program.

c. If SNC proposes not to use methodology in the documents cited above (i.e., NUREG-2180, NUREG-2169, and NUREG/CR-7150) please provide

    i. A description of the proposed methodology (e.g., approach, methods, data, and assumptions) that will be used in the fire PRA.

    ii. Justification of the proposed methodology including comparison with available experimental results.

    iii. An estimate of the current CDF and [large early release frequency] ERF for each quantified hazard with fire PRA results: (1) that would be obtained had the guidance in the cited documents been applied, and (2) obtained using the proposed methodology.

    iv. If the current CDF and LERF estimates do not satisfy the limitations and conditions in Section 4, item 6 of the NEI 06-09 safety evaluation, explain how these guidelines will be met before implementation of the RICT program.

    v. An evaluation on how using the proposed methodology instead of the cited methodology could impact the RICT estimates.

SNC Response to NRC RAI 06

The newly issued NRC guidance documents (i.e., NUREG-2180, NUREG-2169, and NUREG/CR-7150) have been incorporated into the Farley fire PRA that will be used to support the RICT program.

NRC RAI 07 – Sources of Uncertainty – Seasonal Variations

Enclosure 7 of the LAR, which identifies sources of PRA modeling uncertainty, states that no internal events or FPRA modeling uncertainties were identified that require a sensitivity study as part of the RICT program calculations. However, NRC staff identified a source of uncertainty that appeared to have the potential to impact the application. LAR Enclosure 7, Section 2 states that the baseline PRA does not account for seasonal variations caused by external hazards even though "*certain initiating events can be affected by seasonal variations.*" It further states:

> *The RICT Program will include a qualitative consideration of weather events as part of the RMA decision process when [Limiting Condition for Operation] LCO 3.8.1 [AC Sources] CTs [Completion Times] are extended to address this source of uncertainty.*

Section 2.3.4 of NEI 06-09, Revision 0-A, states, in part,

> *"If the PRA model is constructed using data points or basic events that change as a result of time of year or time of cycle (examples include moderator temperature coefficient, summer versus winter alignments for HVAC, seasonal alignments for service water), then the RICT calculation shall either (1) use the more conservative assumption at all time, or (2) be adjusted appropriately to reflect the current (e.g., seasonal or time of cycle) configuration for the feature as modeled in the PRA. Otherwise, time-averaged data may be used in establishing the RICT."*

It is unclear how the qualitative consideration of weather events will be used as part of the RMA decision process. Additionally, the NRC notes that it is possible for other initiating events (e.g., loss of Plant Service Water) to also be impacted by seasonal variations in external hazards. Therefore:

a. Describe how the RMA process will use qualitative considerations of weather events when LCO 3.8.1 completion times are extended to address uncertainties associated with seasonal variations in external hazards. Include in this description whether the qualitative consideration will be used to adjust initiating event frequencies based on seasonal variations and how the proposed use of qualitative considerations of weather events is consistent with the guidance in NEI 06-09, Revision 0-A.

b. Explain and justify whether initiating events other than loss of offsite power included in the [Configuration Risk Management Program] CRMP modeling can be impacted by seasonal variations in external hazards.

c. If initiating events other than loss of offsite power used in the CRMP model can be impacted by seasonal variations in external hazards, then identify these initiating events and justify why seasonal variation in initiating event frequency cannot impact the RICT calculations.

d. As an alternative to part (c), identify initiating events whose event frequencies can be

impacted by seasonal variations in external hazards and the LCOs that can be impacted and confirm that the RICT program will qualitatively consider those events during the Risk Management Action (RMA) development process when the completion time of impacted LCOs are extended.

SNC Response to NRC RAI 07

Weather related impacts on internal event initiators include such occurrences as severe thunderstorms, tornadoes and hurricanes. They vary by time of year. The initiating event frequencies in the base PRA model use time averaged data. In the event of severe weather, site procedures instruct operators to initiate actions to restore systems to service as soon as possible if any major plant systems are out of services or being tested for other contingencies (i.e. secure all items which may become missiles during high winds, suspend switchyard activities, ensure diesel generators are aligned for auto start, maintain plant storage tank levels as high as possible).

The current maintenance rule online risk process accounts for actual severe weather impacts on initiators such as reactor trip and LOOP by adjusting the frequency within the configuration risk tool (Phoenix). The increases are defined by site specific section in the fleet procedures. If the Severe Weather Abnormal Operating Procedure is in effect due to forecasted wind speed, tornado or tornado warning, the procedure directs the operator to increase the Plant Centered Dual Unit LOOP initiator. If severe weather cause transmission system instability, the procedure directs the operator to increase the weather-related Dual Unit LOOP initiator.

This is not specifically a seasonal adjustment. It is based on the likelihood of the weather actually impacting the site, as would arise through such events as a National Weather Service tornado warning. The same tool will be used for RICT calculations; thus, the increased risk of a weather-related initiator would be captured in the RICT calculation. Additionally, the RMA process will include consideration of weather impact on the plant risk.

There are no other initiators that are subject to seasonal variations other than loss of offsite power. The seasonal variations in NEI 06-09 generally relate to changes in success criteria based on parameters such as summer river temperature or outside air temperature. Such variations are accounted for in the Farley PRA by modeling the limiting success criteria requirements. These success criteria variations are therefore directly addressed in the RICT calculation.

NRC RAI 08 – PRA Modeling Conservatisms That May Result in Underestimation of a RICT

The NRC SE of NEI 06-09, Revision 0-A states:

> *When key assumptions introduce a source of uncertainty to the risk calculations (identified in accordance with the requirements of the ASME standard), TR NEI 06-09, Revision 0, requires analysis of the assumptions and accounting for their impact to the RMTS calculated RICTs.*

Enclosure 7 of the LAR states that no IEPRA or FPRA modeling assumptions or uncertainties were identified that require a sensitivity study as part of the RICT program calculations. However, NRC staff notes that a couple of assumptions identified in LAR Enclosure 7 appear to have the potential to impact the RICT calculations for certain SSCs. Though these assumptions

are described as conservative assumptions, NRC staff notes that conservatism in PRA modeling could have a non-conservative impact on the RICT calculations for certain SSCs. If a structures, systems and components (SSC) is part of system not credited in the FPRA or it is supported by a system that assumed to always fail, then the risk increase due to taking that SSC out of service could be masked by the conservative modeling.

a. An entry in LAR Table E7.3 indicates that "some systems are not credited" in the FPRA by treating them as always failed because the associated SSC cables were not traced. This conservative modeling can mask the risk associated with taking certain SSCs out of service and impact on the calculated RICT. It appears to NRC staff that this uncertainty may require sensitivity studies to support the RICT calculations. In light of these observations, address the following:

    i. If the RICT program includes SSCs that are part of system or are supported by a system that is not credited in the FPRA, then justify that the uncredited systems have an inconsequential impact on the RICT calculations.

    ii. Alternatively to item ii above, propose a mechanism to ensure that a sensitivity is performed for the RICT calculations for applicable SSCs to determine the impact of the conservative modeling on the RICT estimates. The proposed mechanism should also ensure that any additional risk associated with the modeling is either accounted for in the RICT calculation or is compensated for using additional RMAs during the RICT.

b. LAR Table E7.1, associated with sources of modeling uncertainty for the IEPRA, states that credit for battery life was limited to "two hours based on conservative FSAR [final safety analysis report] analysis." The disposition for this source of uncertainty states that this uncertainty is "unlikely to be an issue for delta risk applications" citing possible manual actions that could be taken if DC power is lost. However, it is not clear to NRC staff that this assumption, which excludes credit for proceduralized actions that would extend battery life, does not impact the application. This conservative modeling can mask the risk associated with taking certain SSCs out of service (e.g., a DC electrical power subsystem) and entry into previously unanalyzed procedures. All of which have a potential impact on the calculated RICT. It appears to NRC staff that this uncertainty may require sensitivity studies to support the RICT calculations. Therefore address the following:

    i. Justify that the exclusion of credit for actions that would extend battery life has an inconsequential impact on the RICT estimates, including RICTs calculated for LCOs related to the operability of DC electrical power.

    ii. If, in response to part (i) above, it cannot be justified that the modeling of battery life has an inconsequential impact on the RICT estimates, then propose a mechanism to ensure that a sensitivity is performed as part of the RICT calculations associated with those impacted LCOs to determine the impact of the uncertainty on the RICT. The proposed mechanism should also ensure that any additional risk associated with the modeling uncertainty is accounted for in the RICT calculation, or that additional RMAs are applied during the RICT.

SNC Response to NRC RAI 08

As part of the Fire PRA development, there are components that are included in the basic event mapping process but are not included in the circuit analysis task. The basis for this is that these components are either not credited in the Fire PRA or they are of sufficiently low risk such that the overall change in risk is not impacted by the crediting of these components. These components make up a list referred to as “UNL” or unknown location. Since the cables for these components are not identified, and therefore not included in the analysis, these components are assumed to be failed for all scenarios (unless it can be determined that the failures can be excluded, referred to as credit by exclusion).

Of those components that are assumed always to be failed in the FPRA, the containment isolation valves are associated with a RICT scoped system or its support systems; additionally, some components are associated with the cues for operator actions that are credited in the PRA for mitigation following equipment failure. Some of these actions are associated with support systems for RICT systems but do not support normal operability of the systems. These components are included in the Fire PRA model but are treated as failed in Fire PRA quantifications; thus, they are part of the RICT evaluation.

Some of these components in the UNL list are associated with categories selected to obtain insights from UNL sensitivities.

- Containment Isolation Valves
- Containment Vent Outlet Valve 3556
- Train A and B Annunciator for Containment Sump Pump Room Cooling

A sensitivity originally performed for previous Farley Fire PRA model update by not failing these UNL components showed minimal impact on CDF (less than 2%) and an approximately 20-30% reduction in LERF. This provided the basis for selecting the sensitivity categories noted above.

Table RAI 08-1 documents updated sensitivity cases for the credit of the entire UNL list and the specific UNL categories identified above based on the current Farley FPRA model. These cases were run to examine the impact of specific UNL component categories that may be associated with TS for which RICTs may be applied.

Table RAI 08-1

Case U1_CDF_E-11	Nominal Freq.	Resultant CDF	Delta CDF	% Change
Containment Isolation Valves	8.22E-05	8.22E-05	0.00E+00	0.00%
Containment Vent Outlet Valve 3556	8.22E-05	8.22E-05	0.00E+00	0.00%
Train A and B Annunciator for Containment Sump Pump Room Cooling	8.22E-05	8.22E-05	0.00E+00	0.00%
All UNL Credited	8.22E-05	8.28E-05	6.00E-07	0.72%
Case U1_LERF_E-11	**Nominal Freq.**	**Resultant CDF**	**Delta CDF**	**% Change**
Containment Isolation Valves	3.84E-06	3.81E-06	-3.00E-08	-0.79%
Containment Vent Outlet Valve 3556	3.84E-06	3.64E-06	-2.00E-07	-5.49%
Train A and B Annunciator for Containment Sump Pump Room Cooling	3.84E-06	3.84E-06	0.00E+00	0.00%
All UNL Credited	3.84E-06	3.64E-06	-2.00E-07	-5.49%

The CDF impact from the “All UNL Credited” sensitivity case confirms that there would be no significant impact on calculated RICTs as a result of not crediting these components in the Fire PRA. The LERF impact in the “All UNL Credited” sensitivity shows that there is a small sensitivity, which might have a small affect only on certain RICT calculations. The containment isolation valves listed are primarily in small lines and require multiple other component failures for a release to occur.

The one category that showed the most significant decrease in LERF was the containment vent outlet valve (3556). This valve is in series with two containment vent isolation MOVs. These two MOVs (3530 and 3740) are credited in the Fire PRA and have the appropriate hot short probabilities applied. With the assumed failure of the 3556 valve, and the fire induced failure of the 3530 and 3740 valves, the resulting impact is a failure of containment penetration 103. This containment penetration failure is specific to the Fire PRA. As shown in Table RAI 08-1, the LERF UNL impact is nearly entirely associated with the assumed failure of this valve on the UNL.

The overall LERF impact in these sensitivities is an overestimation, because if the cabling routing were known, the cables and components would be damaged in individual fire scenarios, thereby reducing the impact indicated by this sensitivity. From the perspective of the equipment subject to RICT, this sensitivity overstates the possible impact because most of the UNL equipment is not related to such equipment, and the overall sensitivity is due to the assumed failure of the 3556 valve.

Per Table E1.2 in the LAR, the RICTs most sensitive to LERF are the containment isolation TS. The sensitivity above for the UNL containment isolation valves shows very little change in the LERF value, with the exception of the 3556 MOV. This shows that the UNL uncertainty would have little impact on the calculated RICT value for the containment isolation actions.

The impact regarding battery lifetime is mitigated in part in the PRA model by requiring the battery charger for DC demands beyond the assumed two-hour battery life (for other than station blackout scenarios). The impact of the battery lifetime is further constrained by the fact that the turbine driven auxiliary feedwater pump relies on instrument air operated steam admission valves. The steam admission valves fail close on a loss of air; however, an air reservoir is provided that will hold these valves open for a nominal two hours. Site emergency operating procedures direct the operators to take manual control of turbine driven auxiliary feedwater pump during this scenario. This is modeled in the PRA.

Given the reliance on the battery charger for longer term DC loads, a battery charger RICT would be shorter than if a longer battery lifetime were credited. On a component level, the battery failure probabilities in the PRA are based on the full 24-hour PRA mission time. For these reasons, the two-hour battery lifetime is expected to have an inconsequential impact on the application.

NRC RAI 09 – Sources of Model Uncertainty and Parametric Uncertainty Methodology

Limitation and Condition 10 in the NRC staff's SE on NEI 06-09 states:

> *The LAR will provide a discussion of how the key assumptions and sources of uncertainty were identified, and how their impact on the RMTS was assessed and dispositioned.*

Additionally, Section C of RG 1.174, Revision 3, states, in part:

> *In implementing risk-informed decision-making, LB [licensing basis] changes are expected to meet a set of key principles. … In implementing these principles, the staff expects [that]: … Uncertainty receives appropriate consideration in the analyses and interpretation of findings. … NUREG-1855 provides acceptable guidance for the treatment of uncertainties in risk-informed decision-making*

Enclosure 7 of the LAR states that the IEPRA uncertainty analysis was performed based on guidance from NUREG-1855 and cites Revision 0 in the reference section of Enclosure 7, Section 2.0. Revision 0 of NUREG-1855 (2009) primarily addressed sources of model uncertainty for internal events (including internal flood). Revision 0 of NUREG-1855 references EPRI report 1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments", which, among other guidance, provides a generic list of sources of model uncertainty and related assumptions for internal events. Revision 1 of NUREG-1855 (ADAMS Accession No. ML17062A466) further clarifies the NRC staff decision-making process in addressing uncertainties and addresses all hazard groups (e.g., internal events, internal flood, internal fire, seismic, low-power and shutdown, Level 2). NUREG-1855, Revision 1, cites use of EPRI 1016737 and EPRI 1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty", as companion documents to the NUREG-1855 that provide generic lists of sources of model uncertainty for internal events, internal flood, internal fire, and other hazard groups.
NRC staff notes that given the total CDF and LERF results presented in LAR Enclosure 4, there is a small margin between the total CDFs for Farley Units 1 and the RG 1.174 risk acceptance guidelines, and therefore key uncertainties and assumptions could cause the baseline PRA results to challenge the RG 1.174, Revision 3, risk acceptance guidelines.

a. Describe the process used to identify the LAR sources of model uncertainty and related assumptions, including generic and plant-specific sources, in the Farley baseline IEPRA, IFPRA, and FPRAs and that were evaluated for their potential impact on this application. Include in this discussion an explanation of how the process is consistent with guidance in NUREG-1855 and the complementary EPRI documents 1016737 and 1026511, or other NRC-accepted methods.

b. If an updated evaluation of key assumptions and sources of uncertainties was performed since the LAR submittal, provide any new sources of model uncertainty and related assumptions that were not provided in the LAR and provide disposition of impact to the application.

c. For those sources of model uncertainty and related assumptions that could impact the application:

   i. Provide qualitative or quantitative justification that these key uncertainties and assumptions do not cause the baseline PRA results to challenge the RG 1.174, Revision 3, risk acceptance guidelines, collectively or individually. (NRC staff notes that given the total CDF and LERF results presented in LAR Enclosure 4, there is a small margin between the total CDFs for Farley Units 1 and the RG 1.174 risk acceptance guidelines.)

   ii. Provide qualitative or quantitative justification that these key uncertainties and assumptions have no impact on the RICT calculations.

   iii. Alternatively to item ii above, propose a mechanism to ensure that sensitivity studies are performed when a RICT is evaluated to assess the impact of the uncertainty on the RICT. As part of the proposal, ensure that the additional risk is accounted for in the RICT calculation or that additional RMAs are applied during the RICT.

d. Based on RG 1.174, Revision 3, and Section 6.4 of NUREG-1855, Revision 1, for a CC II risk evaluation, the mean values of the risk metrics (total and incremental values) need to be compared against the risk acceptance guidelines. The mean values referred to are the means of the risk metric's probability distributions that result from the propagation of the uncertainties on the PRA input parameters and those model uncertainties explicitly represented in the model. In general, the point estimate CDF and large early release frequency (LERF) values obtained by quantification of the cutset probabilities using mean values for each basic event probability does not produce a true mean of the CDF/LERF. Under certain circumstances, a formal propagation of uncertainty may not be required if it can be demonstrated that the state-of-knowledge correlation (SOKC) is unimportant (i.e., the risk results are well below the acceptance guidelines).

   Confirm that parametric uncertainty evaluations that consider the SOKC have been performed for the IEPRA, IFPRA and FPRA models and the propagated mean total CDF and LERF values were confirmed to meet the RG 1.174 Revision 3 guidelines.

<u>SNC Response to NRC RAI 09</u>

a. The process used to identify PRA model uncertainties and their impact is described in the Farley IEIFPRA uncertainty documentation. In the uncertainty documentation, EPRI 1016737 was used to provide guidance for a structured process for addressing uncertainties in PRA results in the context of risk-informed decision-making. Appendix A of EPRI 1016737

is used as a template to document plant-specific issue characterization and assessments to fully satisfy the related supporting requirements.

Although the uncertainty documentation does not explicitly refer to NUREG-1855 Revision 1, the uncertainty evaluation process and sources in EPRI 1016737 are addressed. The information in LAR table E7.1 was distilled from the IEIFPRA uncertainty documentation. The evaluation documented in the uncertainty documentation were updated to add additional discussion of the application of the EPRI 1016737 approach, to add a table that documents an expanded search for potential sources of internal events and internal flooding model uncertainty via a keyword search of the PRA documentation, and to determine if additional sources of uncertainty exist that might warrant sensitivity analyses beyond those previously identified. The update was done as part of the Appendix X F&O closure process which was prepared after submittal of the 4b LAR.

