V. C. Summer Nuclear Station
Bradham Blvd & Hwy 215, Jenkinsville, SC 29065
Mailing Address:
P.O. Box 88, Jenkinsville, SC 29065
DominionEnergy.com

Dominion Energy®

May 22, 2019

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Serial No. 19-167
VCS-LIC/JRB R1
Docket No. 50-395
License No. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1
LICENSE AMENDMENT REQUEST – LAR-16-01490
NFPA 805 PROGRAM REVISIONS
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

By letter dated August 29, 2018 (Agencywide Document Access and Management System (ADAMS) Package Accession No ML 18242A658), South Carolina Electric and Gas (SCE&G) submitted a license amendment request (LAR) for the Virgil C. Summer Nuclear Station Unit 1 (VCSNS), to make changes to its approved Fire Protection Program (FPP) under 10 CFR 50.48(c). By email dated April 4, 2019 (ML 19095A653), the Nuclear Regulatory Commission (NRC) staff provided a request for additional information (RAI).

The VCSNS responses to PRA RAI 03 (Revised), PRA RAI 04, and PRA RAI 05 are provided in the attachment to this letter.

On April 29, 2019 South Carolina Electric & Gas Company (SCE&G), which is authorized under Facility Operating License NPF-12 to operate and possess Virgil C. Summer Nuclear Station Unit 1, changed its name to Dominion Energy South Carolina, Inc. SCE&G will be requesting a license amendment to reflect this name change, in the near future.

Should you have any questions, please call Mr. Michael Moore at (803) 345-4752.

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

5/22/19
Executed on

George A. Lippard
Site Vice President
V.C. Summer Nuclear Station

Table PRA RAI 03-1 and Table PRA RAI 05b-1 transmitted herewith contains Sensitive Information. When separated from Table PRA RAI 03-1 and Table PRA RAI 05b-1, this document is decontrolled.

~~Sensitive Information – Withhold from Public Disclosure Under 10 CFR 2.390~~

Commitments made in this letter:

1. VCSNS will not credit the RCP abeyance seal in its PRA models until the NRC accepts the RCP abeyance seal model. Within 60 days of NRC approval of LAR-16-01490, VCSNS will revise its PRA model control procedure to require NRC acceptance of an abeyance seal model prior to use.

Attachment: LAR-16-01490 – NFPA 805 Program Revisions Response to Request for Additional Information

cc: G. J. Lindamood – Santee Cooper
C. Haney – NRC Region II
S. A. Williams – NRC Project Mgr.
NRC Resident Inspector

ATTACHMENT

LAR-16-01490 – NFPA 805 Program Revisions
Response to Request for Additional Information

Virgil C. Summer Nuclear Station – Unit 1
South Carolina Electric and Gas

LICENSE AMENDMENT REQUEST LAR-16-01490
NFPA 805 PROGRAM REVISIONS
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

VIRGIL C. SUMMER NUCLEAR STATION UNIT 1

NRC Comment:

By letter dated August 29, 2018 (Agencywide Document Access and Management System (ADAMS) Package Accession No. ML18242A657), South Carolina Electric and Gas (SCE&G), submitted a license amendment request (LAR) for the Virgil C. Summer Nuclear Station, Unit 1 (VCSNS), to make changes to its approved fire protection program (FPP) under 10 CFR 50.48(c). In its LAR, the licensee proposed to make several changes to its FPP including changes to plant modifications, use of performance-based alternatives to the requirements of NFPA 805, Chapter 3, and several clarifications and editorial corrections.

Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment [PRA] Results for Risk-informed Activities," Revision 2 (ADAMS Accession No. ML090410014), provides guidance for addressing probabilistic risk assessment (PRA) acceptability including addressing the need for the PRA model to represent the as-built, as-operated plant. This regulatory guide provides one approach acceptable to the U.S. Nuclear Regulatory Commission (NRC) for determining the technical acceptability of the PRA model. Regulatory Guide 1.200 endorses, with certain clarifications and qualifications, Addendum A to the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" ("PRA Standard"). Section 4.2, "Licensee Submittal Documentation," of RG 1.200 states, in part, that the application should discuss the resolution of the open peer review facts and observations (F&Os) that are applicable to the parts of the PRA required for the application.

By e-mail dated March 13, 2019 (ADAMS Accession No. ML19072A144), the NRC issued three requests for additional information (RAI). Regarding PRA RAI No. 3, on March 18, 2019, SCE&G informed the NRC staff that that the abeyance seals were credited in the PRA model. Based on this information, the below contains a revised RAI No. 3 and two additional RAIs based on NRC staff's review of the F&Os.

