NRR-DMPSPEm Resource

From: Williams, Shawn
Sent: Tuesday, March 12, 2019 2:09 PM
To: Coleman, Jamie Marquess
Cc: Burns, Pamela Diane; Jackson, Nicole D.
Subject: RE: Joseph M. Farley Nuclear Plant, Units 1 and 2 - Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4B" (EPID L-2018-LLA-0210)
Attachments: Farley 4B RAIs.docx

Dear Ms. Coleman,

By letter dated July 27, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18208A619), the Southern Nuclear Operating Company, Inc., (SNC) submitted an amendment request to revise the Joseph M. Farley Nuclear Plant, Units 1 and 2, Technical Specifications. The proposed amendment would modify TS requirements to permit 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.”

The U.S. Nuclear Regulatory Commission (NRC) staff conducted a regulatory audit on February 5-7, 2019 (ADAMS Accession No. ML19042A108), to determine any necessary requests for additional information. The NRC staff has determined that the attached additional information is needed in order to complete its review. During a clarification call on March 12, 2019, Ms. Burns of your staff agreed that SNC would respond within 30 days of the date of this e-mail.

Sincerely,

Shawn A. Williams, Senior Project Manager
Plant Licensing Branch, II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosure:
Request for Additional Information

cc w/encl: Listserv

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REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO ADOPT NEI 06-09, REVISION 0-A

RISK-INFORMED COMPLETION TIMES

SOUTHERN NUCLEAR OPERATING COMPANY

JOSEPH M. FARLEY NUCLEAR OPERATING COMPANY UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

By letter dated July 27, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18208A619) Southern Nuclear Operating Company (SNC, the licensee) submitted a license amendment request (LAR) to modify the Joseph M. Farley Nuclear Plant (Farley) Technical Specification (TS) requirements to permit the use of Risk-Informed Completion Times (RICT) in accordance with Nuclear Energy Institute's (NEI) 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines" (ADAMS Accession No. ML12286A322).

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the request and determined that additional information is necessary to complete the review to determine if the licensee's proposed change implements the guidance in NEI 06-09, Revision 0-A, as accepted in NRC staff's safety evaluation (SE) in letter dated May 17, 2007 (ADAMS Accession No. ML071200238).

Request for Additional Information (RAI) No. 1 – 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.

RAI No. 2 – 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.

RAI No. 3 – 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.

RAI No. 4 – 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.

RAI No. 5 – 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.

RAI No. 6 – Supplanted and Updated Fire PRA Guidance

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.

RAI No. 7 – 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 CRMP [Configuration Risk Management Program] 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.

RAI No. 8 – 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.

RAI 9 – 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>RAI No. 10 – Total Risk Estimates Against RG 1.174 Guidelines</u>

RG 1.174, Revision 3, provides the risk acceptance guidance for total CDF (1×10^{-4} per year) and LERF (1×10^{-5} per year). Enclosure 4 of the LAR shows the total CDF for Unit 1 to be 9.69×10^{-5} per year and for Unit 2 to be 9.22×10^{-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 1×10^{-4} and 1×10^{-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.

RAI No. 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).

RAI No. 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.

RAI No. 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.

RAI No. 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.

RAI No. 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.

RAI No. 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.

RAI No. 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.

RAI No. 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 <u>affected function(s) has no risk impact</u>, 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.

RAI No. 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.

RAI No. 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.