The parametric uncertainty analysis for the Farley IEIFPRA is documented in the quantification/model documentation.

The process used to identify uncertainties and their impact is described for the fire PRA in the Farley Fire PRA quantification/summary documentation. The evaluation examines sources of uncertainty for each of the Fire PRA development tasks defined in NUREG/CR-6850. Although this evaluation does not specifically refer to EPRI 1026511 and therefore does not specifically address each of the Fire PRA entries in Appendix B of that report, the considerations in the fire PRA quantification/summary documentation are consistent with EPRI 1026511 Appendix B and represent an appropriate set of potential sources of uncertainty for this application. The evaluation of sources of uncertainty in the fire PRA as summarized in LAR Table E7.3 was based on the assessment in the quantification/summary documentation. The assessment in the quantification/summary documentation was updated to added detail in response to peer review F&Os but is otherwise consistent with the previous version that was the basis for the discussion in the LAR.

The parametric uncertainty analysis for the Farley Fire PRA is documented in the quantification/summary documentation.

b. The update to the Farley IEIFPRA uncertainty documentation included a table that documents an expanded search for potential sources of internal events and internal flooding model uncertainty via a keyword search of the PRA documentation. Table RAI 09-1 originates from the table in the uncertainty documentation and includes a disposition of impact for the RICT application.

Table RAI 09-1

Assumption/Source of Uncertainty (From Uncertainty Documentation)	Model Sensitivity and Disposition
Room cooling failures are relatively small contributors to bus failure, even assuming no credit for operator response to a room cooling failure.	GOTHIC heat-up analysis determined which electrical equipment, including AC & DC Bus/Switchgears, are dependent upon successful operation of the associated room coolers during the 24-hour mission time modeled in the PRA. The failure of these room coolers were modeled in the applicable system fault trees. Therefore, this is not a key source of uncertainty and will not be an issue for RICT calculations.
Three pre-initiator HFEs were developed in detail. The remaining pre-initiator HFEs were mapped to these representative HFEs.	The Farley Human Reliability Analysis use the typical industry practice in performing detail analysis on an HFE and mapping other HFEs together where procedures are similar to the detail analysis. The mapping was determined based on up-to-date testing/operating procedures. This is not considered a significant source of uncertainty. Therefore, this is not a key source of uncertainty and will not be an issue for RICT calculations.
The HRA is based on the Unit 1 PRA model, which is applicable to Unit 2 as well.	Differences among the units that could relate to the HRA are those that stem from differences in specific system differences, configuration differences, procedures and training differences. The configuration and system differences are accounted for in the development of system fault trees. As part of this development of the system fault trees, a comparison between units were done. There were no significant differences between the units that would affect the HRA. Configuration/system differences are noted in shared-unit procedures and in training material and would have been captured in the HRA analysis. Therefore, this is not a key source of uncertainty and will not be an issue for RICT calculations.
It is assumed that the switch is not placed in the OFF position after performance of the FNP-0-SOP-42.0 processes, which makes the restoration step equivalent to those in the STPs. This is conservative in that it would lead to a pre-initiator HFE.	This assumption is the bases of the pre-initiator HFE. A routine maintenance activity that would render the equipment unavailable. Therefore, this is not a key source of uncertainty and will not be an issue for RICT calculations. If the importance of the HFE were raised during a RICT calculation, RMAs would be developed for the HFE.

Table RAI 09-1

Assumption/Source of Uncertainty (From Uncertainty Documentation)	Model Sensitivity and Disposition
[For HFEs 0ACPMY52P504BG and 0ACXVSY52V520G] The period of performance is assumed to be 12 months. This is potentially non-conservative based on long test interval.	The 12-month period of performance used in the analysis is conservative. Plant scheduled maintenance is on an 18-month interval. This would result in a lower HEP valve for the pre-initiator. Therefore, this is not a key source of uncertainty and will not be an issue for RICT calculations
For Procedure 9.8, an independent checker is assumed.	Manipulation of any safety related system generally require independent verification. This is part of the standard and expectations for conduct of plant operators. Therefore, this is not a key source of uncertainty and will not be an issue for RICT calculations.
Timing. In contrast to pre-initiator Human Failure Events, post-initiator Human Failure Events are dynamic and subject to time variations that range from constrained to expansive. Timing is impacted by the time available and the time required.	The FNP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.
HEP Minimum Values, Dependency, and Recovery Credit. A floor value of 1.00E-5 was used for the minimum independent HEP and a value of 1.00E-06 for joint HEP combinations. The joint HEP combinations are typically based on dependency rules that follow THERP. THERP always assumes a negative dependency between failures. This is conservative in that many times a failure could provide additional cues and indications which could positively impact the second action. The THERP equations for dependency are stated in THERP to be conservative such as close in time was meant to be minutes and are based on engineering judgment. THERP dependency is also used for recovery within an HFE. Over-crediting recovery and using lower dependence and minimum HEP values can under-estimate risk. Under-crediting recovery can over-state the risk and lead to masking.	The FNP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.

Table RAI 09-1

Assumption/Source of Uncertainty (From Uncertainty Documentation)	Model Sensitivity and Disposition
The HRA is based on the Unit 1 PRA model, which is applicable to Unit 2 as well.	Differences among the units that could relate to the HRA are those that stem from differences in specific system differences, configuration differences, procedures and training differences. The configuration and system differences are accounted for in the development of system fault trees. As part of this development of the system fault trees, a comparison between units were done. There were no significant differences between the units that would affect the HRA. Configuration/system differences are noted in shared-unit procedures and in training material and would have been captured in the HRA analysis. Therefore, this is not a key source of uncertainty and will not be an issue for RICT calculations.
Many cases where operators are instructed to use SOP-x.x, Do in accordance with SOP-z.z, etc. If you read through the initial conditions they may or may not apply. How are the operators trained with respect to this?	The FNP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.
EEP-3 Step 15.4: PRA assumes that RCPs are running	The FNP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations. If the importance of the HFE were raised during a RICT calculation, RMAs would be developed for the HFE.
The baseline HRA assumes a shift change at 9 hours after the start of an abnormal event, but if the cues are within 9 hours of one another, the same crew may be present for both actions, so the separation of the cue times is considered.	The FNP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. This is a conservative assumption which is refined as appropriate, as described. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations. If the importance of the HFE were raised during a RICT calculation, RMAs would be developed for the HFE.

Table RAI 09-1

Assumption/Source of Uncertainty (From Uncertainty Documentation)	Model Sensitivity and Disposition
Same Time in HRA Dependency: Actions can be considered to occur at the same time if the cue for one action occurs before the previous action is expected to be completed. The HRAC specifies that the actions occur at the same time if: TdelayN < Tdelay(N-1) + T1/2(N-1) + Tm(N-1).	The FNP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations. If the importance of the HFE were raised during a RICT calculation, RMAs would be developed for the HFE.
This document contains the Analysis for post-initiator HFEs. All assumptions regarding individual analyses are contained and justified within that analysis.	The FNP PRA model is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations. If the importance of the HFE were raised during a RICT calculation, RMAs would be developed for the HFE.
For the DC switchgear rooms, no credit is taken for the normal auxiliary building HVAC system.	This is not an assumption. The ESF room coolers for DC switchgear rooms are designed to maintain air temperatures as required in rooms containing safety-related equipment during and after a design basis loss of coolant accident with a loss of offsite power. The normal auxiliary building HVAC system, during a loss of offsite power, is not powered by an emergency diesel generator. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.

Table RAI 09-1

Assumption/Source of Uncertainty (From Uncertainty Documentation)	Model Sensitivity and Disposition
Plugging of the SW strainers is not included in the Loss of Service Water initiating event fault tree logic models.	Plant experience shows that there has been no loss of SW induced by strainer plugging. The SW system is in continuous operation and the strainers are routinely (every 4 hours) backwashed by the operators. The differential pressure switch across each SW strainer inlet and outlet alarms on the SW structure alarm panel upon an increase in differential pressure. Any alarm on this panel causes a SW Structure Alarm annunciator in the main control room. However, the cause of the alarm can only be determined from the SW structure panel. The strainers are located downstream of the pumps. Therefore, any credible strainer plugging fault would likely fail the pumps first. In addition, the water in the wet pit has been screened in both the river water and service water intake screens. Therefore, strainer plugging would be expected to be a slow process, and if the strainers began to clog, the increased delta-P would be expected to be detected and corrected within a short period of time. However, since routine monitoring of the SWIS may be impacted following other initiating events, plugging of the strainers and failure of operator action to respond to the high differential pressure alarm is modeled in the mitigating system fault trees. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations. If the importance of the HFE were raised during a RICT calculation, RMAs would be developed for the HFE.
[Appendix A item #1] One-Time extension - assume we will seek permanent	The assumption to use a containment integrated leak rate test (ILRT) interval of 15 years was based on a one-time ILRT extension of 15 years. Then, we assumed the 15-year interval would become permanent. The current ILRT interval is now 15 years. Therefore, this does not represent a key source of uncertainty and will not be an issue for RICT calculations.

c. (i) The key assumptions and areas of uncertainty are those delineated in the LAR as noted in part a of this response and stated in part b of this response. As noted in the discussion in Enclosure 7 of the LAR, these sources are not sufficient to challenge the RG 1.174 risk acceptance guidelines, as most of the potential sources have a slightly conservative impact, i.e., are expected to over-estimate CDF or LERF.

(ii) The relevant uncertainties are those listed in the LAR. The discussions presented in LAR Enclosure 7 in Tables E7.1 and E7.3 and in part b of this response provide the bases for concluding these items will have no significant impact on the RICT calculations.

In addition, the Fire PRA quantification/summary documentation includes a discussion of sensitivities that were performed to evaluate the reasonableness of various modeling uncertainties in the FPRA model. As noted in the quantification/summary documentation, the FNP FPRA analysis is believed to represent a somewhat conservative estimation of fire risk, within the constraints of the requirements for a model acceptable for the NFPA-805 program.

d. The parametric uncertainty analyses for the IEIFPRA are documented in Farley PRA quantification/model documentation, and for the Fire PRA are documented in Fire PRA quantification/summary documentation. The parametric uncertainty analysis addresses the SOKC by the use of system level type codes for basic events. This applies the same variability of all components of that type within a system during the analysis. The parametric uncertainty analyses in those notebooks demonstrate that the point estimate mean values provide a close representation of the propagated mean values reflecting SOKC and the propagated mean total CDF and LERF values were confirmed to meet RG 1.174 Revision 3.

NRC RAI 10 – Total Risk Estimates Against RG 1.174 Guidelines

RG 1.174, Revision 3, provides the risk acceptance guidance for total CDF ($1x10^{-4}$ per year) and LERF ($1x10^{-5}$ per year). Enclosure 4 of the LAR shows the total CDF for Unit 1 to be $9.69x10^{-5}$ per year and for Unit 2 to be $9.22x10^{-5}$ per year, thus demonstrating a small margin between the total unit risk and the RG 1.174 risk acceptance guidelines. NRC staff notes that the F&O closure or the response to the preceding RAIs could involve updates to the IEPRA, IFPRA, or FPRA models. Therefore:

a. Provide updated CDF and LERF estimates and summarize any updates performed after the LAR submittal.

b. Demonstrate that after the IEPRA, IFPRA, and FPRA models are updated in response to the RAIs that the total risk for each unit is recalculated from the updated models and confirmed to still be in conformance with RG 1.174, Revision 3, risk acceptance guidance (i.e., a CDF and LERF less than $1x10^{-4}$ and $1x10^{-5}$ per year, respectively).

c. Alternatively, propose a mechanism ensuring that after the IEPRA, IFPRA, and FPRA models are updated in response to RAIs and prior to implementation of the RICT program, the total risk for each unit is recalculated from the updated models and confirmed to still be in conformance with risk acceptance guidance in RG 1.174, Revision 3.

SNC Response to NRC RAI 10

Table RAI 10-1 is an updated version of the table provided in Enclosure 4 of the LAR that reflects the most recent Unit 1 and Unit 2 CDF and LERF values that resulted from the most recent version of the IEIFPRA and Fire PRA. The values for CDF and LERF in Table RAI 10-1 for the Seismic bounding analysis have not changed from what was submitted in the LAR. After the LAR submittal, the IEIFPRA and Fire PRA went through the Appendix X F&O Closure process.

In response to the RAIs, change to the IEIFPRA and Fire PRA models are unnecessary. As mentioned in RAI 03 response, updates to the IEIFPRA due to F&O closures shall be incorporated into the Fire PRA as part of our normal PRA configuration control process prior to the implementation of the RICT program. By incorporating these changes in the Fire PRA, it is understood that total CDF and LERF remain in conformance with RG 1.174, Revision 3, risk acceptance guidance.

Table RAI 10-1

	Unit 1				Unit 2			
	CDF		LERF		CDF		LERF	
	LAR	Most Recent	LAR	Most Recent	LAR	Most Recent	LAR	Most Recent
IEIF	8.91E-06	8.40E-06	1.28E-07	6.89E-08	8.76E-06	8.85E-06	1.03E-07	6.88E-08
FIRE	8.35E-05	8.22E-05	4.21E-06	3.84E-06	7.89E-05	8.03E-05	4.51E-06	4.52E-06
SEIS	4.51E-06		2.07E-06		4.51E-06		2.07E-06	
Other	Screened Out				Screened Out			
TOTAL	9.69E-05	9.51E-05	6.41E-06	5.98E-06	9.22E-05	9.37E-05	6.68E-06	6.66E-06

NRC RAI 11 – Screening of External Hazards

Section 2.3.1, Item 7, of NEI 06-09, "*Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines*," Revision 0-A, states that the "impact of other external events risk shall be addressed in the RMTS program," and explains that one method to do this is by documenting prior to the RMTS program that external events that are not modeled in the probabilistic risk assessment (PRA) are not significant contributors to configuration risk. The SE for NEI 06-09, states that "[o]ther external events are also treated quantitatively, unless it is demonstrated that these risk sources are insignificant contributors to configuration-specific risk." Section 1.2.5 of RG 1.200, "*An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities*," Revision 2 (ADAMS Accession No. ML090410014), states that the contribution of many external events CDF and LERF can be screened out: "(1) if it meets the criteria in NRC's 1975 Standard Review Plan (SRP) or later revision; or (2) if it can be shown using a demonstrably conservative analysis that the mean value of the design-basis hazard used in the plant design is less than 10^{-5} per year and that the conditional core damage probability is less than 10^{-1}, given the occurrence of the design-basis-hazard event; or (3) if it can be shown using demonstrably conservative analysis that the CDF is less than 10^{-6} per year." The screening criteria listed in Section 1.2.5 of RG 1.200 are consistent with those in Section 6-2.3 of the 2009 American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard (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*."

External Flooding

Enclosure 3 of Attachment 2 to the LAR discusses the evaluation of external event challenges. Table E3.A2.1 of the same enclosure discusses the basis for screening the external flooding hazard from consideration for this application. The basis for screening the external flooding hazard includes the results documented in the licensee's flood hazard reevaluation report

(FHRR);(ADAMS Accession No. ML15294A530). In addition, the licensee states that the combined effects river flooding event is estimated to produce a maximum flood elevation that will not top the vehicle barrier system (VBS).

The licensee's basis for screening the external flooding hazard from consideration for this application seems to rely on compliance with the Current Design Basis (CDB) with respect to mitigation of the hazard, and the use of deterministic theoretical maximum values for flood elevations.

However, the licensee's basis does not address the frequency of exposure to flood hazards (including occurrences lower than the design basis) that may impinge upon structures, systems, and components (SSCs) and challenge plant safety, the impact of associated effects and the temporal characteristics of the event (e.g., the period of site inundation), and the reliability of flood protection features and human actions.

For example, the VBS appears to be an SSC whose ability to mitigate an external flooding event, based on its design and function, is not guaranteed.

Based on the FHRR, the licensee states in Table E3.A2.1 of Enclosure 3 of Attachment 2 to the LAR that the frequency of a local intense precipitation (LIP) event capable of producing flood magnitudes reported in the FHRR is estimated to be well below $1x10^{-6}$ per year without providing the basis for the frequency estimation. The FHRR does not provide any estimation of hazard frequency.

The staff notes that Section 6.2-3 of the 2009 ASME/ANS PRA Standard as endorsed in RG 1.200, Revision 2, discusses the importance of recognizing that the demonstratively conservative estimate of a mean value is not a point estimate because the mean frequency can fall above the 95 percentile of the distribution when uncertainties are large.

a. Provide justification using the criteria in Section 6.2-3 of ASME/ANS RA-Sa-2009 for screening the external flooding hazard from this application. The justification should include consideration of uncertainties in the determination of demonstrably conservative mean values as discussed in Section 6.2-3 of ASME/ANS RA-Sa-2009.

b. If the external flooding hazard cannot be screened out in item (a), discuss, using quantitative or qualitative assessments, how the risk from external flooding hazards, especially the LIP and combined events river flooding, will be considered in the risk-informed completion times (RICTs) that are impacted by those hazards. The discussion should include consideration of and, as applicable, the basis for the following factors:

    - The frequency of LIP and combined events river flooding hazards,
    - The impact of LIP and combined events river flooding on plant operation and structures including the ability to cope with upset conditions,
    - The reliability of flood protection measures, and
    - The reliability of operator actions.

c. If the external flooding hazard is screened out in item (a), discuss how it will be ensured that assumptions related to the availability and the functionality of flood protection features (e.g.,

VBS) that are credited for the screening remain valid during RICTs such that the external flooding hazard continues to have an insignificant impact on the configuration-specific risk.

High Wind and Tornados

Enclosure 3 of Attachment 2 to the LAR discusses the evaluation of external event challenges. Table E3.A2.1 of the same enclosure discusses the basis for screening the extreme winds and tornados (including generated missiles) from consideration for this application. The licensee's basis for screening relies on the design of SSCs and a detailed tornado missile risk analysis. The discussion further states that "the site is currently evaluating tornado missiles in response to Regulatory Issue Summary (RIS) 15-06 (ADAMS Accession No. ML15020A419) and that the results of that evaluation "will be reflected in the extreme winds and tornados screening evaluation."

