VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

10 CFR 50.69
June 29, 2020 10 CFR 50.90

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

Serial No.: 20-201
NRA/GDM: R1
Docket Nos.: 50-280/281
License Nos.: DPR-32/37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
APPLICATION TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS"
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

By letter dated December 6, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19343A019), Virginia Electric and Power Company (Dominion Energy Virginia) submitted a license amendment request (LAR) for Surry Power Station (SPS) Units 1 and 2. The proposed license amendment would modify the SPS licensing basis by the addition of a license condition to allow for the implementation of the provisions of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the LAR and requested additional information (RAI) to complete its review. Dominion Energy Virginia's response to the NRC RAI is provided in Attachment 1.

Should you have any questions regarding this submittal or require additional information, please contact Mr. Gary D. Miller at (804) 273-2771.

Sincerely,

Mark D. Sartain
Vice-President – Nuclear Engineering and Fleet Support

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mark D. Sartain, who is Vice President – Nuclear Engineering and Fleet Support of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 29th day of June, 2020.

My Commission Expires: 12/31/20.

CRAIG D SLY
Notary Public
Commonwealth of Virginia
Reg. # 7518653
My Commission Expires December 31, 2020

Notary Public

Commitments made in this letter: See Attachment 4 for a list of the regulatory commitments associated with this response. These commitments will be completed prior to implementation of the Surry Power Station 10 CFR 50.69 categorization process.

Attachments:
1. Response to NRC Request for Additional Information Regarding License Amendment Request to Adopt 10 CFR 50.69

2. Revised Marked-up SPS Units 1 and 2 License Pages

3. Revised Proposed SPS Units 1 and 2 License Pages

4. List of Regulatory Commitments Included in this Correspondence

cc: U.S. Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Ave., NE Suite 1200
Atlanta, Georgia 30303-1257

Mr. Vaughn Thomas
NRC Project Manager - Surry
U.S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 04 F-12
11555 Rockville Pike
Rockville, MD 20852-2738

Mr. G. Edward Miller
NRC Senior Project Manager - North Anna
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 09 E-3
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Surry Power Station

State Health Commissioner
Virginia Department of Health
James Madison Building – 7th Floor
109 Governor Street
Room 730
Richmond, Virginia 23219

Attachment 1

<u>RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO ADOPT 10 CFR 50.69</u>

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Power Station Units 1 and 2

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO ADOPT 10 CFR 50.69

NRC Comment:

By letter dated December 6, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19343A019), Virginia Electric and Power Company (Dominion Energy Virginia, the licensee) submitted a license amendment request (LAR) for the Surry Power Station Units 1 and 2 (SPS). The proposed license amendment would modify the SPS licensing basis, by the addition of a license condition, to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." The U.S. Nuclear Regulatory Commission (NRC) staff from Probabilistic Risk Assessment Licensing Branches A and C (APLA, APLC) have reviewed the LAR and request additional information (RAI) in order to complete the review.

RAI 01 – Proposed License Condition (APLC)

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires, for a license amendment, a description of measures taken to assure the level of detail of the systematic processes that evaluate the plant. The guidance in Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (ADAMS Accession No. ML052910035), allows licensees to implement different approaches, depending on the scope of their probabilistic risk assessment (PRA) (e.g., the approach if a seismic margins analyses is relied upon is different and more limiting than the approach if a seismic PRA is used). Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," (ADAMS Accession No. ML061090627), states, "[a]s part of the NRC's review and approval of a licensee's or applicant's application requesting to implement 10 CFR 50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach."

Section 2.3 of the LAR proposed the following license condition:

> The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; qualitative assessments of seismic insights; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2)

passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 [American Society of Mechanical Engineers/American Nuclear Society (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 specified in License Amendment No. [XXX] dated [AMENDMENT DATE].

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach).

The proposed license condition does not appear to reflect the alternate seismic approach provided in the LAR, which is a modified version of the Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk- Informed Categorization," Tier 1 approach. In light of this observation, the licensee is requested to provide a revised license condition that refers to the alternate seismic approach provided in the application and any subsequent supplements.