PRA RAI 03 (Revised)

In Enclosure 1, Attachment 1 to the licensee's letter dated August 29, 2018, the licensee stated that PRA refinements were made to the reactor coolant pump (RCP) seal loss of coolant accident (LOCA) model based on the RCP seal upgrades. The LAR is not clear on whether these refinements go beyond those described in the NFPA 805 LAR, as supplemented (Approved in Amendment No. 199 – ADAMS Accession No. ML14287A289).

a) *Describe the PRA refinements to the RCP seal LOCA model and discuss whether the approach was used previously to support the NFPA 805 LAR or in another plant LAR and subsequent amendment and if these changes constitute a PRA upgrade as defined in ASME/ANS RA-Sa-2009, Section 1-5.4, as qualified by RG 1.200, Revision 2. Provide the basis for the conclusion regarding whether the refinements are considered a PRA upgrade.*

If the refinements are an upgrade, provide the findings of the peer review(s) performed and associated disposition of the findings as it pertains to the impact on this LAR. Please indicate whether the abeyance seals were credited in the PRA model.

b) *If the abeyance seals were credited in the PRA model, then provide one of the following:*

   i) *Describe and justify the PRA methodology used to model the abeyance seals. Describe how crediting the abeyance seals impact the fire PRA results in Table W-3 of LAR Enclosure 1, Attachment 5 (e.g., describe and provide the results of a sensitivity study, including the total transition CDF, LERF, ΔCDF, and ΔLERF, that does not credit the abeyance seals). OR*

   ii) *Alternatively to part (i), provide updated risk results in Table W-3 of LAR Enclosure 1, Attachment 5 that does not credit the abeyance seals in the PRA and discuss how the updated risk results align with the risk acceptance guidelines of RG 1.205. Propose a mechanism that ensures an NRC accepted abeyance seal model is available before incorporation of an abeyance seal into the PRA model used for self-approval of post-transition changes.*

VCSNS Response:

RAI 03 a)

The fire probabilistic risk assessment (FPRA) that supports the VCSNS National Fire Protection Association (NFPA) Standard 805 Program NRC Safety Evaluation (SE), dated February 11, 2015 (ML14287A289), did not include credit for the N-9000 abeyance seal in its model of reactor coolant pump (RCP) seal failure. The FPRA that supports VCSNS License Amendment Request (LAR) LAR-16-01490 NFPA 805 Program Revisions dated August 29, 2018 (ML113320227) did include credit for the

N-9000 seal abeyance seal in its model of RCP seal failure. The RCP seal failure model was added to the Internal Events PRA model prior to the June 2016 Full Scope Peer Review. The Internal Events model was being updated in preparation for the Full Scope Peer Review, therefore a specific determination of upgrade versus maintenance was not considered. Documentation of the N-9000 RCP model was provided to the peer review team and documented in the Peer Review report. There were no Facts or Observations (F&Os) on the N-9000 RCP seal model.

RAI 03 Part b) discusses the VCSNS decision to remove credit for the N-9000 seal abeyance seal. Since VCSNS has elected to remove credit for the N-9000 seal abeyance seal, further discussion on the modeling of the N-9000 seal is not included here. Please see response to Part (b) for details of the N-9000 seal modeling going forward.

RAI 03 b)

VCSNS has elected to follow option (b)(ii) of the RAI. Credit for the N-9000 seal abeyance seal has been removed from the PRA model. (If in the future an NRC-accepted model of the abeyance seal is available, VCSNS may evaluate incorporating it into its PRA models at that time.)

RCP Seal Model

The key modeling details of the N-9000 RCP seals are as follows:

1. The RCP must be tripped within 60 minutes once seal injection is lost. This is modeled as a human error probability (HEP) in the PRA model. Failure to trip the RCP within 60 minutes leads directly to RCP seal LOCA, with a flow rate of 480 gpm per RCP.
2. For fire scenarios in which the ability to trip the RCPs from the main control room (MCR) is compromised, credit is taken for remotely tripping the RCPs at the power supply breakers. Timing of the remote trip is accounted for in the HEP, and supported by operator interviews and field walkthroughs by Operations.
3. Once an RCP is tripped, the probability of RCP seal LOCA is based on the three N-9000 seal cartridges. The failure is calculated consistent with WCAP-16175 for the Combustion Engineering plant applications.

Time Available for Operator Action to Trip RCPs

The N-9000 seal packages can withstand loss of seal injection and loss of thermal barrier cooling with the RCP running for some time without loss of seal integrity. The longer the seals can withstand loss of seal cooling, the more time is available for operators to trip the RCPs, and the lower the failure probability of that action. Dynamic loss-of-seal-cooling tests of N-9000 RCP seals have shown that RCPs can operate in excess of 60 minutes with no measurable change in seal leakage. These tests only credit the first two seal stages of three of the N-9000 seal package. The Human

Reliability Analysis (HRA) that supports the 2018 VCSNS FPRA models a 60-minute system time window for the operator action to trip RCPs. The 60-minute system time window is retained in the current FPRA. The 2018 RCP trip action uses the same HRA methods in the HRA Calculator as for the 2014 analysis. Therefore, the human reliability analysis of RCP trip associated with this RAI response does not constitute a PRA upgrade.

Updated Risk Results Without Credit for the RCP Abeyance Seal

In response to Part b(ii), updated risk results for Table W-3 of LAR-16-01490 Enclosure 1, Attachment 5 that do not credit the abeyance seals in the PRA are provided below in Table PRA RAI 03-1. In addition, Table PRA RAI 03-1 provides total plant risks from fire, internal events and flooding, and seismic hazards. The discussion below describes how the updated risk results align with the risk acceptance guidelines of Regulatory Guide (RG) 1.205.