The licensee states in Table E3.A2.1 of Enclosure 3 of Attachment 2 to the LAR that the frequency of missile damage to target groups is less than $7x10^{-7}$ per year per Unit. The staff notes that the licensee's site is located in NRC's tornado region I as shown in RG 1.76, Revision 1 (ADAMS Accession No. ML070360253). Additionally, the evaluation performed in response to RIS 15-06 is focused on identified non-compliances against the design basis tornado missile protection. The criteria in Section 6.2-3 of the 2009 ASME/ANS PRA Standard does not appear to have been considered in the evaluation for high winds and tornados (i.e., impacts other than tornado missile risk) in Table E3.A2.1 of Enclosure 3 of Attachment 2 to the LAR.

d. Provide justification using the criteria in Section 6.2-3 of ASME/ANS RA-Sa-2009 for screening high wind and tornados hazard (i.e., impacts other than tornado missile risk) from this application. The justification should include consideration of uncertainties in the determination of demonstrably conservative mean values as discussed in Section 6.2-3 of ASME/ANS RA-Sa-2009.

e. If the high winds and tornados hazard cannot be screened out in item (a), discuss, using quantitative or qualitative assessments, for how the risk from high wind and tornado hazards will be considered in the RICTs that are impacted by those hazards. The discussion should include consideration of, as applicable, the basis for the following factors:

    - The frequency of high winds and tornados,
    - The impact of high winds and tornados on plant operation and structures including the ability to cope with upset conditions, and
    - The reliability of operator actions.

f. Discuss the approach used to obtain the tornado missile damage frequency of $7x10^{-7}$ per year and the appropriateness of that risk analysis to support the screening of the tornado missile risk for the RICTs affected by tornado missile impact.

g. The statement "[r]esults of the [tornado missile protection] TMP evaluation will be reflected in the extreme winds and tornados screening evaluation" in Table E3.A2.1 of Enclosure 3 of Attachment 2 to the LAR implies future consideration of that evaluation for screening the tornado missile risk for the current application. Explain the intent of the cited statement and how results of future evaluations and staff decisions can be factored into the current application. If the TMP evaluation is expected to be used to support the screening for

tornado missile hazard for the current application, discuss the approach used for that evaluation, its appropriateness to support the screening of the tornado missile risk for the RICTs affected by tornado missile impact, and differences compared to the approach discussed in item (c).

SNC Response to NRC RAI 11

a. This response addresses the characterization of the external flooding hazard at Farley, the basis for screening non-controlling flood mechanisms and screening of the controlling mechanisms that required further analysis during the 50.54(f) Request for Information following the recommendation of the Near-Term Task Force (NTTF) 2.1 for flooding. The criteria used to screen the flood causing mechanisms can be found in the ASME/ANS 2009 PRA Standard, Part 6 EXT-B1.

BACKGROUND

The Farley Nuclear Plant has reevaluated its external flooding hazard in accordance with NTTF Rec. 2.1 and Nuclear Regulatory Commission (NRC)'s 10 CFR 50.54(f) Request for Information. This process required licensees to follow present-day guidance and methodologies that are not part of the Standard Review Plan (SRP) for siting new reactors. The process characterizes flooding mechanisms "probable maximum" events, which is defined as the flood that may be expected from the most severe combination of critical meteorological and hydrologic conditions that are reasonably possible. Therefore, the maximum events that are included in the flood hazard reevaluation reports (FHRR) represent a conservative analysis of the flood mechanisms parameters including uncertainties in the modeling techniques.

The FNP FHRR was submitted to NRC on October 21, 2015 (Docket No. 50-348 and 50-364) and the results are outlined in the Mitigating Strategies Flood Hazard Information (MSFHI) letter dated December 10, 2015 (ADAMS ML15343A418). Subsequently, changes were made to the local intense precipitation (LIP) model and analysis. These changes were performed for the Mitigating Strategies Assessment (MSA). The results of the updated LIP analysis are used as an input to the screening analysis.

FLOOD HAZARD CHARACTERIZATION

The results of the FHRR can be found in Table 5-1: Summary Comparison with Current Licensing Basis Flood Hazard of the FHRR. These results are also included in the MSFHI letter issued by NRC staff in the Interim Staff Response, Tables 1 and 2. The results are broken into three categories, not applicable to the site, reevaluated flood mechanisms is bounded by current licensing basis (CLB) and the reevaluated flood mechanism is not bounded by the CLB but the design is such that the plant can withstand the hazard without challenge to key SSCs.

<u>Flood Mechanisms – Not Applicable to Farley</u>

The following flood causing mechanisms were determined to not be applicable to Farley:

1. Storm Surge
2. Seiche
3. Tsunami
4. Ice Induced Flooding

5. Channel Migration of Diversion

These mechanisms screen from consideration based on EXT-B1 Criterion 3 of the ASME/ANS RA-Sb-2009 PRA Standard. These mechanisms were determined to not affect the site or any of its SSCs based on the location of the site. The full calculations, methodology and justification can be found in the FHRR.

Flood Mechanisms – Reevaluation Bounded by CLB

The flooding in streams and rivers mechanism was reevaluated and determined to produce a probable maximum flood (PMF) water surface elevation (WSE) of 141.5 ft. The plant's CLB requires Farley to be designed for a WSE of 144.2 ft (Table 5-1 of the FHRR). Therefore, the CLB protection level is higher than the reevaluated level and this mechanism screens from further consideration based on EXT-B1 Criterion 1. The design of the plant exceeds the reevaluated level using a conservative analysis with adequate margin.

FLOOD MECHANISMS – REEVALUATION NOT BOUNDED BY CLB

A total of three mechanisms were not bounded by the CLB in the FHRR. The following sections address the mechanisms that were required for further analysis in the 10CFR 50.54(f) RFI process (e.g. MSA and FE).

Local Intense Precipitation

The LIP mechanism produces localized flooding throughout the site. The full set of calculations to determine the depths of water against external walls and doors was documented in MSA supporting calculations. This document outlines the reevaluated flood depths and durations at key ingress points to the plant for structures that house safety-related SSCs. The Door IDs, Building, Finished Floor Elevation (FFE), Max Water Surface Elevation, Max Flooding Depth above FFE and Flooding Duration above FFE can be found in Table 8-3: LIP Predicted Flooding Results at the Main Doors.

Combined Effects Flooding

The combined effects flooding mechanism causes a maximum still water flood elevation of 154.4 ft at the FNP structures that contain key SSCs. This elevation is below the FFE of the plant which varies based on location. The minimum FFE is 154.48 ft. The maximum flood elevation including wave run-up is 158.5 ft along the Kontek Vehicle Barrier System (VBS) as reported in Table 5-1 of the FHRR. Where the critical fetch intersects the VBS the maximum flooding against the VBS is 3.46 ft and the VBS is 3.5 ft high. The FHRR also reports the time to reach the maximum SWE of 154.4 is 53 hours from the nominal level of 95 ft.

Failure of Dams and Onsite Water Control/Storage Structures

The maximum WSE calculated for the failure of onsite water control/storage structures is 154.4 ft. Thus, this mechanism is bounded by the combined effects flooding mechanism with an equal WSE but more limiting flood elevation with wave run-up at 158.5ft. This mechanism will be screened from further consideration based on EXT-B1 Criterion 1 and 4. The plant is designed to mitigate the effects of the hazard with a grade elevation of 154.5 ft and the definition of the hazard is contained within the combined effects flooding.

SCREENING OF FLOOD CAUSING MECHANISMS

This section will provide justification for screening LIP and Combined Effects Flooding mechanisms from further consideration as a significant risk contributor at FNP. Both mechanisms were addressed in a Focused Evaluated (FE) prepared following NEI 16-05 guidance and submitted to NRC staff for review on June 12, 2017. The NRC issued a Staff

Assessment on January 24, 2018 requiring no further action or information on FNP's response to the 50.54(f) RFI for external flooding (ADAMS ML17331A410).

Local Intense Precipitation Screening

The LIP mechanism was not bounded by the CLB at FNP and the revised FHRR calculated standing water against external doors around the plant. Ingress of water under the exterior doors was evaluated. The MSA contains the full list of analyzed doors, including exterior doors that have standing water above the thresholds. The calculation considers the time in which the water is above the thresholds, the amount of water possible for ingress and the volume available to collect water for the event duration. This depth of water was compared to the height of any SR SSCs in these buildings and it was found that there is enough storage volume to prevent any impacts to the SR SSCs during a LIP event.

The following doors are required to be in their normally closed position and therefore are credited in the screening of LIP.

Table A-1: Doors credited in LIP Screening

Door ID	Building	Reference Grid Element No.
D-405	Auxiliary Unit No. 1	68240
D-436		57800
D-441		67604
PA109		57034
D-2405	Auxiliary Unit No. 2	68248
D-2436		58211
D-2441		67644
D-468	Containment Unit No. 1	65121
D-2468	Containment Unit No. 2	65167
D-723	Diesel Generator Building	65111
D-724		65422
D-854	Service Water Intake Structure	111062
D-855		112075
T856		110965

Table A-2 summarizes the flooding depths calculated for buildings housing SR SSCs and the available physical margin (APM).

Table A-2: Maximum Flood Depths in Buildings and the APM

Building	Max. Flooding Depth	Available Physical Margin
	ft	ft
Auxiliary Unit No. 1	0.44	1.56
Auxiliary Unit No. 2	0.28	1.72
Diesel Generator Building	0.26	1

Complete details on assumptions, methods and inputs (AIMs) can be found in the MSA submittal to NRC:

> "The total flood water volume postulated to flow through DG Building doors yields a maximum flooding depth inside the Switchgear Rooms of 0.26 ft, which is below the 1 ft flood protection curbs protecting the switchgear required for operation of the diesel generators. In the vicinity of the DFOSTs flood water from LIP is not deep enough to challenge the tank lids.
> The total flood water volume postulated to flow through the seven AB doors for Units 1 & 2 results in a maximum flooding depth of 0.44 ft and 0.28 ft at the TDAFWP rooms of AB Units 1 and 2, respectively. The TDAFWP is positioned on a pedestal 2 ft above the floor protected by a 0.5 ft flood protection curb.
> Three doors were evaluated for the SWIS. Flow through doors D-854 and D-855 result in 0.43 ft of flooding at the 167 ft elevation of the SWIS. Flow through door T856 results in 0.77 ft of flooding in the SWIS CO2 Bottle Room.
> The Boron Injection FLEX Pumps and RCS Make-up Pumps at the 100 ft elevation and electrical components critical to FLEX strategies housed in the switchgear rooms at the 121 ft elevation and load center rooms at elevation 139 ft are not impacted by LIP flooding."

Therefore, LIP has been screened from further consideration (except as described in the response to 11.c) in this risk informed application. No impacts to risk-significant SSCs from water intrusion was calculated and there are no challenges to any safety related functions due to the LIP expected with the doors in Table A-1 in their normally closed position. Criterion 1 of SR EXT-B1 in Part 6 of the 2009 PRA Standard applies to the LIP mechanism, as the plant design bounds the effects caused by any flooding from a LIP with sufficient margin.

<u>Combined Effects Flooding Screening</u>

The Combined Effects Flooding mechanism was not bounded by the CLB after reevaluation. However, the maximum SWE calculated (reached in 53 hours) was 154.4 ft which is below the 154.5 ft grade of the power block. Therefore, there are no impacts identified to any SR SSCs from the SWE of the river flood.
Combining Effects from winds that are anticipated due to the storm yields periodic wave run-up that was calculated at a maximum elevation of 158.5 ft. This elevation would challenge the exterior doors to keep water out of buildings housing SR SSCs; however, the plant is surrounded by a Kontek vehicle barrier system (VBS) which was evaluated for its capability to stop wave propagation well before it reaches the power block. As discussed in the MSA supporting calculations, VBS is well suited for stopping wave propagation and its placement is far enough from the power block to not allow waves to increase the water surface elevation above the site grade of 154.5 ft.

The Kontek VBS is credited in the screening criteria of the combined effects flooding mechanism. The total calculated load from flooding at one 10 ft wall section of the VBS is 5,511 lbs and the design load is in excess of 1.5 million lbs, demonstrating adequate margin to withstand wave action due to winds.

Therefore, the Combined Effects Flooding mechanism has been screened from further consideration (except as described in the response to 11.c) in this risk informed application. No impacts to risk-significant SSCs or challenges to any safety related functions due to flood waters identified in all the cited sources. Criterion 1 of SR EXT-B1 in Part 6 of the 2009 PRA Standard applies to this mechanism as the plant design bounds the effects caused by any flooding from the river or its associated effects with sufficient margin and without any human actions necessary. In addition, the use of theoretical maximum values for the hazards ensures that damage potential from higher frequency, lower magnitude values are bounded. More frequent, lower magnitude flooding mechanism would not pose additional challenges to the plant design.

SUMMARY OF EXTERNAL FLOOD CAUSING MECHANISM SCREENING RESULTS

The following table summarizes the results of the preliminary screening evaluation for external flood causing mechanisms following the guidance in the 2009 PRA Standard Part 6 Requirement EXT-B1. All flooding mechanisms that are required to meet HLR-EXT-A are included below.

Table A-3 Flood Causing Mechanisms Criteria for Screening and Summary of Evaluation		
Flood Causing Mechanism	EXT-B1 Criteria Used for Screening	Summary of Evaluation
Seiche	Criterion 3	Not applicable based on-site location
Storm Surge	Criterion 3	Not applicable based on-site location
Tsunami	Criterion 3	Not applicable based on-site location
Ice Induced Flooding	Criterion 3	Not applicable based on-site location
Channel Migration or Diversion	Criterion 3	Not applicable
Failure of Dams and Onsite Water Control/Storage Structures	Criterion 1 Criterion 4	Maximum WSE of 154.4 ft which is equal to the Combined Effects Flooding Mechanisms maximum WSE and below plant grade with no impacts to SR SSC identified
Streams and Rivers	Criterion 1	Maximum WSE of 144.2 ft which is below plant grade with no impacts to SR SSC identified
Local Intense Precipitation	Criterion 1	Analyzed in the FE and in-leakage from doors calculated to not impact any SR SSCs. Doors credited in the screening analysis

Table A-3 Flood Causing Mechanisms Criteria for Screening and Summary of Evaluation		
Combined Effects Flooding	Criterion 1	Kontek VBS evaluated to stop wave propagation (158.5 ft along VBS) and is credited in the screening analysis. Results in maximum WSE of 154.4 ft which is below the grade of the power block. No impacts to SR SSC are identified due to flooding.

b. The flooding hazard screens as described in 11.a.

c. The Kontek VBS and the credited doors in Table A-1 are credited for screening of the external flooding hazard and the configuration of these SSCs will be considered during the process of RMA development for a RICT.

   The VBS is a system with a tightly-controlled configuration; evolutions where the configuration is changed from its normal configuration are infrequent and brief in nature. Therefore, the risk from such an evolution is small. It is conceivable that the VBS may be made unavailable during a RICT to conduct maintenance, move equipment, or otherwise safely exit the LCO. However, the RMA process will address the configuration of the VBS to ensure awareness of its role in preventing external flooding damage and to minimize the unavailability.

   The doors listed in Table A-1 are normally closed doors and their importance in preventing external flooding damage will likewise be addressed in the RMA process. In the circumstance where such a door needs to be opened to conduct maintenance, move equipment, or otherwise safely exit the LCO, the RMA process will ensure the unavailability of the door to prevent damage from the external flooding hazard is minimized, thereby reducing the risk.

d. The screening criteria from the original LAR [B5] were:

   "As noted in Regulatory Guide 1.200, the fundamental criteria that have been recognized for screening-out events are the following: an event can be screened out if either (1) it meets the criteria in the NRC's 1975 Standard Review Plan (SRP) or a later revision (Reference 5); or (2) if it can be shown using a demonstrably conservative analysis that the mean value of the design-basis hazard used in the plant design is less than 10^{-5}- per year and conditional core damage probability is less than 10^{-1}, given the occurrence of the design-basis-hazard event; or (3) if it can be shown using a demonstrably conservative analysis that the CDF is less than 1E-06 per year."

   The responses to the RAIs use the screening criteria in Part 6 of the ASME/ANS RA-Sa-2009 PRA Standard, specifically the progressive screening defined in HLR-EXT-B and HLR-EXT-C.

   All above ground Category I structures required for safe shutdown are designed to withstand tornado loadings. Based on Section 3.3.2.1 of the FNP FSAR, the design includes (1) dynamic wind pressure associated with a tornado having a wind velocity of 300 mph and translational velocity of 60 mph, and (2) 3 psi pressure differential.

   Per Table 6-1 of NUREG/CR-4461, Rev. 2, the 10^{-7} annual probability tornado wind speed is 283 mph (i.e., less than the 300 mph design) for Joseph M. Farley, based on the more conservative F-scale; using the more recent EF-scale the 10^{-7} annual probability tornado

wind speed is even lower (217 mph). The tornado wind hazard analyses done in NUREG/CR-4461 include the consideration of uncertainties. Based on the plant design for wind pressure and the low frequency (<10^{-7}/yr) of design tornadoes, a demonstrably conservative estimate of CDF associated with tornado hazards other than tornado missile is much less than 10^{-6}/yr. Therefore, this hazard can be screened from consideration based on EXT-C1 Criterion C of ASME/ANS RA-Sa-2009. Tornado wind speeds are bounding for straight and, since FNP is not on the coast, hurricane winds. Therefore, all other non-missile high wind hazards can be screened based on EXT-B1, Criterion 1.

e. As described in response to item (d), high wind and tornado hazards are screened for impacts other than tornado missiles. Therefore, no further discussion is required.

f. The tornado missile damage frequency of 7×10^{-7}/yr described in the LAR is based on a TORMIS analysis of FNP and is documented in the FSAR, Section 3.5.1.2.

The TORMIS analysis results are provided in Table B-1.

Table B-1
FNP TORMIS Target Damage Probabilities [B4]

Target	TORMIS Annual Damage Probability	
	Unit 1	Unit 2
SRV's: Unit 1 Group 1	1.03E-08	
SRV's: Unit 1 Group 2	1.22E-08	
SRV's: Unit 1 Group 3	1.14E-08	
ARV: Unit 1 Group 1	6.82E-08	
ARV: Unit 1 Group 2	2.67E-08	
ARV: Unit 1 Group 3	6.07E-08	
SRV's: Unit 2 Group 1		2.51E-08
SRV's: Unit 2 Group 2		3.33E-08
SRV's: Unit 2 Group 3		2.90E-08
ARV: Unit 2 Group 1		9.67E-08
ARV: Unit 2 Group 2		5.72E-08
ARV: Unit 2 Group 3		1.55E-07
Air Intake Filter - Diesel 2C	4.29E-08	
Air Intake Filter - Diesel 1B	5.08E-08	
Air Intake Filter - Diesel 2B		6.67E-08
Air Intake Filter - Diesel 1C	1.07E-07	1.07E-07
Air Intake Filter - Diesel 1-2A	4.91E-08	4.91E-08
Intake Silencer - Diesel 2C	9.31E-10	

Table B-1
FNP TORMIS Target Damage Probabilities [B4]

Target	TORMIS Annual Damage Probability	
	Unit 1	Unit 2
Intake Silencer - Diesel 1B	1.14E-09	
Intake Silencer - Diesel 2B		1.05E-09
Intake Silencer - Diesel 1C	1.28E-09	1.28E-09
Intake Silencer - Diesel 1-2A	8.92E-10	8.92E-10
Muffler - Diesel 2C	0.00E+00	
Muffler - Diesel 1B	0.00E+00	
Muffler - Diesel 2B		0.00E+00
Muffler - Diesel 1C	0.00E+00	0.00E+00
Muffler - Diesel 1-2A	0.00E+00	0.00E+00
Ventilator - Oil Storage Room - 2C	0.00E+00	
Ventilator - Oil Storage Room - 1B	4.01E-12	
Ventilator - Oil Storage Room - 2B		0.00E+00
Ventilator - Oil Storage Room - 1C	8.90E-11	8.90E-11
Ventilator - Oil Storage Room - 1-2A	1.38E-12	1.38E-12
Roof Ventilator - Gen Rm - Diesel 2C	5.19E-11	
Roof Ventilator - Gen Rm - Diesel 1B	5.60E-12	
Roof Ventilator - Gen Rm - Diesel 2B		6.95E-11
Roof Ventilator - Gen Rm - Diesel 1C	4.24E-10	4.24E-10
Roof Ventilator - Gen Rm - Diesel 1-2A	1.78E-10	1.78E-10
Roof Ventilator - SG Rm A	2.84E-10	
Roof Ventilator - SG Rm B		4.11E-10
Manhole Cover A1M54 & A2M53	4.14E-08	4.14E-08
Manhole Cover B1M52 & B2M51	2.61E-08	2.61E-08
Unit 1 RMWST: 6" Suction Pipes	2.91E-08	
Unit 2 RMWST: 6" Suction Pipes		2.45E-08
Unit 1 Safety Injection System Suction Line (HBC-42)	5.23E-10	
Unit 2 Safety Injection System Suction Line (HBC-42)		3.94E-09
Exhaust Pipe for EDG 2C	1.31E-11	

Table B-1
FNP TORMIS Target Damage Probabilities [B4]

Target	TORMIS Annual Damage Probability	
	Unit 1	Unit 2
Exhaust Pipe for EDG 1B	0.00E+00	
Exhaust Pipe for EDG 2B		0.00E+00
Exhaust Pipe for EDG 1C		5.88E-10
Exhaust Pipe for EDG 1-2A	0.00E+00	
TOTAL	**5.42E-07**	**7.20E-07**

Based on the results in Table B-1, the total damage frequency for the Unit 2 FNP TORMIS targets is approximately 7E-7/yr, and less for Unit 1. The main difference between the two units is the increased likelihood of damage to the Unit 2 SRVs and ARVs as compared to Unit 1, as determined by the TORMIS simulations.
Generic conservative assumptions in the TORMIS analysis result in conservative damage frequencies. Additionally, there are conservative assumptions with respect to specific targets that also affect the results. For example, the failure of the SRVs to provide adequate heat removal is assumed due to crimping of the exhaust stacks. However, only one of five exhaust stacks on a SG is required for adequate heat removal. Additionally, the design of the SRV exhaust piping would still allow for adequate steam flow even when the exhaust stack is crimped or crushed.

g. In lieu of providing the results of a draft assessment for use in decision-making for this application, as stated in the LAR, SNC is providing an analysis using the results of TORMIS. This analysis is supplemented with additional qualitative considerations to support the screening of the tornado missile hazard.

Use of TORMIS Results in Screening Tornado Missile Risk

The TORMIS analysis described in response to item (f) determines the total arithmetic sum of the damage frequency for the identified unprotected SSCs, but does not determine risk (e.g., CDF or LERF). However, given the conservatism in TORMIS analyses and the fact that multiple targets must be failed in order to cause core damage, the CDF associated with tornado missiles is estimated to be much less than 1E-6/yr.

Risk Associated with Additional Tornado Missile Targets

Subsequent to the 4b LAR submittal [B5], Southern Nuclear completed a review of the Farley site tornado missile protection in response to RIS 2015-06. Walkdowns of the site were performed as part of the review, consistent with EXT-D1 of ASME/ANS RA-Sa-2009. This review identified several additional safety-related SSCs that were not protected against tornado missiles and not included in the FNP TORMIS analysis. The additional targets are:

a) Diesel Fuel Oil Storage Tank Vents

There are five 1” vent pipes that could be crimped if hit by a tornado missile, impeding the transfer of fuel oil thereby resulting in the failure of the associated EDGs. These are small targets, approximately 1” in diameter and 1’6” tall. Although

they are located on ground level, they are very small targets and are unlikely to be hit by a tornado missile. They are located approximately 20' from each other, so it is unlikely that a single tornado missile would cause the failure of multiple vent pipes. Although the failure mode of concern is crimping, if a vent pipe is struck by an energetic and/or heavy missile, the vent pipe could instead be sheared off, obviating the crimping concern.

The FNP Severe Weather AOP, FNP-0-AOP-21.0, provides direction to review any open Tornado Missile Contingency Plans in the event of a tornado sighting, tornado warning, or severe thunderstorm, and to minimize potential tornado missiles on-site. Appendix II, Step 6 of this procedure provides specific direction for inspecting and taking necessary actions for any crimped or blocked fuel oil tank vents. This is a priority mitigating action, the first one to be taken following a tornado strike. Actions include stopping any fuel oil pumps associated with non-vented tanks and repairing the vents.

FNP-0-AOP-21.0 refers to "Crash Cart Work Orders," which are planned, packaged, and staged work orders to immediately repair tornado missile damage to the diesel fuel oil storage tank vents. Tools are available in the cold tool room located in the maintenance shop and are easily accessible if necessary. Work orders are prepared for each of the EDG fuel oil tank vents.

Based on the relatively small target size, very low probability of being damaged, proceduralized steps to inspect and correct any damage, and adequate time to correct the damage, these targets have a negligible effect on the mitigation of a tornado event on the site, and thus are considered a negligible contribution to CDF and LERF.

b) Ventilation Hoods on the Service Water Intake Structure (SWIS) Roof

There are several ventilation hoods on the roof of the SWIS that are vulnerable to tornado missile strikes. A tornado missile could crush or crimp the hoods, reducing air flow and adversely affecting the cooling of the equipment in the SWIS. Additionally, if the ventilation hoods are either damaged or removed by a missile strike, rainwater could enter the area below the hoods, potentially affecting electrical equipment located near the hoods.

Since these targets are not included in the current FNP TORMIS analysis, the likelihood of tornado missile damage to them cannot be quantitatively determined. However, an evaluation is provided to show the relatively low likelihood of tornado missile damage leading to adverse effects on the SW components in the SWIS.

1. The likelihood of tornado missile hits and damage to the intake hoods is affected by the number of missiles in the vicinity of the SWIS. TORMIS analyses typically consider missiles within approximately 2500' of a target. The SWIS is located approximately 3000' from the power block, and approximately 75% of the area within 2500' of the SWIS is outside the 2500' zone surrounding the power block. The missile inventory used for the TORMIS analysis for other targets (EDG components, SRVs, etc.) is approximately 230,000 [B1]. A large majority of the area within 2500' of the SWIS is grass or water, which does not produce any damaging missiles. It is estimated that the missile inventory affecting the SWIS is approximately half the number of missiles within 2500' of the power block, primarily consisting of trees and a few buildings. The area within 500' of the SWIS is relatively free of missiles, and there are only a few small structures and

groups of trees within 1000' of the SWIS. Therefore, the frequency of missile strikes on SWIS targets, based on missile inventory, is expected to be lower than the other TORMIS targets evaluated in FNP TORMIS analysis [B4].

2. The targets are relatively small. The area of the targets affects the likelihood of a tornado missile hit. The vent hoods vary in size, with the smallest ones approximately 3' wide x 3' long x 5' tall, and the largest ones approximately 4.5' wide x 6' wide 5.5' tall. These are comparable in surface area to the DG room ventilators in Table B-1. Although the SWIS ventilation hoods are not designed for tornado missiles, complete crushing or removal of a ventilation hood would require an energetic missile to strike the target within a limited set of trajectories and orientations. Many of the ventilation hoods spaced apart such that a single missile strike is unlikely to fail multiple redundant targets.
3. The majority of vent hood failures do not have a significant impact on the functionality of the SWIS components. Table B-2 discusses each of the vent hoods and the two failure modes (crushing and removal).

Table B-2
SWIS Ventilation Hood Targets

Location/Function	Impact of Crushing	Impact of Damage/Removal
Pump Room Air Intake (4 vent hoods)	3 exhaust fans are required for adequate pump room ventilation, but there is no analysis for the number of air intakes required [B15]. The air intakes are separated by at least 16', so the likelihood of multiple hoods being crushed by a single missile is very low. Additional air flow is provided via a 24" sump drain line in the pump room.	The hood leads to ductwork which have outlet louvers near the bottom of the floor of the pump room. Any water intrusion would drain to the sump.
Pump Room Exhaust (6)	Design calculations show that 3 of the 6 exhaust fans are needed to provide adequate cooling for the SW pumps [B15]. The vent hoods are spaced approximately 5' apart, so only the largest missiles could potentially crush two vent hoods with a single missile strike. Even then, the missile trajectory and orientation would need to be very precise to cause multiple hoods to be crushed. The design calculations are intended to ensure adequate service life of the SW pumps, so some overheating as a result of reduced ventilation would not result in imminent failure.	Removal of a ventilation hood could potentially result in rain water dripping onto the top of a SW pump motor. The top of the motors have a splash shield and the motors are not expected to be affected by rain water dripping on them. 2 of 5 SW pumps on each unit are not below an exhaust fan and would not be affected at all.

Table B-2
SWIS Ventilation Hood Targets

Location/Function	Impact of Crushing	Impact of Damage/Removal
Switchgear Room Air Intake (1 per Switchgear Room)	One air intake or one exhaust is required to provide adequate cooling to the switchgear room [B15]. The intake and exhaust ventilation hoods are approximately 20' apart, so it is extremely unlikely that a single missile would crush more than one ventilation hood. The design calculations are intended to ensure adequate service life of the electrical switchgear components, so some overheating as a result of reduced ventilation would not result in imminent failure.	The hood leads to ductwork which have outlet louvers approximately 3 feet off the floor of the switchgear room. Any rain water would drip out of the louvers, onto the floor, and into the trench along the walls, eventually draining into the SWIS pump room. Thus electrical equipment is not directly affected and flooding is not a concern.
Switchgear Room Exhaust (2 per Switchgear Room)	One air intake or one exhaust is required to provide adequate cooling to the switchgear room. The intake and exhaust ventilation hoods are approximately 20' apart, so it is extremely unlikely that a single missile would crush more than one ventilation hood. The design calculations are intended to ensure adequate service life of the electrical switchgear components, so some overheating as a result of reduced ventilation would not result in imminent failure	One of the exhaust hoods is located above a load center. Although only a portion of the switchgear is exposed to potential rain water dripping from above, it could potentially result in a failure of the switchgear. The vent hood is a relatively small target, and complete removal would be required to allow any appreciable rain water to enter the opening. There is steel grating, a fan, and fan cage in the opening, reducing direct exposure to rainwater.

Table B-2
SWIS Ventilation Hood Targets

Location/Function	Impact of Crushing	Impact of Damage/Removal
Battery Charger Room Air Intake	HVAC is not required for the battery chargers or batteries for safe shutdown functionality, so crushing is not a concern.	The hood leads to ductwork which has outlet louvers several feet off the floor, near a disconnect switch for one of the SW pumps. It is unlikely that rainwater would affect the disconnect switch. Nonetheless, only a single pump would be affected. Any rain water would drip out of the louvers, onto the floor, and into the trench along the walls, eventually draining into the SWIS pump room. Thus electrical equipment is not directly affected and flooding is not a concern.
Battery Room Exhaust	HVAC is not required for the battery chargers or batteries for safe shutdown functionality, so crushing is not a concern.	The exhaust fan openings are located above a few battery cells. If rainwater were to drip onto the battery cells, it is very unlikely that a fault would occur due to the configuration of the top of the batteries.

4. Several compensatory actions were proceduralized following the discovery of these additional tornado missile targets. Specifically, FNP-0-AOP-21.0 and NMP-OS-017 address tornado missile impacts.

   a. The FNP Severe Weather AOP, FNP-0-AOP-21.0, provides direction to review any open Tornado Missile Contingency Plans in the event of a tornado sighting, tornado warning, or severe thunderstorm, as well as minimizing potential tornado missiles. Appendix II refers to FNP-0-SOP-0.22 and directs the operators to inspect the plant for damage, including to the SWIS roof vent hoods. If in-leakage is identified, operators are directed to take immediate measures to temporarily protect electrical equipment by containing or deflecting leakage.

   b. “Crash Cart Work Orders” have been planned and developed in the event that immediate repair is required due to tornado missile damage to the SWIS Ventilation Hoods and the Diesel Fuel Oil Storage Tank Vents. A crash cart work order is a planned, packaged, and staged work order for implementation in the event damage to the Diesel Fuel Oil Storage Tank Vents and SWIS roof vents is identified. Tools are available in the cold tool room located in the maintenance shop and are easily accessible if necessary. Work orders are prepared for each of the SWIS roof vents and each of the EDG fuel oil tank vents

c. Procedures in response to high ambient temperatures in the SWIS (FNP-0-ARP-8.0) direct operators to investigate, mitigate, and correct the cause of the high temperatures.

d. NMP-OS-017 provides severe weather preparation checklists for operations and maintenance that would reduce the potential for tornado missiles and provide for quick response to any tornado missile damage following an event.

Tornado Missile Screening

As described previously, the current FNP TORMIS analysis supports a conclusion that CDF is less than 1E-6/yr due to tornado missile damage to exposed targets. The damage frequency of additional targets that are not accounted for in the TORMIS analysis is not calculated. However, the additional targets do not add significant tornado missile risk based on:

- The individual targets are small, minimizing the likelihood of a tornado missile strike
- Redundant targets are spaced apart, such that it is extremely unlikely that a single missile could damage multiple targets.
- SSC functionality is not impacted by many of the target failures
- Proceduralized compensatory actions provide for quick identification and correction of tornado missile damage to these targets. Most postulated SSC failures are not immediate, providing time for corrective actions to be taken.

Therefore, the CDF from the tornado missile hazard is estimated to be less than 1E-6/yr and is screened using EXT-C1, Criterion C.

LERF Considerations

There are no tornado missile targets associated with containment integrity, and the containment itself is extremely rugged with respect to tornado missiles. Therefore, Conditional Large Early Release Probability (CLERP) values should be consistent with other loss of offsite power sequences, such that no LERF-specific considerations are required in screening tornado missiles.

NRC RAI 12 – License Condition

LAR, Attachment 1, Section 4 states that to ensure changes in PRA methods are addressed wording similar to the following will be adopted in the License Condition:

The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC for generic use. If the licensee wishes to change its methods, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval via a license amendment.

LAR, Attachment 1, Section 4 also states that LAR Attachment 5 contains the marked-up and clean pages for the operating license with this particular condition included. However, Attachment 5 was not submitted as part of the LAR dated July 27, 2018. Provide the changes to the operating license that with the license condition that contains the wording cited above.

SNC Response to NRC RAI 12

The License Condition Markup and Clean type pages are included in Enclosure 2 and Enclosure 3 of this letter. The wording is identical to the wording cited above, in NRC RAI 12.

NRC RAI 13 – LCO 3.6.2 C, Containment Air Locks

The LAR Enclosure 1, Table E1.1 regarding LCO 3.6.2 Condition C (One or more Containment Air Locks inoperable) indicates that the SSCs are modeled consistent with the TS scope and so can be directly evaluated using the CRMP tool. It also states that the PRA success criteria are consistent with the design basis success criteria. The same table indicates that the design basis success criterion is "Post-Accident Containment Leakage Rates within limits." The NRC staff notes that system success criteria is typically modeled in a PRA by using fault tree logic to define how many components or trains are needed for success. Therefore:

a. Explain how the Containment Air Locks are modeled in the PRA model (CRMP model) supporting the RICT program to reflect the design basis success criteria and how impact on CDF and/or LERF can be estimated for the RICT calculation. Include an explanation of how the CRMP model is adjusted to account for post-accident containment leakage rates that are out of limits.

b. Provide justification that LCO 3.6.2 C is not a LOF [loss of function] condition in which all required trains or subsystems of a TS required system are inoperable.

SNC Response to NRC RAI 13

a. The containment airlock is not explicitly modeled in the PRA. The risk impact of an open or not fully closed containment airlock will be considered in the RICT calculation through basic events in the Farley PRA LERF logic representing a non-intact containment boundary, e.g., ADMN-PEN-NI (ADMINISTRATIVELY CONTROLLED PENETRATIONS NOT ISOLATED). These events will be used as surrogates for the status of the containment airlock RICT calculation, similar to the Vogtle RICT program. Failure of these events represent the possibility of radioactive releases from containment as would be the case if one of more of the airlock doors was non-functional. This is conservative for the single airlock door action as the remaining door forms an operable barrier to release. There is no CDF impact modeled for an open airlock in the Farley PRA. The equipment tag numbers for the personnel airlock, escape airlock, and the equipment hatch, along with LCO 3.6.2.C, will be mapped to the containment isolation failure basic event in the CRMP model. Selecting any of these tag numbers or LCO 3.6.2.C to remove it from service in the CRMP model, reflecting an open airlock, results in a direct impact on LERF. As noted in Table E1.2, removal of any airlock from service in the CRMP model produces a relatively short RICT that is limited by LERF.

b. If a LOF occurs while in 3.6.2.C, 3.6.1.B will also be entered. The LOF is handled in 3.6.1.B and this Condition is not being included within the scope of the Farley RICT program. 3.6.2 required action C.1 requires evaluating the overall containment leakage rate per LCO 3.6.1. If the overall containment leakage rate is greater than max allowable value, LCO 3.6.1 Condition B would be entered to address the loss of function for containment. If the overall containment leakage rate is less than max allowable value, a loss of function does not exist and the RICT program would be applied to required action 3.6.2 C.3.

NRC RAI 14 – PRA Modeling of Technical Specifications SSCs in Scope of the RICT Program

Regulatory Position 2.3.3 of RG 1.174 states that the level of detail in the PRA should be sufficient to model the impact of the proposed licensing basis change.

The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated. Full-scale applications of the PRA should reflect this cause-effect relationship in a quantification of the impact of the proposed licensing basis change on the PRA elements.

The SE for NEI 06-09 states that a RICT can be applied to SSCs that are either modeled in the PRA, or whose impact can be quantified using conservative or bounding approaches. The LAR did not provide sufficient description of the PRA modeling for some systems, therefore address the following:

a. For LCO 3.6.6, explain how the Containment Spray System is modeled in the PRA model (CRMP model) supporting the RICT program to reflect the design basis success criteria and how impact on CDF and/or LERF can be estimated for the RICT calculation, consistent with the guidance in NEI 06-09 Revision 0-A.

b. For LCO 3.6.6, explain how the Containment Cooling system is modeled in the PRA model (CRMP model) supporting the RICT program and how impact on CDF and/or LERF can be estimated for the RICT calculation, consistent with the guidance in NEI 06-09 Revision 0-A.

c. For LCO 3.7.11.E (Two Control Room Air Conditioning System (CRACS) Trains Inoperable), LAR Enclosure 1, Table E1.1 states the following:

   "*Not Modeled – Documented in the PRA basis as not needed to prevent core damage*."