Delta risk is calculated to compare the post-transition plant to the "compliant" plant, as defined in RG 1.205. Delta risk is evaluated to ensure that the guidelines of RG 1.205 and RG 1.174 are met. Table PRA RAI 03-1 provides delta risk for each plant fire area. From Table PRA RAI 03-1, the change in fire risk for the plant is as follows:

- Delta CDF = -5.5E-5 per reactor year
- Delta LERF = -1.4E-6 per reactor year

The delta risks are negative due to the way the compliant plant has been modeled, as discussed below. Because the delta risks are negative, they meet the guidelines of RG 1.174, as stated in the following excerpts from RG 1.174, Section 2.4:

> *If the application clearly shows a decrease in CDF, the change has satisfied the relevant principle of risk-informed regulation with respect to CDF.*
>
> *If the application clearly shows a decrease in LERF, the change has satisfied the relevant principle of risk-informed regulation with respect to LERF.*

With total CDF and LERF for fire, internal events and flooding, and seismic hazards satisfying CDF < 1E-04/yr and LERF < 1E-05/yr, the PRA risk results are within the range of Region II on the horizontal axes of RG 1.174, Figures 4 and 5. Meeting these risk criteria allows flexibility for future changes, as noted in the following excerpts from RG 1.174, Section 2.4:

> *When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year (i.e., the increase in CDF falls within Region II of Figure 4), applications are considered only if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year.*
>
> *When the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year (i.e., the increase in LERF falls within Region II of Figure 5), applications*

are considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year.

The model of the compliant plant for the delta risk calculation does not include credit for the Alternate Seal Injection (ASI) plant modification, which is a risk reduction modification that was installed for the transition to NFPA 805. Excluding ASI from the compliant plant model results in negative delta CDF and delta LERF for most fire areas and for the plant overall. The ASI modification may be omitted from the model of the compliant plant because ASI is an NFPA 805 modification that is not required for deterministic compliance. There is precedent for omitting risk reduction modifications from the compliant plant, resulting in an overall reduction in risk resulting from NFPA 805 transition.

The motivation for not taking credit for the ASI modification in the compliant plant was to remove conservatism in the delta risk calculation and accurately reflect delta CDF compliance with the RG 1.174 guidelines for transition to NFPA 805. If the compliant plant were defined to include all NFPA 805 modifications, the plant delta CDF and delta LERF would be 1.04E-5 per reactor year and 3.22E-7 per reactor year, respectively. There are other NFPA 805 modifications that could also be omitted from the compliant plant model. Although removing the additional NFPA 805 modifications from the compliant plant model would drive the already-negative delta risks further negative, it would not change the conclusion that the delta risks meet the criteria in RG 1.174.

An additional update was made to the variant and compliant plant models as part of the response to this RAI. Conservative mapping of some cables to basic events that result in loss of offsite power was removed. This change resulted in reductions of CDF and LERF that more than offset the risk increases due to removal of credit for the abeyance seal. In addition, the seismic hazard risks, as provided in Table PRA RAI 03-1, have been updated following submittal of LAR-16-01490.

RG 1.205 guidance includes an evaluation of the "additional risk of recovery actions." Recovery actions are defined in NPFA 805 (Section 1.6.52) as those "activities to achieve the nuclear safety performance criteria that take place outside of the main control room or outside of the primary control station(s) for the equipment being operated, including the replacement or modification of components." At VCSNS, recovery actions credited for the NFPA 805 transition are limited to: (a) alternate shutdown activities associated with MCR abandonment due to fire-induced equipment failures or habitability concerns due to fire conditions, and (b) locally tripping reactor coolant pumps (RCPs) to prevent LOCAs associated with RCP seal failure. The additional risk of recovery actions for each fire area in which recovery actions are credited is conservatively estimated as the difference in total risk between the variant plant and an alternative compliant plant, which is defined to include credit for the ASI plant modification.

The alternative compliant plant is used to avoid obscuring the additional risk of recovery actions with the negative delta risk that would be introduced by omitting ASI from the compliant plant. The additional risk of recovery actions is a fraction of the total delta risk because the human error probability adjustments for the risk of recovery actions in the compliant plant are included in the total delta risk. Therefore, the delta risk calculated provides a bounding value for the risk of recovery actions. Table PRA RAI 03-1 provides bounding estimates of the additional risk of credited recovery actions by applicable fire area and for the entire plant. The conservatively estimated risks of recovery actions, presented in Table PRA RAI 03-1, meet the acceptance criteria defined by Region II of Figures 4 and 5 in RG 1.174.

Mechanism to Ensure NRC Acceptance Prior to Crediting RCP Abeyance Seal

VCSNS will not credit the RCP abeyance seal in its PRA models until the NRC accepts the RCP abeyance seal model. Within 60 days of NRC approval of LAR-16-01490, VCSNS will revise its PRA model control procedure to require NRC acceptance of an abeyance seal model prior to use.