   Address the following:

   i. If the CRACS system is not modeled in the PRA, justify why this condition can be included in the scope of the RICT program, consistent with the guidance in NEI 06-09 Revision 0-A.

   ii. Explain how the CRACS is modeled in the CRMP model supporting the RICT program and how a change in CDF and/or LERF can be calculated for the RICT estimate. Include description of the CRACS success criteria that is modeled and explanation of the impact that failure of CRACS has on other systems modeled in the CRMP.

d. The licensee's slides (ADAMS Accession No. ML18290B094) presented at the October 16, 2018 public meeting (ADAMS Accession No. ML18306A313) summarized the design for the Farley electrical distribution systems. As described at the public meeting, swing Emergency Diesel Generators (EDG) 1-2A and 1C are shared between Unit 1 and Unit 2. EDG 1-2A is rated at 4075 kW and EDG 1C is rated at 2850 kW. They can be aligned to the 'A' train buses of the AC electrical distribution system of each unit. If a safety injection (SI) signal is received, EDG 1-2A aligns to the A train of the unit that received the first SI signal, and EDG 1C aligns to the 'A' train of the opposite unit. During a simultaneous dual unit Loss of Offsite

Power (LOSP) without SI, EDG 1-2A is assigned to Unit 1 and EDG 1C is assigned to Unit 2. For a LOSP on single unit, EDG 1-2A aligns to that unit. Given the sharing of the EDGs, it appears that the RICT estimates for one unit can be impacted by the configuration of the opposite unit.

Briefly summarize how the sharing of the EDGs between Unit 1 and Unit 2 and the preferred alignment of DG 1-2A is modeled in the PRA and the CRMP tool and justify how it adequately captures the risk impact on one unit from the opposite unit's real-time configuration.

SNC Response to NRC RAI 14

a. The Containment Spray system is explicitly modeled in the Farley PRA models and can be numerically quantified for impact on LERF. The Containment Spray impact on CDF is accounted for in the PRA success criteria as operation of Containment Spray affects RWST inventory and timing during ECCS injection. Containment Spray modeling includes system components such as pumps and valves, and system dependencies such as electrical and cooling water systems. Operation of the Containment Spray system in the injection mode is modeled as it impacts the timing of the transfer to high-head or low-head recirculation and containment isolation requirements. If Containment Spray started to operate and then later Containment Spray pumps fail to run, it would create potential LERF pathways if isolation of the Containment Spray penetrations fail.

   Only the Containment Spray system components associated with the injection phase of Containment Spray system operation are modeled in the PRA model. The failed equipment would be directly failed in the RICT calculation. In a sensitivity where the Containment Spray pumps were assumed failed in the IEIFPRA model, at normal truncation, there was no impact on LERF.

b. The Containment Cooling system is explicitly modeled in the Farley PRA. The system consists of 4 fan coolers served by the Service Water system. Any 2 of the 4 fan coolers is modeled as success in the PRA. The modeling includes the containment cooling fans and motors, cooling coils, dampers in the air supply and return ducting, various valves in the SW supply to the cooling units, and start and power supply logic. The failed equipment would be directly failed in the RICT calculation. The impact on CDF and LERF are minimal based on a sensitivity where all Containment Fan Coolers were assumed failed in the internal events including internal flooding PRA model.

c. LCO 3.7.11.E "Two CRACS Trains Inoperable" will be removed from the scope of the Farley RICT Program. Because no other Conditions were requested under this LCO, new markups are not needed. SNC would like to withdraw this request from the scope of the NRC review.

d. For a LOSP on a single unit, EDG 1-2A aligns to that unit. In the PRA models, during a single unit LOSP, the availability of both EDG 1/2A and EDG 1C are captured in both the Unit 1 and Unit 2 model.

   During a simultaneous dual unit LOSP without a SI, EDG 1-2A is assigned to Unit 1 and EDG 1C is assigned to Unit 2. In the Unit 1 model, during a dual unit LOSP (with no SI), only the availability of EDG 1-2A is modeled. In the Unit 2 model, during a dual unit LOSP (with no SI), only the availability of EDG 1C is modeled.

During a simultaneous dual unit Loss of Offsite Power (LOSP) with a SI, EDG 1-2A will be assigned the unit that received the SI signal first. In both the Unit 1 and Unit 2 models, during a dual unit LOSP with an SI on that unit, the availability of EDG 1-2A is modeled including a basic event that represents a probability that EDG 1-2A is required for the opposite unit. The probability of this basic event is calculated from a top gate in the opposite unit's model. For example, the Unit 1 model basic event description is, "Diesel Generator 1/2A Required for Unit 2." The top gate in the Unit 2 is, "DG 1/2A Required for Unit 2." The top gate models a dual unit LOSP initiator and failures that would cause an SI to occur. Every time the PRA models are updated, these top gates in each unit's model are quantified and the results are then placed in the basic event in the opposite unit's model.

Since the composite unavailability of the swing diesel(s) for a given unit due to being required for the opposite unit is based on the average probability of simultaneous dual unit LOSP and SI, it is possible that some evolutions on the opposite unit that increase the probability of one of those individual initiating events may dynamically increase that unavailability to some degree. Additionally, for some periods of time, the unavailability would be lower than the average calculated value.

Initially, the RAW values of these basic events were assessed in the average model. The RAW values are less than what would normally be considered risk significant using NUMARC 93-01 guidance, and indeed do not appear in the Unit 2 cutsets at all. Additionally, a sensitivity was performed to assess the impact to CDF and LERF of increasing this average swing diesel unavailability. The basic event value for the 1/2A Diesel Generator from each unit was increased two orders of magnitude, from its calculated value of 5.79E-3 to 5.79E-1 (Unit 1) and from the calculated value of 5.70E-4 to 5.70E-2 (Unit 2). The changes to CDF and LERF were consistent with what was expected from the RAW values.

Given the results of the sensitivity that shown minimal impact from increasing the value of the unavailability event, the use of the average value for the event in the CRMP tool will adequately capture the risk impact due to changes in the opposite unit's real-time configuration. Additionally, the RMA process for a RICT in a given unit will consider such changes.

NRC RAI 15 – Potential TS Loss of Function (LOF) Conditions

The LAR states that the Farley application is consistent with the Risk-Informed Technical Specifications Program approved by the NRC for SNC's Vogtle Electric Generating Plant (VEGP), Units 1 and 2 (ADAMS Accession Number ML15127A669). As approved for VEGP, the LCO conditions that are considered TS LOF (i.e. those conditions that represent a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE) are modified by additional notes providing restrictions applicable to the TS LOF Conditions, including a backstop of 24 hours, restricting voluntary entry into the TS LOF condition, and additional criteria for declaring SSCs PRA Functional.

a. For LCO 3.6.6, Containment Spray and Cooling Systems, Conditions D and E, the licensee stated in LAR Enclosure 1, Table E1.1, that the design success criteria for the containment cooling system is one of two containment cooling trains. The NRC staff notes that this design success criteria cannot be met in Condition E of LCO 3.6.6 when two containment cooling trains are inoperable, and would therefore would be considered a TS LOF.

Justify why having two of two containment cooling trains inoperable in LCO 3.6.6 Condition E is not considered a loss of function, or provide updated TS markups for this condition.

b. For LCO 3.7.4 A, Atmospheric Relief Valves (ARVs), the licensee stated in LAR Enclosure 1, Table E1.1, that there are three ARVs, and the PRA success criteria states that four of four ARVs are required for Anticipated Transient Without Trip (ATWT) conditions. The NRC staff notes there is a discrepancy between the number of ARVs covered by LCO 3.7.4 and the number of ARVs required for PRA success criteria. If all of the ARVs are required to mitigate an ATWT condition, as implied by LAR Table E1.1, then LCO 3.7.4 Condition A, where one ARV is inoperable, and LCO 3.7.4 Condition B, where two ARVs are inoperable, would be considered a LOF.

   Justify why having one or two ARVs inoperable in LCO 3.7.4 Conditions A and B is not a LOF during ATWT conditions, or provide updated TS markups for this condition.

c. As approved for VEGP, the TS LCO condition with both EDG inoperable is treated as a LOF. For Farley LCO 3.8.1 Condition E, two DG sets inoperable, the licensee states (LAR page E1-13) that this is consistent with the VEGP SE. However, the staff notes that Farley LCO 3.8.1 Condition E is not marked as loss of function.

   Confirm SNC's intent to treat LCO 3.8.1 Condition E as loss of function, and provide updated TS markups for LCO 3.8.1.E.

<u>SNC Response to NRC RAI 15</u>

a. A condition with both trains of Containment Cooling inoperable would be a loss of function. This appears to have been inadvertently omitted from Table E1.1. The TS 3.6.6.G entry in Table E1.1 of Enclosure 1 of the LAR implies that a loss of both trains of containment cooling is considered to be a loss of function. (new condition E) The front stop for Condition D is 72 hours, for this LCO, SNC would like to withdraw this condition from the scope of the RICT program. Enclosures 2 (page 3.6.6-2) and Enclosure 3 (page 3.6.6-2) contain new TS markup, clean type pages for this LCO.

b. The number of ARVs in each Farley unit is 3. The actual success criteria used in the PRA model is for an ATWT event is 4/5 MSSVs, not ARVs. The success for secondary side cooling in an ATWT does not depend on the ARVs. LCO Conditions A and B are not Technical Specification loss of function. The TS Bases for 3.7.4 refer only to maintaining one ARV line available to conduct a cooldown following a SGTR event and refer to the Steam Dump System and the MSSVs also being available, and the low probability of an event occurring during this period that would require the ARV lines.

   In reviewing the changes needed to respond to these RAIs, SNC found that a change was needed to address an issue with LCO 3.7.4 markups. Insert 13 has been revised to retain the existing 24 hour completion time for three required ARV lines inoperable. Because the licensing basis is already approved for up to 24 hours, SNC will not apply the RICT program to new Condition C. Consistent with the LAR, because 2 ARV inoperable is not considered a LOF, SNC will apply the RICT program without LOF constraints to new Condition B. New TS Markups and Clean Type pages are provided in

Enclosure 2 (Revised Insert 13, See LAR Attachment 2 page 3.7.4-1) and Enclosure 3 (3.7.4-1 and 3.7.4-2).

c. The current version of Farley LCO 3.8.1 E. includes 3 Completion Times which correspond to different combinations of DG failures. The 2-hour Completion Time is invoked when all 3 DGs, capable of being aligned to a particular unit, are inoperable. This failure combination includes DG 1-2A, DG 1C and either DG 1B or DG 2B. The 8-hour Completion Time is invoked when DG1-2A and either DG 1B or DG 2B are inoperable. The 24-hour Completion Time is invoked when DG 1C and either DG 1B or DG 2B are inoperable.

   For the combination of DG failures which currently has a 24-hour Completion Time, SNC will retain the current licensing basis and will not apply the RICT program to this combination of DG failures. SNC considers the combinations of DG failures with 2- and 8-hour Completion Times as useful candidates to apply the RICT program and agrees these combinations are a loss of function, which is consistent with the Vogtle safety evaluation. SNC will apply the appropriate LOF RICT program constraints to these conditions. Consequently, Condition E will need to be divided into three Conditions to incorporate this change, one Condition will be added for each of the Completion Times. The new TS markups and clean type pages for LCO 3.8.1 are contained in Enclosures 2 (pages 3.8.1-4 and 3.8.1-5) and Enclosure 3 (3.8.1-5 through 3.8.1-15 or 16) to this letter.

NRC RAI 16 – LCO 3.7.6.D Condensate Storage Tank PRA Success Criteria

Regulatory Position 2.3.3 of RG 1.174 states that the level of detail in the PRA should be sufficient to model the impact of the proposed licensing basis change. The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated. Full-scale applications of the PRA should reflect this cause-effect relationship in a quantification of the impact of the proposed licensing basis change on the PRA elements.

The LAR Enclosure 1, Table E1.1, regarding the Condensate Storage Tank (CST), proposes to credit the availability of Plant Service Water suction to the AFW [Auxiliary Feedwater System] pumps as the success criteria for the CST in the event the CST becomes inoperable. The disposition for LCO 3.7.6. Condition D (i.e., CST Inoperable) indicates that this condition is a TS Loss of Function (LOF) condition and states:

> *An NRC approval is sought as part of this LAR submittal to credit use of plant service water as modeled in the PRA as an alternate source of water to recover degraded CST design basis parameters for establishing PRA Functionality.*

The statement cited above and proposed changes to LCO 3.7.6 Condition D indicate that credit for an alternate SSC other than an SSC covered by the TS is being proposed for PRA Functionality, which is inconsistent with the administrative control in TS 5.5.20, item f.1 which states:

> *Any structures, systems, and components (SSCs) credited in the PRA Functionality determination shall be the same SSCs relied upon to perform the specified Technical Specification safety function.*

In light of the observations above:

a. Reconcile the two statements from the LAR cited above. Include clarification of whether credit for alternate SSCs other than the SSC covered by the TS is being proposed for use in a PRA Functional determination for LCO 3.7.6 Condition D.

b. If alternate SSCs other than the SSC covered by the TS is being proposed for PRA Functionality for LCO 3.7.6 Condition D, provide detailed justification how sufficient defense-in-depth and safety margins are maintained in this condition, consistent with the risk-informed principles in RG 1.174.

SNC Response to NRC RAI 16

a. There is a small typo in the NRC RAI wording above. SNC did not request approval of Condition D. In all circumstances in the above wording, Condition D should refer to Condition A.
SNC has decided to remove TS 3.7.6 Condition A from the scope of TSs requested by this application. Because no other Conditions were requested under this LCO, new markups are not needed. SNC would like to withdraw this request from the scope of NRC review.

b. Not applicable.

NRC RAI 17 – LAR Inconsistencies

Address the following inconsistencies noted in the LAR:

a. LAR provides two versions to the wording for the proposed addition to the TS Administrative Controls Section 5.5.20.g: LAR Attachment 1 (pages E1-4 and E1-5) and LAR Attachment 2, Insert 27.

   The LAR Attachment 1 states however that the text "is consistent with TSTF-505 and NEI 06-09, Revision 0-A and amended for the adjustments made to the Vogtle Electric Generating Plant (VEGP)". The staff notes however that the text in LAR Attachment 1 text appears to differ from VEGP and the text in LAR Attachment 2 text is consistent with the VEGP SE.

   Clarify this discrepancy between LAR Attachment 1 and 2, and confirm that SNC intends to use the same text as the VEGP TS Administrative section, consistent with the statements made in the LAR.

b. For LCO 3.5.2 (ECCS Operating), LAR Attachment 1 states that Condition A is to be modified as "One or more trains inoperable AND at least 100% of the ECCS flow equivalent to a single Operable ECCS [Emergency Core Cooling System] train available."

   However, in the TS markup in LAR Attachment 2 and the proposed clean-typed TS pages provided in LAR Attachment 3, Condition A states "One or more trains inoperable."

   Clarify this discrepancy between the proposed modified LCO 3.5.2 Condition A in LAR Attachment 1 and LAR Attachments 2 and 3.

SNC Response to NRC RAI 17

a. The wording in Attachment 1 is not the wording that SNC would like to adopt for Plant Farley. SNC intends to adopt the wording documented in Attachment 2 (Technical Specifications Markups). The wording in Attachment 2 is also consistent with Attachment 3 (Technical Specifications Clean Type pages).
b. The words, "AND at least 100% of the ECCS flow equivalent to a single Operable ECCS train available" should not be included in Attachment 1. The correct wording is in Attachment 2 and Attachment 3 which excludes "AND at least 100% of the ECCS flow equivalent to a single Operable ECCS train available."

NRC RAI 18 – PRA Functionality for Systems Not Credited in the PRA

NEI 06-09, Section 2.3.1, Step #11 and Section 3.2.3 provides guidance on performing PRA Functionality determination. LAR Enclosure 8, Section 2.4 discusses Farley's procedure for determining whether SSCs that are declared TS inoperable can be considered PRA functional. According to the LAR, the procedure identifies three specific conditions in which a TS inoperable SSC can be PRA Functional. For Condition #3, the LAR states the following based on guidance from NEI 06-09:

> *If the condition causing the inoperability per Technical Specifications impacts only function(s) that are not modeled in CRMP and the [Farley Nuclear Plant] FNP PRA has concluded that the affected function(s) has no risk impact, then the SSC may be considered PRA functional.*

The NRC staff notes that the reason for excluding certain Technical Specifications functions or SSCs from the PRA models can be based on explicitly stated or implicit assumptions made in the PRA modeling or because credit for the SSC or function was excluded from the PRA. Confirm that such PRA assumptions and modeling decisions are assessed during the PRA Functionality determination to conclude that an inoperable SSC not explicitly modeled in the PRA can be considered PRA Functional.

SNC Response to NRC RAI 18

SNC confirms that PRA assumptions and modeling decisions must be assessed during the PRA functionality determination to conclude that an inoperable SSC not explicitly modeled in the PRA can be considered PRA functional. Any case involving failure of a non-explicitly modeled component will fit Scenario A or B, as described below.

Scenario A: RICT system guidelines document no impact to CDF and LERF
The RICT system guidelines document that the Tech Spec function or SSC is not included in the PRA logic model due to having no impact on CDF and LERF. For a case that fits Scenario A, the inoperable SSC may be considered PRA functional in the RICT/RMAT calculations.
Scenario A Example: A pump is inoperable due to low cooling water flow to the pump seal. The RICT system guideline documents that the pump's seal cooling is not included in the PRA logic model because the pump will function with no seal cooling for a significant duration beyond the 24 hour PRA mission time. The RICT is calculated with the pump functional.

Scenario B: Case does not fit Scenario A
The RICT system guidelines do not document a basis or the documented basis for excluding the Tech Spec function or SSC from the PRA logic model does not fit scenario A. For any case

that fits Scenario B, the RICT/RMAT calculations are performed with the inoperable SSC not PRA functional, or another bounding method is used to affect an increase to CDF and LERF to account for the failed Tech Spec function or SSC.
Cases that fit scenario B:

- The RICT system guidelines document no basis for excluding the failed Tech Spec function or SSC from the PRA logic model.
- The RICT system guidelines document that the failed Tech Spec function or SSC is not included in the PRA logic model based on inherent reliability (i.e. very low probability of failure).
- The RICT system guidelines document that the failed Tech Spec function or SSC is included in the logic model with assumed failure (i.e. no credit for its mitigative value).

Scenario B Example: A motor-operated valve is inoperable due to failing in a closed or intermediate position during a test stroke. The RICT system guideline documents that the valve is not modeled for failure to close in the PRA due to no impact on CDF and LERF. The RICT system guideline documents that the valve is not modeled for failure to open in the PRA because the valve is maintained open during at-power operation. The SSC is not PRA functional. The RICT is calculated with a surrogate valve failed closed or a surrogate pump failed, which bounds the failed valve's impact to CDF and LERF.

NRC RAI 19 – Identification of Compensatory Measures and RMAs

The NRC SE portion of the NEI 06-09 0-A, states that the LAR will describe the process to identify and provide compensatory measures and RMAs [risk management actions] during extended CTs. LAR Enclosure 10 identifies four kinds of RMAs (i.e., actions to provide increased risk awareness and control, reduction of the duration of maintenance activities, reduction of the magnitude of risk increase, and minimization of the risk of a common cause failure). LAR Enclosure 10 also provides numerous specific examples of RMAs for electrical distribution related LCOs. LAR Enclosure 10 does not describe what criteria or insights (e.g., important fire areas, important operator actions) are used to determine what RMAs to apply in specific instances.

Describe what criteria or insights (e.g., important fire areas, important operator actions) are used to determine the compensatory measures and RMAs for specific plant configurations.

SNC Response to NRC RAI 19

Determination of RMAs involves the use of both qualitative and quantitative considerations for the specific plant configuration and the practical means available to manage risk. The scope and number of RMAs developed and implemented are reached in a graded manner.

SNC procedure NMP-GM-031-003 contains the instructions for development and implementation of RMAs to meet requirements of 10CFR50.65(a)(4) and the RICT program. A lot of these instructions were developed prior to the RICT program in support of the 10CFR50.65(a)(4) program. The SNC procedure implements the same RMA requirements for 10CFR50.65(a)(4) and RICT, except for requirements for Common Cause RMAs which only apply to RICT implementation. The table below states the RMA requirements for various potential conditions.

Beyond Front Stop?	Beyond RMAT?	Risk Profile	RMA Requirements
No	No	Green	none
	Yes	Green	none
		Yellow	*Level 1 and Fire RMAs
		Orange	*Level 1 & 2 and Fire RMAs
		Red **	*Level 1, 2 & 3 RMAs
Yes	No	Green	none
	Yes	Green	*Level 1 and Fire RMAs
		Yellow	*Level 1 and Fire RMAs
		Orange	*Level 1 & 2 and Fire RMAs
		Red ***	*Level 1, 2 & 3 and Fire RMAs

* RMAs are implemented as applicable to the Hazard(s) driving Risk
** RICT entry is not applicable because the RICT is less than the front stop
*** RICT will expire prior to reaching a Red profile

NMP-GM-031 defines RMAs as follows:

RISK MANAGEMENT ACTIONS (RMA)- Actions that reduce risk associated with maintenance, testing, tagouts, system alignments, equipment failures, external environmental factors, or activities that increase the likelihood of initiating events.

a. NON-FIRE RMAs – Actions that reduce the likelihood of initiating events (e.g. LOCAs, Rx Trip, SGTR, internal floods, etc) and increase the likelihood that important equipment will be available to mitigate an initiating event. Non-fire RMAs are categorized into levels as follows:

(1) LEVEL 1 RMAs - Actions that increase awareness of the risk, control of the activity, and rigor associated with planning of activities.
(2) LEVEL 2 RMAs - Actions that minimize activity durations and the magnitude of risk increase.
(3) LEVEL 3 RMAs - Actions that escalate risk awareness and instill an acute urgency to lower risk.

b. FIRE RMAs – Actions that reduce the likelihood of a fire causing an initiating event or damaging accident mitigation equipment.

c. COMMON CAUSE RMAs – RMAs that target the success of the redundant and/or diverse SSCs of the failed SSC and, if possible, reduce the frequency of initiating events which call upon the function(s) performed by the failed SSC.

NMP-GM-031-003 section 4.2 contains immediate actions for emergent non-green risk profiles (i.e. ICDP>1E-6 or ILERP>1E-7), including stopping work that could generate a plant transient or initiating event, taking action to restore to functional status any out of service components that provide redundancy to the failed SSC(s), deferring any activities that might impact availability of functional components, including fire detection and suppression equipment, and initiating response teams and augmented staffing.

In addition to implementation of RMAs developed qualitatively, NMP-GM-031-003 directs development of RMAs based on use of the CRMP tool to identify RMA candidates for managing risks associated with non-fire events (i.e. internal events and internal flooding), fire events, and common cause failures.

For non-fire events, the following RMA candidate reports are generated with the CRMP tool:

- Out of Service Component Importance - for non-fire events, lists the out of service components that are most important to return to functional status in order to mitigate the specified top event (core damage or large early release).
- In Service Component Importance - for non-fire events, lists the inservice components that are most important to maintain as functional in order to mitigate the specified top event (core damage or large early release).
- Initiator Importance - lists the non-fire initiating events which, if prevented from occurring, will result in the greatest reduction of risk of the specified top event (core damage or large early release).

For fire events, the following RMA candidate reports are generated with the CRMP tool:

- In Service Component Importance - for fire initiating events, lists the inservice components that are most important to maintain as functional in order to mitigate the specified top event (core damage or large early release).
- Fire Zone Initiator Importance - lists the fire zones in which a fire may become a fire initiating event resulting in the top event (core damage or large early release).
- Fire Zone Component Importance - lists the fire zones in which a fire could destroy components that are important for mitigating the top event (core damage or large early release).

NMP-GM-031-003 contains the following guidance for development of RMAs
Non-Fire RMAs:

LEVEL 1 RMAs increase risk awareness, control of the activity, and rigor associated with planning of activities. LEVEL 1 RMAs are not limited to, but may include:

- Brief operating shifts and increase operator awareness of configuration specific risks.
- Conduct pre-job briefing of maintenance personnel, emphasizing risk aspects of planned maintenance activities.
- Increase control of activities that could result in an initiating event (e.g. loss of offsite power). These initiating events should be identified using the Initiator Importance portion of the NON-FIRE RMA Candidates report.
- Protect functional components that are most important for mitigating non-fire events. These components should be identified using the In Service Component Importance portion of the NON-FIRE RMA Candidates report.
- Require a knowledgeable observer or subject matter expert to be present for the maintenance activity, or for applicable portions of the activity.

LEVEL 2 RMAs minimize the duration of the maintenance activity (usually addressed in system outage plans for planned entries) and reduce the magnitude of risk increase. LEVEL 2 RMAs are not limited to, but may include:

- Pre-stage parts and materials.
- Conduct training on mockups to familiarize maintenance personnel with the activity.
- Perform maintenance around the clock.
- Establish contingency plans to restore to functional status those out of service components that are most important to accident mitigation. These components should be identified using the Out of Service Component Importance portion of the NON-FIRE RMA Candidates report.
- Defer activities that could result in an initiating event (e.g. loss of offsite power). These initiating events should be identified using the Initiator Importance portion of the NON-FIRE RMA Candidates report.
- Protect a greater number of the functional components that are most important for mitigating non-fire events. These components should be identified using the In Service Component Importance portion of the NON-FIRE RMA Candidates report.
- Establish alternate success paths for performance of the safety function of the out-of-service equipment. Equipment used to establish these alternate success paths need not necessarily be within the overall scope of the maintenance rule (can use portable equipment).
- Evaluate and implement alternate plant alignments that minimize risk.

  For example, minimize the number of components running on the protected train safety bus during a diesel generator extended AOT, such that load shed of the protected safety bus loads is more likely to succeed.
- Walkdowns of key safety systems by on-shift SROs and management personnel before and during the work activity.
- Increasing surveillance frequencies of key safety functions by testing alternate equipment prior to the planned work or frequent inspections of standby equipment during work.
- Establish other compensatory measures such as temporary power or pumps.
- Reschedule risk significant work.
- Reduce the duration of risk significant work.

LEVEL 3 RMAs escalate risk awareness and instill an acute urgency to lower plant risk. LEVEL 3 RMAs include, but are not limited to, LEVEL 1 and 2 RMAs, supplemented by

elevated vigilance, urgency, awareness and allocation of all available resources to bring the plant risk below RED.

- Take immediate action to restore to functional status those out of service components that are most important to accident mitigation.
- These components should be identified using the Out of Service Component Importance portion of the NON-FIRE RMA Candidates report.
- If unable to transition below RED in a reasonable amount of time, not to exceed 3 days from the time a RED condition was entered, consider an orderly transition to Mode 3.
- Contact RIE staff for additional guidance on potential means of lowering risk below RED.

FIRE RMAs

FIRE RMAs reduce the likelihood that internal fire results in a core damage event and/or large early radiological release event. Fire RMA recommendations are listed below:

- Implement actions that generate increased fire risk awareness, control, and coordination.
- Confirm the availability of alternate success paths for safe shutdown if required.
- Protect functional components that are most important for mitigating fire events. These components should be identified using the In Service Component Importance portion of the FIRE RMA Candidates report.
- Select important fire zones for increased controls from the Fire Zone Initiator Importance portion and the Fire Zone Component Importance portion of the FIRE RMA Candidates report.
- For the selected fire zones, verify and maintain functionality of the following:
    - Detection
    - Suppression
    - Barriers
    - Fire Pumps
- For any selected fire zones with degraded or unavailable fire protection equipment, the following actions may be taken:
- Place restrictions on work activities (including “hot work”) that could cause fires.
- Place restrictions on storage and movement of transient combustibles.

- Perform walkdowns to verify orderly storage of transient combustibles.
- Implement fire watches.
- Install temporary fire barriers such as fire wraps, blankets, or other approved barriers to protect cables or other SSCs from being damaged.
- Pre-stage firefighting personnel and/or equipment to reduce fire severity and propagation.
- Perform thermography to identify electrical hot spots.
- Defer circuit breaker operations for 480V and higher rating breakers.

COMMON CAUSE RMAs

Prior to exceeding the FRONT STOP, develop and implement COMMON CAUSE RMAs that target the success of the redundant and/or diverse SSCs of the failed SSC and, if possible, reduce the frequency of initiating events which call upon the function(s) performed by the failed SSC. Documentation of RMAs shall be available for NRC review.

COMMON CAUSE RMAs lower configuration risk by focusing on

a. Availability of SSCs providing redundancy to the failed SSC,
b. Availability of <u>diverse</u> SSCs (e.g. normal charging pump) providing redundancy for functions performed by the failed SSC,
c. Reducing the likelihood of events that can impact the availability of SSCs described in (a) and (b),
d. Reducing the likelihood of events for which event mitigation may require operation of SSCs described in (a) and (b),
e. Readiness of operators to respond to initiating events assuming SSCs susceptible to failure by common cause will fail, and
f. Readiness of maintenance to respond to additional failures of SSCs described in (a) and (b).

COMMON CAUSE RMAs <u>SHALL</u> include the following actions:

- Defer maintenance and testing activities that could generate an initiating event for which event mitigation may require operation of SSCs susceptible to failure by common cause.
- Establish a comp action, shift brief, or standing order that focuses on actions operators will take in response to an initiating event and failure of SSCs susceptible to failure by common cause.

- For the SSCs that provide redundancy to the failed SSC,
    - Reduce the likelihood of unavailability, including for support systems and power supplies.
    - Perform non-intrusive inspections.
    - Defer maintenance and testing activities that could impact availability of the SSC.
- For diverse SSCs (e.g. normal charging pump) that provide redundancy for functions performed by the failed SSC,
    - Reduce the likelihood of unavailability, including for support systems and power supplies.
    - Perform non-intrusive inspections.
    - Defer maintenance and testing activities that could impact availability of the SSC.

COMMON CAUSE RMAs may include the following actions:

- For applicable standby SSCs, perform an operability/functionality run.
- Establish an alternate functional capability (e.g. installation of portable equipment).
- Generate and implement a contingency plan to
    - Enable prompt installation of an alternate functional capability (e.g. shiftly review of procedures on use of portable equipment), or
    - Enable prompt restoration of functionality of a failed SSC (e.g. maintenance crash cart)
- For applicable running components, monitor parameters more frequently and/or expand the scope of parameter monitoring.
- For applicable SSCs, perform monitoring and inspection activities based on review of information/data from previous testing, maintenance, and/or operating experience.

Per TS, use of a RICT is permitted for emergent conditions (i.e. equipment failure) which represent a loss of a specified safety function or inoperability of all required trains of a system required to be operable if one or more trains are considered PRA Functional. In other words, RICT is applicable to a LOF condition only when PRA functionality is established for at least one train. The LOF condition itself does not invoke a requirement for RMAs; however, the emergent failure/condition requires that the potential for common cause (CC) failure be addressed by one of the methods stated in TS, one of which is implementation of RMAs not already credited in the RICT calculation that target the success of redundant and/or diverse SSCs and, if possible, reduce the frequency of initiating events which call upon the functions performed by the failed SSC.

A LOF condition in which a RICT can apply would most likely occur when, during planned maintenance on a train/component, a condition is discovered on the redundant train that results in inoperability, but the redundant train is PRA functional despite the discovered condition. In this case, the train/component in maintenance dictates the RMAT prior to the LOF and after the LOF. In other words, RMAs were either in place prior to the LOF, become required prior to expiration of the 24-hour back stop, or the RMAT occurs after the expiration of the 24 hour back stop. From a practical standpoint, entry into a shutdown LCO with a 24-hour back stop would significantly elevate vigilance, urgency, awareness and allocation of resources, regardless of the RMAT.

Thus, RMAs associated with a TS LOF in which PRA Functional components are credited are identified by the same methods detailed in the SNC response to NRC Additional Discussion Topic 4a.

NRC RAI 20 – 5.5.20 Risk Informed Completion Time Program

The RICT program is invoked in the CT column (by Insert 3) of the associated Condition statement, with the words, "OR In Accordance with the Risk Informed Completion Time Program." The 5.5.20 RICT Program incorporates by reference the requirements of NEI 06-09-0A, and for emphasis it reiterates significant requirements in paragraphs a through h. In the request to implement the RICT Program, Notes are proposed to be added to the Condition statements addressing potential LOF Conditions. The second Note in the inserts addressing LOF which states, "The following Section 5.5.20 constraints are applicable: parts b, c.2, c.3, d, e, f, g and h," is unnecessary in that it is redundant to what is explicitly stated in 5.5.20, and it raises doubt to the applicability to other RICT Program requirements, such as the omitted "part a," which are still applicable. Please consider removing from the Condition Notes, the duplicative 5.5.20 requirements or justify why the Notes are needed.

SNC Response to NRC RAI 20

For all LOF conditions, a revised Note 1 is proposed to clarify that RICT entry is not permitted for LOF Conditions. The revisions clarify that each Condition can be entered voluntarily but a RICT cannot be calculated to extend the Completion Time. The additional note, calling out subsections of the Administrative Section 5.5.20, has been removed for all LOF Conditions as requested. TS markups and clean type pages are included in Enclosures 2 and 3 of this letter.

Additionally, SNC identified an issue with some of the previously provided TS markups. In 7 circumstances, SNC did not identify the appropriate end state for LOF Conditions. On June 10, 2016, Amendments 202/198 were approved to modify end states from Mode 5 to Mode 4 for specific TS conditions based on an analysis in WCAP-16294. One of the constraints discussed in the NRC safety evaluation report is that the modified end states are limited to conditions where entry is initiated by inoperability of a single train of equipment. The TS pages, included in the LAR, incorrectly applied a Mode 4 end state for the following loss of function Conditions: 3.6.6 B, 3.7.7 B, 3.7.8 B, 3.8.1 J, 3.8.4 F, 3.8.7 B, 3.8.9 A. SNC has provided corrected, revised markups for these Conditions in Enclosures 2 and 3 of this letter.

SNC Supplemental Request

While reviewing the changes made for RAI 20, SNC discovered that the LAR identified TS 3.6.3 Condition C as an LOF condition. The current completion time is 72 hours which exceeds programmatic LOF backstop of 24 hours. The specified time period is reasonable considering

the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity. Condition C is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of the Standard Review Plan 6.2.4. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, and a blind flange. Condition C is written to specifically address those penetration flow paths in a closed system. Therefore, Condition C is not considered a LOF condition because the closed system acts as a reliable boundary for containment. Additionally, precedent for determining that Condition C should not be treated as a LOF exists in the approval for TSTF-505-A Revision 2, Risk-Informed Extended Completion Times. This revision to TS 3.6.3 Condition C is reflected in Enclosures 2 and 3 of this letter.

While correcting End states for loss of function Conditions, SNC identified a required clarification for TS 3.7.8 Condition D. The words “for reasons other than Condition B” were added to the entry Condition to clarify that this condition does not apply when one service water system (SWS) automatic turbine building isolation valve is inoperable in each SWS train. While having one SWS automatic turbine building isolation valve inoperable in each train does cause both trains to be inoperable, there is no loss of function. As discussed in the TS Bases 3.7.8 Condition B, with the unit in this condition, the remaining operable SWS turbine building isolation valves in each train are adequate to perform the SWS non-essential load isolation function. Because Condition B is not a loss of function Condition, and it currently has a 72-hour completion time, SNC requests to apply the RICT program to TS 3.7.8 Condition B. All 4 of these SWS turbine building isolation valves are modeled in the PRA for Plant Farley and the PRA success criteria is the same as the design basis criteria: one of two trains of SWS must be available. This request is reflected in the markups and clean type pages in Enclosures 2 and 3 of this letter.

Joseph M. Farley Nuclear Plant Units 1&2
Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications, SNC Response to NRC Request for Additional Information (RAI)

Enclosure 2

Operating License and Technical Specifications Markups
(50 pages included)

Operating Licenses Markups

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~221~~, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission.

a. Southern Nuclear shall not operate the reactor in Operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

b. Deleted per Amendment 13

c. Deleted per Amendment 2

d. Deleted per Amendment 2

e. Deleted per Amendment 152

Deleted per Amendment 2

f. Deleted per Amendment 158

g. Southern Nuclear shall maintain a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical parameters and control points for these parameters;

2) Identification of the procedures used to quantify parameters that are critical to control points;

3) Identification of process sampling points;

4) A procedure for the recording and management of data;

5) Procedures defining corrective actions for off control point chemistry conditions; and

6) A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.

h. The Additional Conditions contained in Appendix C, as revised through Amendment No. ~~146~~, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the additional conditions.

i. Deleted per Amendment 152

(4) Fire Protection

Southern Nuclear Operating Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment requests dated September 25, 2012; April 25, 2016; and supplements dated December 20, 2012; September 16, 2013; October 30, 2013; November 12, 2013; April 23, 2014; May 23, 2014; July 3, 2014; August 11, 2014; August 29, 2014; October 13, 2014; January16, 2015, and August 11, 2017, as approved in the safety evaluation reports dated March 10, 2015, October 17, 2016, and November 1, 2017. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

a. Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at Farley. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

Insert at end of the operating license

APPENDIX C

ADDITIONAL CONDITIONS
FACILITY OPERATING LICENSE NO. NPF-2

Amendment Number	Additional Condition	Condition Completion Date
	Southern Nuclear Operating Company (SNC) is approved to implement the Risk Informed Completion Time (RICT) Program as specified in the license amendment request submittal dated July 27, 2018, as supplemented on the following dates:	Concurrent with the implementation of the Risk Informed Completion Time Program
	Updates from the Findings and Observation resolutions of the Internal Events Internal Flooding Probabilistic Risk Assessment (PRA) model shall be incorporated into the Fire PRA per the internal SNC PRA configuration process, prior to implementation of the RICT program.	
	The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC for generic use. If the licensee wishes to change its methods, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval, via a license amendment.	