SENSITIVE INFORMATION REMOVED

WITHHOLD FROM PUBLIC DISCLOSURE UNDER
10 CFR 2.390

SENSITIVE INFORMATION REMOVED

WITHHOLD FROM PUBLIC DISCLOSURE UNDER
10 CFR 2.390

SENSITIVE INFORMATION REMOVED

WITHHOLD FROM PUBLIC DISCLOSURE UNDER
10 CFR 2.390

SENSITIVE INFORMATION REMOVED

WITHHOLD FROM PUBLIC DISCLOSURE UNDER
10 CFR 2.390

PRA RAI 04

Enclosure 1, Attachment 1, Section 2 of the LAR states the fire PRA (FPRA) was updated for refinements to the internal events PRA (IEPRA). The LAR describes some of these refinements and indicates that a full-scope peer review was performed in June 2016. The LAR is not clear as to whether any refinements were made after the 2016 peer review that could be considered PRA upgrades. Also, the dispositions to F&Os from the 2016 peer review of the IEPRA in LAR Enclosure 1, Attachment 8 indicate additional changes may have been made to the IEPRA since the peer review. Therefore, it is uncertain whether the latest IEPRA incorporated in the FPRA meets the PRA acceptability guidance in RG 1.200, Revision 2. The staff requests the licensee provide the following additional information:

a) *Describe the changes made to the IEPRA since the full-scope peer review conducted in June 2016. This description should be of sufficient detail to determine whether the changes are considered PRA maintenance or PRA upgrades as defined in ASME/ANS RA-Sa-2009, Section 1-5.4, as qualified by RG 1.200, Revision 2. Include in your discussion: (1) any new methodologies (i.e., summarize the original method in the PRA and the new method); (2) changes in scope that impact the significant accident sequences or the significant accident progression sequences; (3) changes in capability that impact the significant accident sequences or the significant accident progression sequences.*

b) *For each change described in Part (a) above, indicate whether the change was PRA maintenance or a PRA upgrade, along with justification for this determination.*

c) *For each PRA upgrade identified in Part (b) above, either:*

    i. *Provide the findings of the peer review(s) performed on the upgrade and the disposition of the findings as it pertains to the impact on this LAR. OR,*

    ii. *Provide sufficient information for NRC staff to compare the technical adequacy of the analysis to RG 1.200, Revision 2, or provide a bounding or sensitivity evaluation of its effect until a focused-scope peer review can be completed.*

d) *Refinements were made to the IEPRA and internal flooding PRA (IFPRA). It is not clear why the risk values (i.e., CDF, LERF) for these hazards in LAR Enclosure 1, Attachment 5 (page 3) are the same as those for the NFPA 805 LAR. Provide updated IEPRA and IFPRA risk values in LAR Attachment 5 or provide clarification as to why these values remain unchanged from the NFPA 805 LAR.*

VCSNS Response:

RAI 04 a)

The table below shows the changes to the internal events PRA model since the 2016 full scope peer review. These are divided into 8a, 8b and 8c model revisions. "Configuration model" below refers to the risk monitor used for maintenance rule a4 (of 10 CFR 50.65).

The changes were reviewed against the definitions in ASME/ANS RA-Sa-2009 and determined to be PRA maintenance, not PRA upgrades. Justification for these determinations are summarized in the table below.

Table RAI 04-1			
8a	**Change**	**Comment**	**Reason this is not an upgrade**
Item 1	EOOS changes	Includes splitting offsite power.	Configuration model only.
Item 2	Data Update	Updated data from our last major revision that was left out of the database.	Data update process was peer reviewed in 2016 and has not changed.
Item 3	Change the Recovery Rule File	To support Fire PRA Quantification.	This eliminated double counting of some cutsets. It is not a method change.
Item 4	Fire Model Changes	Fault tree changes for fire model.	Fire modeling changes to support Fire PRA model (not internal events). No new methods used.
8b	**Change**	**Comment**	**Reason this is not an upgrade**
Item 1	Incorporate Fire Model Changes	Containment isolation credit discussed below in item 3.	Crediting different actions for manual back-up to automatic containment isolation. No new methods used.
Item 2	ECR50695E Emergency Feedwater Flow Control Modification Changes	Active AOV function replaced with passive cavitating venturi.	Incorporated a plant modification to replace active AOV function with passive equipment to meet the success criteria (See additional discussion under part b) below.)
Item 3	Eliminate O--CNTMISOL-HE	HRA event improperly recovers equipment needed to implement it.	Event to use phase A containment isolation switch. Manual isolation is now credited as performed for individual valves per procedure.

Item 4	Incorporate 8a_2 fault tree changes	Fire Emergency Procedure (FEP) status gates were updated.	This changed status gates for the configuration monitoring but did not affect core damage frequency (CDF) or large early release frequency (LERF).
Item 5	EOOS Changes		Changes to the configuration model - does not affect the quantified part of the model.
8c	**Change**	**Comment**	**Reason this is not an upgrade**
Item 1	CAFTA change to Correct Pressurizer Pressure Transmitter		Included some previously unmodeled transmitters and power supplies, but no new methods were used.
Item 2	EOOS Change to Re-Instate operator compensatory measures		Change to the configuration model - does not affect the quantified part of the model.
Item 3	Inclusion of XIT5905-EV in EOOS		Change to the configuration model - does not affect the quantified part of the model.

RAI 04 b)

All of the changes listed above under item a) are considered maintenance updates.