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license.

(3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 2775 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~218~~, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Deleted per Amendment 144

(4) Deleted per Amendment 149

(5) Deleted per Amendment 144

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

(8) Deleted per Amendment 144
(9) Deleted per Amendment 144
(10) Deleted per Amendment 144
(11) Deleted per Amendment 144
(12) Deleted per Amendment 144
(13) Deleted per Amendment 144
(14) Deleted per Amendment 144
(15) Deleted per Amendment 144
(16) Deleted per Amendment 144
(17) Deleted per Amendment 144
(18) Deleted per Amendment 144
(19) Deleted per Amendment 144
(20) Deleted per Amendment 144
(21) Deleted per Amendment 144

(22) Additional Conditions

The Additional conditions contained in Appendix C, as revised through Amendment No. ~~137~~, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the additional conditions.

(23) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed license. Until that update is complete, Southern Nuclear may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Southern Nuclear evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements of that section.

Insert at end of the operating license

APPENDIX C

ADDITIONAL CONDITIONS
FACILITY OPERATING LICENSE NO. NPF-8

Amendment Number	Additional Condition	Condition Completion Date
	Southern Nuclear Operating Company (SNC) is approved to implement the Risk Informed Completion Time (RICT) Program as specified in the license amendment request submittal dated July 27, 2018, as supplemented on the following dates:	Concurrent with the implementation of the Risk Informed Completion Time Program
	Updates from the Findings and Observation resolutions of the Internal Events Internal Flooding Probabilistic Risk Assessment (PRA) model shall be incorporated into the Fire PRA per the internal SNC PRA configuration process, prior to implementation of the RICT program.	
	The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC for generic use. If the licensee wishes to change its methods, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval, via a license amendment.	

Technical Specifications Markups

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

Insert 1

Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B **is** exited and operation may **continue** in accordance with Condition A, **provided** the Completion Time for Required Action A.2 has not expired.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

RAI Revision: Per RAI 20 response, in Condition B, deleted Note 2 and modified Note 1. Modified wording in paragraphs 2 and 3 to reflect the Note changes

INSERT 1

EXAMPLE 1.3-8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when the second subsystem is intentionally made inoperable. ---------- Two subsystems inoperable.	B.1 Restore one subsystem to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

When a subsystem is declared inoperable, Condition A is entered. The 7 day Completion Time may be applied as discussed in Example 1.3-2. However, the licensee may elect to apply the Risk Informed Completion Time Program which permits calculation of a Risk Informed Completion Time (RICT) that may be used to complete the Required Action beyond the 7 day Completion Time. The RICT cannot exceed 30 days. After the 7 day Completion Time has expired, the subsystem must be restored to OPERABLE status within the RICT or Condition C must also be entered.

If a second subsystem is declared inoperable, Condition B must also be entered. The Condition is modified by a Note. The Note states that RICT entry is not permitted for this loss of function Condition when the second subsystem is intentionally made inoperable. RICT program entry is only allowed if one subsystem is inoperable for any reason and the second subsystem is found to be inoperable, or if both subsystems are found to be inoperable at the same time. If Condition B is entered and RICT entry is not permitted, at least one subsystem must be restored to OPERABLE status within 1 hour. If one subsystem is not restored within one hour, Condition C must also be entered.

The licensee may be able to apply a RICT to extend the Completion Time beyond 1 hour, but not longer than 24 hours, if the requirements of the Risk Informed Completion Time Program are met. If two subsystems are inoperable and RICT entry is permitted, at least one subsystem must be restored within the calculated RICT. If one subsystem cannot be restored within the calculated RICT, Condition C must also be entered.

The Risk Informed Completion Time Program requires recalculation of the RICT to reflect changing plant conditions. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

If the 7 day Completion Time clock of Condition A or the 1 hour Completion Time clock of Condition B have expired and subsequent changes in plant conditions result in exiting the applicability of the Risk Informed Completion Time Program without restoring the inoperable subsystem to OPERABLE status, Condition C is also entered and the Completion Time clocks for Required Actions C.1 and C.2 start.

If the RICT expires or is recalculated to be less than the elapsed time since the Condition was entered and the inoperable subsystem has not been restored to OPERABLE status, Condition C is also entered and the Completion Time clocks for Required Actions C.1 and C.2 start. If the inoperable subsystems are restored to OPERABLE status after Condition C is entered, Conditions A, B, and C are exited, and therefore, the Required Actions of Condition C may be terminated.

RAI Revision: No change

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > the Low Temperature Overpressure Protection (LTOP) System applicability temperature specified in the PTLR.

NOTE

The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

Insert 2

Insert 3

ACTIONS

	CONDITION		REQUIRED ACTION	COMPLETION TIME
A.	One pressurizer safety valve inoperable.	A.1	Restore valve to OPERABLE status.	15 minutes
B.	Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 AND B.2	Be in MODE 3. Be in MODE 4 with any RCS cold leg temperatures ≤ the LTOP System applicability temperature specified in the PTLR.	6 hours 12 hours

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20

INSERT 2

---------NOTE----------
RICT entry is not
permitted for this loss
of function Condition
when a pressurizer
safety valve is
intentionally made
inoperable.

RAI Revision: No change

INSERT 3

OR

In accordance with the Risk Informed Completion Time Program

RAI Revision: No change

Insert 4

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. Two block valves inoperable.	F.1	Place associated PORVs in manual control.	1 hour
	AND		
	F.2	Restore one block valve to OPERABLE status.	2 hours
G. Required Action and associated Completion Time of Condition F not met.	G.1	Be in MODE 3.	6 hours
	AND		
	G.2	Be in MODE 4.	12 hours

Insert 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	----------NOTES---------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. ---------- Perform a complete cycle of each block valve.	In accordance with the Surveillance Frequency Control Program

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20

INSERT 4

---------NOTE----------
RICT entry is not
permitted for this
loss of function
Condition when a
second block valve
is intentionally made
inoperable.

RAI Revision: No change

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Three ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > 1000 psig.

NOTE
In MODE 3, with RCS pressure > 1000 psig, the accumulators may be inoperable for up to 12 hours to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	24 hours
~~C.~~ D ~~Required Action and associated~~ Completion Time ~~of Condition A or B~~ B, or C not met.	~~C~~.1 Be in MODE 3.	6 hours
	AND	
	~~C~~.2 Reduce RCS pressure to ≤ 1000 psig.	12 hours
~~D.~~ ~~Two or more accumulators inoperable.~~	~~D.1 Enter LCO 3.0.3.~~	~~Immediately~~

Insert 5

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20

INSERT 5

C.	----------NOTE---------- RICT entry not permitted for this loss of function Condition when two or more ECCS accumulators are intentionally made inoperable. ---------- Two or more accumulators inoperable for reasons other than boron concentration not within limits.	C.1	Restore accumulators to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

RAI Revision: No change

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

~~NOTES~~

~~1. RWST piping may be unisolated from non safety related piping for ≤ 4 hours under administrative controls to perform SR 3.5.4.3.*~~

~~2. RWST piping may be unisolated from non safety related piping for ≤ 30 days per fuel cycle under administrative controls for filtration or silica removal.*~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour

Insert 6

Insert 3

~~*These Notes can only be applied during the next two fuel Cycles for each Unit. These Notes cannot be used after Refueling Outages 1R26 (Spring 2015) and 2R24 (Spring 2016).~~

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20

INSERT 6

---------NOTE----------
RICT entry not permitted for this loss of function Condition when the RWST is intentionally made inoperable.

RAI Revision: Insert 8 was deleted as described in SNC Supplemental Request.

Insert 7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ~~----------NOTE----------~~ ~~Only applicable to penetration flow paths with two containment isolation valves.~~ ---------- One or more penetration flow paths with two containment isolation valves inoperable except for purge valve penetration leakage not within limit.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. ----------NOTE---------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ---------- One or more penetration flow paths with one containment isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours
	AND	
	C.2 ----------NOTE---------- Isolation devices in high radiation areas may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days

Insert 3

RAI Revision: Note 2 revised and Note 3 deleted in response to RAI 20

INSERT 7

----------NOTES----------

1. Only applicable to penetration flow paths with two containment isolation valves.
2. RICT entry is not permitted when the second containment isolation valve is intentionally made inoperable.

RAI Revision: Moved Insert 9 to next page to apply an End State of Mode 5 for Condition F

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One containment spray train inoperable.	A.1	Restore containment spray train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	54 hours
C. One containment cooling train inoperable.	C.1	Restore containment cooling train to OPERABLE status.	7 days

Insert 3

RAI Revision: Removed Insert 3 from Condition D in response to RAI 15a

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two containment cooling trains inoperable.	D.1 Restore one containment cooling train to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. AND E.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours
~~F. Two containment spray trains inoperable.~~ ~~OR~~ Any combination of three or more trains inoperable.	~~F.1 Enter LCO 3.0.3.~~	~~Immediately~~

Insert 9

Insert 10

G

1 hour

Insert 3

Restore required trains to OPERABLE status.

Insert 11

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	----------NOTE---------- Not required to be met for system vent flow paths opened under administrative control. ---------- Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20

INSERT 9

F. ---------NOTE---------- RICT entry is not permitted for this loss of function Condition when a second containment spray train is intentionally made inoperable. ---------- Two containment spray trains inoperable.	F.1 Restore one containment spray train to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20

INSERT 10

---------NOTE----------
RICT entry is not
permitted for this loss
of function Condition
when a third train is
intentionally made
inoperable.

RAI Revision: Added "Condition F" to entry conditions

INSERT 11

H.	Required Action and associated Completion Time of Condition F or G not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 5.	6 hours 36 hours

RAI Revision: No change

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs per steam line shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when one MSIV in each steam line is closed.

ACTIONS

NOTE
Separate Condition entry is allowed for each steam line.

Insert 3

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more steam lines with one MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	72 hours
B. One or more steam lines with two MSIVs inoperable in MODE 1.	B.1 Restore one MSIV to OPERABLE status in affected steam line.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 2.	6 hours
D. One or more steam lines with one MSIV inoperable in MODE 2 or 3.	D.1 Verify one MSIV closed in affected steam line.	7 days AND Once per 7 days thereafter

Insert 12

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20

INSERT 12

----------NOTE----------

RICT entry is not permitted for this loss of function condition when a second MSIV, in one or more steam lines, is intentionally made inoperable.

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Relief Valves (ARVs)

LCO 3.7.4 Three ARV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

Insert 13

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ARV line inoperable.	A.1 Restore required ARV line to OPERABLE status.	7 days
~~B. Two or more required ARV lines inoperable.~~	~~B.1 Restore all but one ARV line to OPERABLE status.~~	~~24 hours~~
~~C~~. Required Action and associated Completion Time not met.	~~C~~.1 Be in MODE 3. AND ~~C~~.2 Be in MODE 4.	6 hours 18 hours

Insert 3

D

D

of Condition A, B, or C

RAI Revision: Insert 13 revised per response in RAI 15b. Condition C removed from the scope of the RMTS program.

INSERT 13

B.	Two required ARV lines Inoperable	C.1	Restore one required ARV line to OPERABLE status.	24 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
C.	Three required ARV lines inoperable	C.1	Restore one required ARV line to OPERABLE status	24 hours

RAI Revision: "Required action ...Condition C not met" moved from Condition B to Condition D to apply an End State of Mode 5

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," for residual heat removal loops made inoperable by CCW. ---------- Restore CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	12 hours

Insert 3

Insert 16

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20.
Added Condition D for Condition C not met to apply an End State of Mode 5

INSERT 16

C.	----------NOTE---------- RICT entry not permitted for this loss of function Condition when the second CCW train is intentionally made inoperable. ---------- Two CCW trains inoperable	C.1	Restore one CCW train to OPERABLE status	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program
D.	Required Action and associated Completion Time of Condition C not met.	D.1 <u>AND</u> D.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

RAI Revision: Moved Insert 17 down to apply an End State of Mode 5. Per SNC supplemental request, 3.7.8 Condition B is being added to the scope of this license amendment request.

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SWS train inoperable.	A.1 ----------NOTES---------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources — Operating," for emergency diesel generator made inoperable by SWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," for residual heat removal loops made inoperable by SWS. ---------- Restore SWS train to OPERABLE status.	72 hours~~*~~
B. One SWS automatic turbine building isolation valve inoperable in each SWS train.	B.1 Restore both inoperable turbine building isolation valves to OPERABLE status.	72 hours

Insert 3

~~* For the FNP Unit 2 October 06, 2009 entry into Technical Specification 3.7.8, the Service Water Train A may be inoperable for a period not to exceed 7 days provided that during the extended completion time for Train A, two Train A pumps are available (OPERABLE except during a seismic event).~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

Insert 17

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	----------NOTE---------- Isolation of SWS flow to individual components does not render the SWS inoperable. ---------- Verify each accessible SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Verify each SWS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.4	Verify the integrity of the SWS buried piping by visual inspection of the ground area.	In accordance with the Surveillance Frequency Control Program

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20. Condition E added to apply an End State of Mode 5. Added the following words to Condition D: "for reasons other than Condition B"

INSERT 17

D. ---------NOTE--------- RICT entry not permitted for this loss of function Condition when the second SWS train is intentionally made inoperable. ---------- Two SWS trains inoperable for reasons other than Condition B.	D.1 Restore one SWS train to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
E. Required Action and associated Completion Time of Condition D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

3.7 PLANT SYSTEMS

3.7.19 Engineered Safety Feature (ESF) Room Coolers

LCO 3.7.19 ESF Room Coolers shall be OPERABLE.

APPLICABILITY: When associated ESF equipment is required to be OPERABLE.

ACTIONS

NOTE
Separate Condition entry is allowed for each ESF Room Cooler subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ESF Room Cooler subsystem Train inoperable.	A.1 Restore the affected ESF Room Cooler subsystem Train to OPERABLE status.	72 hours
~~B.~~ C Required Action and associated Completion Time of Condition A or B not met. ~~OR~~ ~~Two trains of the same ESF Room Cooler subsystem inoperable.~~	~~B.~~1 Be in MODE 3. AND ~~B.~~2 Be in MODE 5.	6 hours 36 hours

Insert 21

Insert 3

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20

INSERT 21

B.	----------NOTE------- RICT entry is not permitted for this loss of function Condition when a second train of the same ESF Room Cooler subsystem inoperable is intentionally made inoperable. ---------- Two trains of the same ESF Room Cooler subsystem inoperable	B.1	Restore one of the same ESF Room Cooler subsystem to OPERABLE status	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

RAI Revision: Added Insert 21a as discussed in RAI 15c.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required offsite circuit inoperable. AND One DG set inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems — Operating," when Condition D is entered with no AC power source to any train. ----------	
	D.1 Restore required offsite circuit to OPERABLE status.	24 hours
Insert 3	OR	
	D.2 Restore DG set to OPERABLE status.	24 hours
~~E. Two DG sets inoperable.~~ Insert 21a	~~E.1 Restore one DG set to OPERABLE status.~~	~~2 hours if all three DGs are inoperable~~ ~~OR~~ ~~8 hours if DG 1-2A and DG 1(2)B are inoperable~~ ~~OR~~ ~~24 hours if DG 1C and DG 1(2)B are inoperable~~
~~F.~~H. Required Action and associated Completion Time of Condition C, ~~or~~ E, F, or G not met.	~~F.1~~ H.1 Be in MODE 3.	6 hours

RAI Revision: Moved Insert 22 down and added Condition L to apply an End State of Mode 5 when Condition K is not met.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~G~~. One automatic load sequencer inoperable.	~~G~~.1 Restore automatic load sequencer to OPERABLE status.	12 hours Insert 3
~~H~~. Required Action and associated Completion Time of Condition A, B, D, ~~or G~~ not met. or I J Insert 22	~~H~~.1 Be in MODE 3. AND ~~H~~.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours
~~I. Three or more required AC sources inoperable.~~	~~I.1 Enter LCO 3.0.3~~.	~~Immediately~~

I

RAI Revision: Added Insert 21a as discussed in RAI 15c. Created note consistent with RAI 20 response.

INSERT 21A

E.	DG 1C is inoperable AND DG Set B inoperable	E.1	Restore one DG set to OPERABLE status.	24 hours
F.	----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a second DG set is intentionally made inoperable. ---------- DG 1-2A is inoperable AND DG Set B inoperable	F.1	Restore one DG set to OPERABLE status.	8 hours OR In accordance with the Risk Informed Completion Time Program
G.	----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a second DG set is intentionally made inoperable ---------- DG 1C is inoperable AND DG 1-2A is inoperable AND DG Set B inoperable	G.1	Restore one DG set to OPERABLE status.	2 hours OR In accordance with the Risk Informed Completion Time Program

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20. Added Condition L to apply a Mode 5 End State when Condition K is not met.

INSERT 22

K. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a third AC source is intentionally made inoperable. ---------- Three or more required AC sources inoperable.	K.1 Restore required AC sources to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
L. Required Action and associated Completion Time of Condition K not met.	L.1 Be in MODE 3. AND L.2 Be in MODE 5.	6 hours 36 hours

RAI Revision: No change

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One required SWIS DC electrical power subsystem inoperable. OR Required Action and associated Completion Time of Condition D not met.	E.1 Declare the associated Service Water System train inoperable.	Immediately

Insert 23

Insert 24

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 127.8 V on float charge.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify post-to-post battery connection resistance of each cell-to-cell and terminal connection is ≤ 150 microhms for the Auxiliary Building batteries and ≤ 1500 microhms for the SWIS batteries.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration.	In accordance with the Surveillance Frequency Control Program

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20

INSERT 23

F. ----------NOTES---------- RICT entry not permitted for this loss of function Condition when a second DC power electrical subsystem is intentionally removed from service. ---------- Two or more DC electrical subsystems inoperable that result in a loss of function	F.1 Restore required DC electrical subsystems to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

RAI Revision: Modified to apply Mode 5 as the End State for Condition F not met.

INSERT 24

G.	Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 5.	6 hours 36 hours

RAI Revision: Moved Insert 25 down to apply a Mode 5 End State to Condition C

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters — Operating

LCO 3.8.7 The required Train A and Train B inverters shall be OPERABLE.