Item 2 under change 8b warrants additional discussion. This change (ECR 50695E) was a physical modification to the plant. The success criteria are for the EFW pumps to provide adequate flow to the steam generators for decay heat removal. The original plant design used a set of air operator valves (AOVs) on the discharge lines of the emergency feedwater (EFW) pumps. The function of the AOVs was two-fold. First, the valves were throttled to preclude EFW pump runout and balance the EFW pump flow between the three steam generators. The second function was to isolate flow to a faulted (secondary side break) steam generator. The faulted steam generator has a much lower pressure and the EFW pumps preferentially flow to the faulted steam generator. If high flow were sensed to a faulted steam generator, the associated AOV would close, directing flow to the intact steam generators. This active function was required to meet the success criteria to provide adequate flow to the intact steam generators.

The plant modification installed passive cavitating venturis in each EFW line to the steam generators. The venturis prevent EFW pump runout and balance flow between the three steam generators. The venturis also act to limit flow to a faulted (depressurized) steam generator. Closure of the associated AOV is no longer required to provide adequate flow to the intact steam generators. The success criteria to provide adequate flow to intact steam generators is met by passive means instead of active means following the plant modification. An associated change to this plant modification is that for a feedline break transient, 2 of 3 EFW pumps are now required. This is reflected in the PRA model. The net impact of the plant change is a 1% increase in CDF and a 3% increase in LERF.

ASME/ANS RA-Sa-2009 defines PRA Upgrade as "the incorporation into a PRA model a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences."

- The changes made to the model did not involve a new methodology. The same fault tree software, CAFTA, was employed.
- The success criteria did not change. The success criteria is to provide adequate flow to the intact steam generators for decay heat removal. Passive means were installed to meet the flow requirements, replacing the active function of the AOVs. The scope of the model change did not change accident sequence or progression.
- The capability of the PRA model has not changed. The risk insights, CDF and LERF computations all follow the same method.

In summary, the PRA model changes summarized above were made to reflect the as-built, as-operated plant. As defined in ASME/ANS RA-Sa-2009, PRA maintenance is "the update of the PRA models to reflect plant changes such as modifications, procedure changes or plant performance (data)." The PRA model changes since the June 2016 Peer Review are all classified as PRA maintenance.

RAI 04 c)

No PRA upgrades were identified. As discussed in response to RAI 04 b) above, all PRA model changes since the June 2016 Peer Review have been PRA maintenance as defined in ASME/ANS RA-Sa-2009.

RAI 04 d)

Letter from NRC to VCSNS (ML14287A289) dated February 11, 2015 lists a CDF of 3.3E-06/yr and LERF of 1.0E-07/yr for internal events (Table 3.4.6-1 on page 106). This information was provided in VCSNS letter RC-14-0067, Table RAI 98-3, page 29 of 29. The value provided in RC-14-0067 was a typo or transcription error for CDF. The correct value for Internal Events CDF in the second quarter of 2014 was 4.3 E-06/yr. The current CDF, as reported in LAR-16-01490, is 3.3E-06/yr. The LERF value has varied slightly, but the value to two significant digits is unchanged.

PRA RAI 05

Enclosure 1, Attachment 8 of the LAR provides PRA peer review F&Os and dispositions for the IEPRA and IFPRA. Address the following questions related to the dispositions of the IEPRA F&Os for this LAR:

a) *In F&O 2-18 (associated with supporting requirement, SR, HR-F2 of ASME/ANS RA-Sa-2009), the time window for successful completion of human failure events (HFE) (i.e., time available to perform operator actions) is not based on accident sequence based timings. The F&O provides an example where the operator action to supply alternate AC power (i.e., operator action OA_AAC_SBO) is based on "an assumption" rather than an actual accident sequence timing value. The associated disposition addresses this example by stating alternate AC power was designed to be available in 60 minutes and this time was used as the time required to complete the operator action. The disposition does not seem to address the issue identified by the F&O, because the F&O pertains to the time available to perform operator actions (i.e., system time window), while the disposition addresses the time required to complete the operator action (i.e., the time it takes to perform operator action). Also, the disposition only addresses the two examples discussed in the F&O; however, these examples may not necessarily represent all instances of this issue as indicated by the peer review team's assessment. Lastly, no basis is provided for the following statement in the disposition and it is not clear what is meant by "may be small changes" in that statement:*

> *There may be small changes to the HFE values for these items for the Internal Events model but no impact on Fire PRA.*

Considering the observations above, explain why resolution of this F&O has no impact on the FPRA and how this resolution impacts the IEPRA.

b) *In F&O 4-01 (associated with SR SC-A5 of ASME/ANS RA-Sa-2009), the IEPRA utilizes mission times less than 24 hours. The associated disposition states, "[t]he Peer Review team did not fully agree with this approach [(i.e., the approach discussed in the disposition and used in the PRA)] and recommended changing all mission time[s] to 24 hours. If adjustments to mission times are found necessary, the closure of this F&O is not expected to have a significant impact on CDF or LERF." The licensee's resolution to this F&O is based on an approach that is disagreed upon by the peer review team, and adjustments to mission times can have a significant impact on basic event failure probabilities and the resulting risk. Therefore, it is not clear how resolution of this F&O is not expected to have a significant impact on risk. Considering the observations above, clearly explain why resolution of this F&O is not expected to have a significant impact on FPRA risk. For example, describe and provide the results of a sensitivity study (such results should include fire plant total CDF, LERF, ΔCDF, and ΔLERF in Table W-3 of LAR Enclosure 1, Attachment 5) and discuss how the RG 1.205 risk acceptance guidelines continue to be met.*

c) *In F&O 4-05 (associated with SR SC-B4 of ASME/ANS RA-Sa-2009), thermal-hydraulic analyses (e.g., MAAP5 runs) and control room simulation runs were used to support success criteria and PRA assumptions; however, use of simulation runs are questionable for the reasons explained in the F&O. The associated disposition does not address use of simulation runs in development of success criteria and PRA assumptions. Explain how simulation runs were used to support development of success criteria and PRA assumptions. Justify how these simulator runs constitutes a thermal-hydraulic analysis and provides an adequate basis to support its use in development of success criteria and PRA assumptions consistent with ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2.*

VCSNS Response:

RAI 05 a)

F&O 2-18 was written against supporting requirement (SR) HR-F2. Overall the SR was Met at Capability Category (CC)-III. The assessment of HR-F2 is stated as follows in the Peer Review Report:

> *F&O: 2-18 ASSESSMENT: CC-III Met*
> *(a) Accident sequence specific timings are used for some HFEs and can be found in the associated documentation (Attachment 2 of CN-RAM-14-33 and the HRA database). However, some examples were found where the Time window for success is not based on AS/SC based timings (e.g. OA_AAC_SBO, etc.) or does not use the limiting timing for which it is applied (e.g. BCPM--XPP39CHE, etc.).*
> *(b) AS specific procedures are used.*
> *(c) Availability of cues/indications for detection and evaluation errors.*
> *(d) details of the tasks related to individual components are documented.*

The peer review found "some examples" of the time window for success is not based on AS/SC. Given the SR is Met a CC-III the Peer Review team concerns were not extensive and overall the HEP were considered appropriate. The peer team only gave two examples of specific concerns. The two HEP values are discussed in greater detail as follows.

> *OA_AAC_SBO is the HEP for the Failure to Align ACC in 1 hour (SBO). The 1 hour Time window for Success (T_{SW}) was set as the design basis established in NUMARC-8700 when the ACC was being designed. The Peer Review position was that an AS/SC time basis should be used and recommended the time be based on RCP seal leakoff. Given the low leakage of the N-9000 seals, the 60-minute T_{SW} is conservatively low. Increasing T_{SW} would likely not have a significant impact on the HEP value since the 60-minute T_{SW} already exceeds time required (27 minutes) with margin. Overall, the Time window for*

Success (T_{SW}) is conservative and increase to an AS/SC time is expected to have little impact on the HEP.

BCPM-XPP39CHE is the HEP for the Operator Fails to rack in and start SW pump XPP-39C. VC Summer is provided with a swing Service Water (SW) pump (XPP-39C) that can be aligned to either SW train. The HEP Time window for Success (T_{SW}) is calculated based on CCW system heatup time limits. The Peer Review comment was that "... the EDG will overspeed trip shortly after starting..." for a loss of offsite power scenario when the EDG will start automatically. Looking at the scenario (loss of SW to the EDG), the Peer Review comment likely intended to state the EDG will <u>over heat</u> shortly after starting. The T_{SW} for this scenario is different than CCW system heatup scenario. The T_{SW} for how long an EDG can run without SW flow has not been established at this time.

To assess the impact, a sensitivity run was made for the Internal Events PRA model. The HEP for BCPM-XPP39CHE was set to 1 (no credit taken for operation action to align swing SW pump) for all loss of offsite power scenarios. The change in CDF is 1.6% and the change in LERF is 0.7%. The HEP is not a significant risk contributor and final resolution of the F&O will not significantly impact risk results.

VCSNS made the following statement in the disposition on F&O 2-18

There may be small changes to the HFE values for these items for the Internal Events model but no impact on Fire PRA.

The basis of this statement is that, in general, small changes in either the Time window for Success (T_{SW}) or the required time to complete operator actions have small impacts on calculated HEP, given that there is margin between the two values. Nothing further was intended than this general statement.

RAI 05 b)

A sensitivity study was conducted to investigate the impact of revised mission times on calculated FPRA risk. The baseline for the sensitivity study was the revised FPRA model that was prepared in response to PRA RAI 03. The sensitivity case was created by starting from the baseline case and increasing all mission times that were less than 24 hours to 24 hours. Total plant CDF increases from 4.72E-05 to 5.25E-5 (an 11% increase). Total plant LERF increases from 2.42E-6 to 2.72E-06 (a 12% increase).

For the ΔCDF and ΔLERF calculations, the sensitivity case described in the previous paragraph was used as the variant plant model. A corresponding compliant plant model was created from the compliant plant model of PRA RAI 03 by increasing all mission times that were less than 24 hours to 24 hours. Table PRA RAI 05b-1 provides CDF, LERF, ΔCDF, and ΔLERF in a format based on Table W-3 of LAR Enclosure 1, Attachment 5. The discussion below describes how the updated risk results align with the risk acceptance guidelines of RG 1.205.