----------NOTE----------
Two inverters may be disconnected from their associated DC bus for ≤ 24 hours to perform an equalizing charge on their associated common battery, provided:

a. The associated AC vital buses are energized from their Class 1E constant voltage source transformers; and

b. All other AC vital buses are energized from their associated OPERABLE inverters.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 ----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any vital bus de-energized. ---------- Restore inverter to OPERABLE status.	24 hours

Insert 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	12 hours

of Condition A

Insert 25

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to required AC vital buses.	In accordance with the Surveillance Frequency Control Program

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20. Applied a Mode 5 End State to Condition C

INSERT 25

C.	----------NOTES---------- RICT entry is not permitted for this loss of function Condition when the second required inverter is intentionally made inoperable. ---------- Two or more required inverters inoperable.	D.1	Restore required inverters to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program
D.	Required Action and associated Completion Time of Condition C not met.	D.1 <u>AND</u> D.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more AC electrical power distribution subsystems inoperable for reasons other than Condition A, B, or C.	D.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours Insert 3
E. One or more AC vital buses inoperable.	E.1 Restore AC vital bus subsystem(s) to OPERABLE status.	8 hours Insert 3
F. One Auxiliary Building DC electrical power distribution subsystem inoperable.	F.1 Restore Auxiliary Building DC electrical power distribution subsystem to OPERABLE status.	2 hours Insert 3
G. Required Action and associated Completion Time of Condition D, E, or F not met.	G.1 Be in MODE 3. AND G.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours
H. One Service Water Intake Structure (SWIS) DC electrical power distribution subsystem inoperable. Insert 26	H.1 Declare the associated Service Water train inoperable.	Immediately
~~I. Two trains with inoperable distribution subsystems that result in a loss of safety function.~~	~~I.1 Enter LCO 3.0.3.~~	~~Immediately~~

RAI Revision: Note 1 revised and Note 2 deleted in response to RAI 20.
Applied a Mode 5 End State to Condition I

INSERT 26

I.	----------NOTES---------- RICT entry is not permitted for this loss of function condition when two or more electrical power distribution trains are intentionally made inoperable. ---------- Two trains with inoperable electrical distribution subsystems that result in a loss of function.	I.1	Restore one train to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
J.	Required Action and associated Completion Time of Condition I not met.	J.1 AND J.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours

Joseph M. Farley Nuclear Plant Units 1&2
Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications, SNC Response to NRC Request for Additional Information (RAI)

Enclosure 3

Operating License and Technical Specifications Clean Type Pages
(40 pages included)

Operating Licenses Clean Type Pages

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. , are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission.

a. Southern Nuclear shall not operate the reactor in Operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

b. Deleted per Amendment 13

c. Deleted per Amendment 2

d. Deleted per Amendment 2

e. Deleted per Amendment 152

Deleted per Amendment 2

f. Deleted per Amendment 158

g. Southern Nuclear shall maintain a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical parameters and control points for these parameters;

2) Identification of the procedures used to quantify parameters that are critical to control points;

3) Identification of process sampling points;

4) A procedure for the recording and management of data;

5) Procedures defining corrective actions for off control point chemistry conditions; and

6) A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events required to initiate corrective action.

h. The Additional Conditions contained in Appendix C, as revised through Amendment No. , are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the additional conditions.

i. Deleted per Amendment 152

(4) Fire Protection

Southern Nuclear Operating Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment requests dated September 25, 2012; April 25, 2016; and supplements dated December 20, 2012; September 16, 2013; October 30, 2013; November 12, 2013; April 23, 2014; May 23, 2014; July 3, 2014; August 11, 2014; August 29, 2014; October 13, 2014; January16, 2015, and August 11, 2017, as approved in the safety evaluation reports dated March 10, 2015, October 17, 2016, and November 1, 2017. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

a. Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at Farley. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

APPENDIX C

ADDITIONAL CONDITIONS
FACILITY OPERATING LICENSE NO. NPF-2

Amendment Number	Additional Condition	Condition Completion Date
	Southern Nuclear Operating Company (SNC) is approved to implement the Risk Informed Completion Time (RICT) Program as specified in the license amendment request submittal dated July 27, 2018, as supplemented on the following dates:	Concurrent with the implementation of the Risk Informed Completion Time Program
	Updates from the Findings and Observation resolutions of the Internal Events Internal Flooding Probabilistic Risk Assessment (PRA) model shall be incorporated into the Fire PRA per the internal SNC PRA configuration process, prior to implementation of the RICT program.	
	The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC for generic use. If the licensee wishes to change its methods, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval, via a license amendment.	

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license.

(3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 2775 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. , are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Deleted per Amendment 144

(4) Deleted per Amendment 149

(5) Deleted per Amendment 144

(b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.18.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from February 8, 2016, the date of the most recent successful tracer gas test, as stated in the August 25, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

(c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.18.d, shall be within 24 months, plus the 180 days allowed by SR 3.0.2, as measured from July 11, 2015, the date of the most recent successful pressure measurement test, or within 180 days if not performed previously.

(8) Deleted per Amendment 144
(9) Deleted per Amendment 144
(10) Deleted per Amendment 144
(11) Deleted per Amendment 144
(12) Deleted per Amendment 144
(13) Deleted per Amendment 144
(14) Deleted per Amendment 144
(15) Deleted per Amendment 144
(16) Deleted per Amendment 144
(17) Deleted per Amendment 144
(18) Deleted per Amendment 144
(19) Deleted per Amendment 144
(20) Deleted per Amendment 144
(21) Deleted per Amendment 144

(22) Additional Conditions

The Additional conditions contained in Appendix C, as revised through Amendment No. , are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the additional conditions.

(23) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed license. Until that update is complete, Southern Nuclear may make changes to the programs and activities described in the supplement without prior Commission approval, provided that Southern Nuclear evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements of that section.

APPENDIX C

ADDITIONAL CONDITIONS
FACILITY OPERATING LICENSE NO. NPF-8

Amendment Number	Additional Condition	Condition Completion Date
	Southern Nuclear Operating Company (SNC) is approved to implement the Risk Informed Completion Time (RICT) Program as specified in the license amendment request submittal dated July 27, 2018, as supplemented on the following dates:	Concurrent with the implementation of the Risk Informed Completion Time Program
	Updates from the Findings and Observation resolutions of the Internal Events Internal Flooding Probabilistic Risk Assessment (PRA) model shall be incorporated into the Fire PRA per the internal SNC PRA configuration process, prior to implementation of the RICT program.	
	The risk assessment approach and methods, shall be acceptable to the NRC, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience of the plant as specified in RG 1.200. Methods to assess the risk from extending the completion times must be PRA methods accepted as part of this license amendment, or other methods approved by the NRC for generic use. If the licensee wishes to change its methods, and the change is outside the bounds of this license condition, the licensee will seek prior NRC approval, via a license amendment.	

Technical Specifications Clean Type Pages

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

(continued)

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Restore subsystem to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. ----------NOTE--------- RICT entry is not permitted for this loss of function Condition when the second subsystem is intentionally made inoperable. ---------- Two subsystems inoperable.	B.1 Restore one subsystem to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

(continued)

EXAMPLES

EXAMPLE 1.3-8 (continued)

When a subsystem is declared inoperable, Condition A is entered. The 7 day Completion Time may be applied as discussed in Example 1.3-2. However, the licensee may elect to apply the Risk Informed Completion Time Program which permits calculation of a Risk Informed Completion Time (RICT) that may be used to complete the Required Action beyond the 7 day Completion Time. The RICT cannot exceed 30 days. After the 7 day Completion Time has expired, the subsystem must be restored to OPERABLE status within the RICT or Condition C must also be entered.

If a second subsystem is declared inoperable, Condition B must also be entered. The Condition is modified by a Note. The Note states that RICT entry is not permitted for this loss of function Condition when the second subsystem is intentionally made inoperable. RICT program entry is only allowed if one subsystem is inoperable for any reason and the second subsystem is found to be inoperable, or if both subsystems are found to be inoperable at the same time. If Condition B is entered and RICT entry is not permitted, at least one subsystem must be restored to OPERABLE status within 1 hour. If one subsystem is not restored within one hour, Condition C must also be entered.

The Licensee may be able to apply a RICT to extend the Completion Time beyond 1 hour, but not longer than 24 hours, if the requirements of the Risk Informed Completion Time Program are met. If two subsystems are inoperable and RICT entry is permitted, at least one subsystem must be restored within the calculated RICT. If one subsystem cannot be restored within the calculated RICT, Condition C must also be entered.

The Risk Informed Completion Time Program requires recalculation of the RICT to reflect changing plant conditions. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

If the 7 day Completion Time clock of Condition A or the 1 hour Completion Time clock of Condition B have expired and subsequent changes in plant conditions result in exiting the applicability of the Risk Informed Completion Time Program without restoring the inoperable subsystem to OPERABLE status, Condition C is also entered and the Completion Time clocks for Required Actions C.1 and C.2 start.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-8 (continued)

If the RICT expires or is recalculated to be less than the elapsed time since the Condition was entered and the inoperable subsystem has not been restored to OPERABLE status, Condition C is also entered and the Completion Time clocks for Required Actions C.1 and C.2 start. If the inoperable subsystems are restored to OPERABLE status after Condition C is entered, Conditions A, B, and C are exited, and therefore, the Required Actions of Condition C may be terminated.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > the Low Temperature Overpressure Protection (LTOP) System applicability temperature specified in the PTLR.

----------NOTE----------
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a pressurizer safety valve is intentionally made inoperable. ---------- One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes OR In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any RCS cold leg temperatures ≤ the LTOP System applicability temperature specified in the PTLR.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a second block valve is intentionally made inoperable. ---------- Two block valves inoperable.	F.1 Place associated PORVs in manual control. AND F.2 Restore one block valve to OPERABLE status.	1 hour 2 hours OR In accordance with the Risk Informed Completion Time Program
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	----------NOTES---------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. ---------- Perform a complete cycle of each block valve.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- RICT entry not permitted for this loss of function Condition when two or more ECCS accumulators are intentionally made inoperable. ---------- Two or more accumulators inoperable for reasons other than boron concentration not within limits.	C.1 Restore accumulators to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Reduce RCS pressure to ≤ 1000 psig.	6 hours 12 hours

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. <u>OR</u> RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. ----------NOTE---------- RICT entry not permitted for this loss of function Condition when the RWST is intentionally made inoperable. ---------- RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ----------NOTE---------- 1. Only applicable to penetration flow paths with two containment isolation valves. 2. RICT entry is not permitted when the second containment isolation valve is intentionally made inoperable. ---------- One or more penetration flow paths with two containment isolation valves inoperable except for purge valve penetration leakage not within limit.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ---------- One or more penetration flow paths with one containment isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours OR In accordance with the Risk Informed Completion Time Program
	AND	
	C.2 ----------NOTE---------- Isolation devices in high radiation areas may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 54 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One containment cooling train inoperable.	C.1 Restore containment cooling train to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
D. Two containment cooling trains inoperable.	D.1 Restore one containment cooling train to OPERABLE status.	72 hours
E Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. AND E.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours
F. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a second containment spray train is intentionally made inoperable. ---------- Two containment spray trains inoperable.	F.1 Restore one containment spray train to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a third train is intentionally made inoperable. ---------- Any combination of three or more trains inoperable.	G.1 Restore required trains to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
H. Required Action and associated Completion Time of Condition F or G not met.	H.1 Be in MODE 3. AND H.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	----------NOTE---------- Not required to be met for system vent flow paths opened under administrative control. ---------- Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs per steam line shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when one MSIV in each steam line is closed.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each steam line.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more steam lines with one MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
B. ----------NOTE---------- RICT entry is not permitted for this loss of function condition when a second MSIV, in one or more steam lines, is intentionally made inoperable. ---------- One or more steam lines with two MSIVs inoperable in MODE 1.	B.1 Restore one MSIV to OPERABLE status in affected steam line.	4 hours OR In accordance with the Risk Informed Completion Time Program

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Relief Valves (ARVs)

LCO 3.7.4 Three ARV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ARV line inoperable.	A.1 Restore required ARV line to OPERABLE status.	7 days OR In accordance with the Risk Informed Completion Time Program
B. Two required ARV lines inoperable.	B.1 Restore one required ARV line to OPERABLE status.	24 hours OR In accordance with the Risk Informed Completion Time Program
C. Three required ARV lines inoperable.	C.1 Restore one required ARV line to OPERABLE status.	24 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 18 hours

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CCW train inoperable.	A.1 ----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," for residual heat removal loops made inoperable by CCW. ---------- Restore CCW train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program.
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- RICT entry not permitted for this loss of function Condition when the second CCW train is intentionally made inoperable. ---------- Two CCW trains inoperable.	C.1 Restore one CCW train to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program.
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	----------NOTE---------- Isolation of CCW flow to individual components does not render the CCW System inoperable. ---------- Verify each accessible CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.2	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SWS train inoperable.	A.1 ----------NOTES---------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources — Operating," for emergency diesel generator made inoperable by SWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops — MODE 4," for residual heat removal loops made inoperable by SWS. ---------- Restore SWS train to OPERABLE status.	72 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One SWS automatic turbine building isolation valve inoperable in each SWS train.	B.1 Restore both inoperable turbine building isolation valves to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours
D. ----------NOTE---------- RICT entry not permitted for this loss of function Condition when the second SWS train is intentionally made inoperable. ---------- Two SWS trains inoperable for reasons other than Condition B.	D.1 Restore one SWS train to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
E. Required Action and associated Completion Time of Condition D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	----------NOTE---------- Isolation of SWS flow to individual components does not render the SWS inoperable. ---------- Verify each accessible SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.2	Verify each SWS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.3	Verify each SWS pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.8.4	Verify the integrity of the SWS buried piping by visual inspection of the ground area.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.19 Engineered Safety Feature (ESF) Room Coolers

LCO 3.7.19 ESF Room Coolers shall be OPERABLE.

APPLICABILITY: When associated ESF equipment is required to be OPERABLE.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each ESF Room Cooler subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ESF Room Cooler subsystem Train inoperable.	A.1 Restore the affected ESF Room Cooler subsystem Train to OPERABLE status.	72 hours OR In accordance with the Risk Informed Completion Time Program
B. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a second train of the same ESF Room Cooler subsystem inoperable is intentionally made inoperable. ---------- Two trains of the same ESF Room Cooler subsystem inoperable.	B.1 Restore one of the same ESF Room Cooler subsystems to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.19.1	Verify each ESF Room Cooler system manual valve servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.19.2	Verify each ESF Room Cooler fan starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One required offsite circuit inoperable. AND One DG set inoperable.	----------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems — Operating," when Condition D is entered with no AC power source to any train. ----------	
	D.1 Restore required offsite circuit to OPERABLE status. OR	24 hours OR In accordance with the Risk Informed Completion Time Program
	D.2 Restore DG set to OPERABLE status.	24 hours OR In accordance with the Risk Informed Completion Time Program
E. DG 1C is inoperable. AND DG Set B inoperable.	E.1 Restore one DG set to OPERABLE status.	24 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a second DG set is intentionally made inoperable. ---------- DG 1-2A is inoperable. AND DG Set B inoperable.	F.1 Restore one DG set to OPERABLE status.	8 hours OR In accordance with the Risk Informed Completion Time Program
G. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a second DG set is intentionally made inoperable. ---------- DG 1C is inoperable. AND DG 1-2A is inoperable. AND DG Set B inoperable.	G.1 Restore one DG set to OPERABLE status.	2 hours OR In accordance with the Risk Informed Completion Time Program
H. Required Action and associated Completion Time of Condition C, E, F, or G not met.	H.1 Be in MODE 3.	6 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. One automatic load sequencer inoperable.	I.1 Restore automatic load sequencer to OPERABLE status.	12 hours OR In accordance with the Risk Informed Completion Time Program
J. Required Action and associated Completion Time of Condition A, B, D, or I not met.	J.1 Be in MODE 3 AND J.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours
K. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when a third AC source is intentionally made inoperable ---------- Three or more required AC sources inoperable.	K.1 Restore required AC sources to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
L. Required Action and associated Completion Time of Condition K not met.	L.1 Be in MODE 3. AND L.2 Be in MODE 5.	6 hours 36 hours

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. ----------NOTE---------- RICT entry not permitted for this loss of function Condition when a second DC power electrical subsystem is intentionally removed from service. ---------- Two or more DC electrical subsystems inoperable that result in a loss of function.	F.1 Restore required DC electrical subsystems to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 127.8 V on float charge.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. OR Verify post-to-post battery connection resistance of each cell-to-cell and terminal connection is ≤ 150 microhms for the Auxiliary Building batteries and ≤ 1500 microhms for the SWIS batteries.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.4	Remove visible terminal corrosion, verify battery cell-to-cell and terminal connections are coated with anti-corrosion material.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.5	Verify post-to-post battery connection resistance of each cell-to-cell and terminal connection is ≤ 150 microhms for the Auxiliary Building batteries and ≤ 1500 microhms for the SWIS batteries	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.6	----------NOTE---------- This Surveillance may be performed in MODE 1, 2, 3, 4, 5, or 6 provided spare or redundant charger(s) placed in service are within surveillance frequency to maintain DC subsystem(s) OPERABLE. ---------- Verify each required Auxiliary Building battery charger supplies ≥ 536 amps at ≥ 125 V for ≥ 4 hours and each required SWIS battery charger supplies ≥ 3 amps at ≥ 125 V for ≥ 4 hours.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 ----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	6 hours 12 hours
C. ----------NOTE---------- RICT entry is not permitted for this loss of function Condition when the second required inverter is intentionally made inoperable. ---------- Two or more required inverters inoperable.	C.1 Restore required inverters to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to required AC vital buses.	In accordance with the Surveillance Frequency Control Program

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One or more AC electrical power distribution subsystems inoperable for reasons other than Condition A, B, or C.	D.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.	8 hours OR In accordance with the Risk Informed Completion Time Program
E. One or more AC vital buses inoperable.	E.1 Restore AC vital bus subsystem(s) to OPERABLE status.	8 hours OR In accordance with the Risk Informed Completion Time Program
F. One Auxiliary Building DC electrical power distribution subsystem inoperable.	F.1 Restore Auxiliary Building DC electrical power distribution subsystem to OPERABLE status.	2 hours OR In accordance with the Risk Informed Completion Time Program

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. Required Action and associated Completion Time of Condition D, E, or F not met.	G.1	Be in MODE 3.	6 hours
	AND		
	G.2	----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	12 hours
H. One Service Water Intake Structure (SWIS) DC electrical power distribution subsystem inoperable.	H.1	Declare the associated Service Water train inoperable.	Immediately
I. ----------NOTE---------- RICT entry is not permitted for this loss of function condition when two or more electrical power distribution trains are intentionally made inoperable. ---------- Two trains with inoperable electrical distribution subsystems that result in a loss of function.	I.1	Restore one train to OPERABLE status.	1 hour OR In accordance with the Risk Informed Completion Time Program
J. Required Action and associated Completion Time of Condition I not met.	J.1	Be in MODE 3.	6 hours
	AND		
	J.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program