Delta risk is calculated to compare the post transition plant to the "compliant" plant, as defined in RG 1.205. Delta risk is evaluated to ensure that the guidelines of RG 1.205 and RG 1.174 are met. Table PRA RAI 05b-1 provides delta risk for each plant fire area. From Table PRA RAI 05b-1, the change in fire risk for the plant is as follows:

- Delta CDF = -5.39E-05 per reactor year
- Delta LERF = -1.36E-06 per reactor year

The delta risks are negative due to the way the compliant plant has been modeled as discussed below. Because the delta risks are negative, they meet the guidelines of RG 1.174, as stated in the following excerpts from RG 1.174, Section 2.4:

> *If the application clearly shows a decrease in CDF, the change has satisfied the relevant principle of risk-informed regulation with respect to CDF.*
>
> *If the application clearly shows a decrease in LERF, the change has satisfied the relevant principle of risk-informed regulation with respect to LERF.*

With total CDF and LERF for fire, internal events and flooding, and seismic hazards satisfying CDF < 1E-04 and LERF < 1E-05, the PRA risk results are within the range of Region II on the horizontal axes of Figures 4 and 5 of RG 1.174. Meeting these risk criteria allows flexibility for future changes, as noted in the following excerpts from RG 1.174, Section 2.4:

> *When the calculated increase in CDF is in the range of 10^{-6} per reactor year to 10^{-5} per reactor year (i.e., the increase in CDF falls within Region II of Figure 4), applications are considered only if it can be reasonably shown that the total CDF is less than 10^{-4} per reactor year.*
>
> *When the calculated increase in LERF is in the range of 10^{-7} per reactor year to 10^{-6} per reactor year (i.e., the increase in LERF falls within Region II of Figure 5), applications are considered only if it can be reasonably shown that the total LERF is less than 10^{-5} per reactor year.*

The model of the compliant plant for the delta risk calculation does not include credit for the Alternate Seal Injection (ASI) plant modification, which is a risk reduction modification that was installed for the transition to NFPA 805. Excluding ASI from the compliant plant model results in negative delta CDF and delta LERF for most fire areas and for the plant overall. The ASI modification may be omitted from the model of the compliant plant, because ASI is an NFPA 805 modification that is not required for deterministic compliance. There is precedent for omitting risk reduction modifications from the compliant plant, resulting in an overall reduction in risk resulting from NFPA 805 transition.

The motivation for not taking credit for the ASI modification in the compliant plant was to remove conservatism in the delta risk calculation and accurately reflect delta CDF compliance with the RG 1.174 guidelines for transition to NFPA 805. There are other NFPA 805 modifications that could also be omitted from the compliant plant model. Although removing the additional NFPA 805 modifications from the compliant plant model would drive the already-negative delta risks further negative, it would not change the conclusion that the delta risks meet the criteria in RG 1.174.

As noted in the response to PRA RAI 03, an additional update was made to the variant and compliant plant models as part of the response to PRA RAI 03. Conservative mapping of some cables to basic events that result in loss of offsite power was removed. This change resulted in reductions of CDF and LERF that more than offset the risk increases due to removal of credit for the abeyance seal. In addition, the seismic hazard risks as provided in Table PRA RAI 05b-1 have been updated following submittal of LAR-16-01490.

RG 1.205 guidance includes an evaluation of the "additional risk of recovery actions." Recovery actions are defined in NPFA 805 (Section 1.6.52) as those "activities to achieve the nuclear safety performance criteria that take place outside of the main control room or outside of the primary control station(s) for the equipment being operated, including the replacement or modification of components." At VCSNS, recovery actions credited for the NFPA 805 transition are limited to: (a) alternate shutdown activities associated with MCR abandonment due to fire-induced equipment failures or habitability concerns due to fire conditions, and (b) locally tripping RCPs to prevent LOCAs associated with RCP seal failure. The additional risk of recovery actions for each fire area in which recovery actions are credited is conservatively estimated as the difference in total risk between the variant plant and an alternative compliant plant, which is the same as the initial compliant plant for PRA RAI 05, except that it includes credit for the ASI plant modification.

The alternative compliant plant is used to avoid obscuring the additional risk of recovery actions with the negative delta risk that would be introduced by omitting ASI from the compliant plant. The additional risk of recovery actions is a fraction of the total delta risk because the human error probability adjustments for the risk of recovery actions in the compliant plant are included in the total delta risk. Therefore, the delta risk calculated provides a bounding value for the risk of recovery actions. Table PRA RAI 05b-1 provides bounding estimates of the additional risk of credited recovery actions by applicable fire area and for the entire plant. The conservatively estimated risks of recovery actions presented in Table PRA RAI 05b-1 meet the acceptance criteria defined by Region II of Figures 4 and 5 of RG 1.174.

SENSITIVE INFORMATION REMOVED

WITHHOLD FROM PUBLIC DISCLOSURE UNDER
10 CFR 2.390

SENSITIVE INFORMATION REMOVED

WITHHOLD FROM PUBLIC DISCLOSURE UNDER
10 CFR 2.390

SENSITIVE INFORMATION REMOVED

WITHHOLD FROM PUBLIC DISCLOSURE UNDER
10 CFR 2.390

SENSITIVE INFORMATION REMOVED

WITHHOLD FROM PUBLIC DISCLOSURE UNDER
10 CFR 2.390

RAI 05 c)

F&O 4-05 states "Utility used a mixture of MAAP5 runs and control room simulation runs to support their success criteria, HRA timing and any assumptions they used in the PRA." As a part of the ongoing F&O closures efforts, a summary calculation has been developed which details the success criteria supporting documents developed over time for the VCSNS Internal Events PRA model. This was a recommended action from F&O 4-05. The success criteria are based on thermal-hydraulic models, generic modeling standard with plant difference evaluation, design basis analysis, and operator actions. The type of thermal hydraulic analysis (i.e. computer code and version) and reference to the documentation are included for each success criteria. An example of a design basis analysis is 1 of 2 low head safety injection pumps provide flow to 2 of 3 RCS cold legs for long-term low-pressure recirculation for Large Break LOCA. An example of the generic modeling is the anticipated transient without SCRAM (ATWS) success criteria.

In no case is a simulator run used for success criteria.

Table RAI-05c-1 provides a sample of the success criteria document for Small Break LOCA. This is a snap shot of the success criteria at the time of the Peer Review and does not constitute Current Licensing Basis information.

Table RAI-05c-1 Snap Shot of Small Break LOCA Success Criteria			
Top Name	**Success Criteria**	**Analysis Used to Define SC**	**Supporting Documentation**
RT	Reactor trip on low pressurizer pressure.	Fault Tree Analysis	CN-RAM-14-021 (Reference 23)
HPI	1 of 2 high-pressure trains delivering flow to 2 of 3 RCS cold legs.	Design Basis Analysis, MAAP5.01[3] Analyses	CN-COA-91-129 (Reference 7) CN-RAM-13-041 (Reference 3) (Section 5.2.8, page 52) (indicates one injection port sufficient) FSAR Section 6.3.2 (Reference 24)
	Basis for not crediting low-pressure cold leg recirculation when HPI fails.	MAAP5.01 Analysis	CN-RAM-13-041 (Reference 3) Section 5.1.5
EFW	1 of 3 EFW pumps providing flow to 1 of 3 SGs.	Design Basis Analysis	CN-COA-91-129 Pages 23-25 (Reference 7) CN-RAM-13-041 (Reference 3) (Section 5.2.8, page 52) FSAR (Reference 24)
FB	Operator action to initiate Feed and Bleed (OAB1).	HFE Analysis.	CN-RAM-14-033 (Reference 22)
	Opening of 1 of 3 pressurizer PORVs.	MAAP5.01	CN-RAM-13-041 (Reference 3) Section 5.1.7
	Successful HPI.	See top HPI	See top HPI
SGP	Operator action to depressurize the secondary side (OAD_1).	HFE Analysis	CN-RAM-14-033 (Reference 22)
	Secondary side pressure relief. 1 of 3 SG PORVs if MSIVs are closed, otherwise 1 of 3 ADVs or 1 of 2 condenser steam dump valves.	IPE Calculation	CN-COA-91-129 Page 46 (Reference 7)
RBC	1 of 2 RBCUs to provide reactor building cooling.	MAAP 5.01 (not required if LTC successful however included in Referenced Analyses)	CN-RAM-13-041 Section 5.2.8, and 5.1.11.1 (Reference 3) CN-RAM-14-032, Rev 0, page 67 (Reference 5, Section 7.2.9)[2]
LTC	1 of 2 high-pressure trains delivering flow to 2 of 3 RCS cold legs.	See top HPI[1]	See top HPI
	Operator action to align high- pressure cold leg recirculation (OAR4).	MAAP5.01	CN-RAM-13-041 (Reference 3) Section 5.2.7 This analysis is considered bounding.

Notes:
[1]All 3 injection lines would be available because even the largest small LOCA break size is smaller than an injection line. CN-COA-91-129 Page 49. (Reference 7)
[2] Reference 5, Section 7.2.9 notes "A plant specific MAAP analysis for the 2" medium LOCA was made to demonstrate that spray would not automatically actuate if at least 1 RBCU was running (see Appendix B). Therefore, since the mass and energy release rates are lower, spray should not automatically actuate for any small LOCA event as long as heat sink is maintained and at least 1 RBCU is running."
[3] MAAP 5.0.1 analysis for feed and bleed is considered bounding for SLOCA success criteria and operator action timing.

Table RAI-05c-1 References:

3. CN-RAM-13-041, Revision 0, "V.C. Summer Nuclear Station – Accident Sequence and Success Criteria Update," May 2014.

5. CN-RAM-14-032, Revision 0, "V.C. Summer Nuclear Station Unit 1 Internal Events PRA, Success Criteria Notebook," May 2016.

7. CN-COA-91-129, Revision 0, "V.C. Summer Nuclear Station South Carolina Electric and Gas Company, Individual Plan Examination, Success Criteria Notebook," April 1993.

22. CN-RAM-14-033, Revision 0, "V.C. Summer Nuclear Station Unit 1 Internal Events PRA, HRA Notebook," May 2016.

23. CN-RAM-14-021, Revision 0, "V.C. Summer Nuclear Station Unit 1 Internal Events PRA, Reactor Protection System Notebook," May 2016.

24. VCSNS U1 FSAR, May 2013.