Dominion Energy Virginia Response

The proposed License Condition "V" to be included in the Surry Units 1 and 2 Operating Licenses (OLs) has been revised to specifically address the use of a modified version of the EPRI 3002012988, *"Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk- Informed Categorization,"* Tier 1 approach to assess seismic risk. The revised marked-up and proposed typed OL pages are provided in Attachments 2 and 3, respectively.

RAI 02 – Overlap of Functions and Components (APLA)

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires, for a license amendment, a description of measures taken to assure the level of detail of the systematic processes that evaluate the plant. The guidance in NEI 00-04 allows licensees to implement different approaches, depending on the scope of their PRA.

Section 7.1 of NEI 00-04 states, "[d]ue to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC, or part thereof, should be assigned the highest risk significance for any function that the SSC or part thereof supports." Section 4 of NEI 00-04 states that a candidate low-safety-significant (LSS) SSC that supports an interfacing system should remain uncategorized until all interfacing systems

are categorized. The LAR does not discuss consideration or implementation of the guidance in Section 7.1 of NEI 00-04.

The licensee is requested to explain how the categorization process will be implemented to ensure that the cited guidance in NEI 00-04 will be followed and that any functions/SSCs that serve as an interface between two or more systems will not be categorized until the categorization for all of the systems that they support is completed and that SSCs that support multiple functions will be assigned the highest risk significance for any of the functions they support.

Dominion Energy Virginia Response

Per NEI 00-04 Section 4, for interfacing systems, all impacts an SSC has on interfacing systems will be considered. In the event an SSC in a system being categorized supports multiple systems, its risk significance will be reflected in the System Categorization Document (SCD) for the system defined by its boundaries and its impact on all interfacing systems. Per Section 7.1 of NEI 00-04, an SSC being considered for categorization will be assigned the highest risk significance of any portion of the process, whether from a function or component level evaluation.

Dominion Energy Virginia will perform the categorization of any functions/SSCs that serve as an interface between two or more systems in accordance with its categorization procedures. One of the initial steps in Dominion Energy's system categorization procedure is to develop a list of system functions. If the system includes components that support functions of other systems, they are identified as interfacing system components.

If an SSC supports two functions, one being LSS and the other being HSS, the SSC will receive a categorization of HSS. If an SSC supports only an LSS function but is considered HSS from a component level evaluation, the SSC will be considered HSS. The only exception applies to those components that support an HSS function but do not have a credible means to fail the HSS function. These components may be considered LSS in accordance with the guidance in Section 10.2 of NEI 00-04.

Typically, interfacing system components will remain uncategorized until all the interfacing systems are categorized. In most cases, interfacing system components that support uncategorized interfacing systems (and are LSS for the system being categorized) will be uncategorized and will thus retain any special treatments currently being applied.

In a case where it is desired to categorize the interfacing system component, that component's impact on all systems it interfaces with will be considered in the SCD for the system to which the SSC belongs. The evaluation will assess the interfacing SSC's contribution to functions in the interfacing systems and whether the interfacing system function(s) is/are risk significant. The interfacing SSC will then be assigned a risk

significance consistent with the guidance in NEI 00-04 Section 4 that all system interfaces have been considered. It should be noted this impact evaluation would also be performed for the passive categorization portion of the process (i.e., impact of pressure boundary failure of the interfacing SSC on the interfacing system). This approach will not categorize SSCs in the interfacing system; only the interfacing SSC(s) from the subject system will be categorized. All SSCs and functions of the interfacing system will be categorized when the interfacing system is categorized.

RAI 03 – Crediting of FLEX in the PRA Model (APLA)

Paragraphs (c)(1)(i) and (ii) of 10 CFR 50.69 require that a licensee's PRA be of sufficient quality and level of detail to support the SSC categorization process, and that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience. 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 staff's position concerning incorporating mitigating strategies (FLEX) into a PRA in support of risk-informed decision making in accordance with the guidance in RG 1.200, Revision 2 (ADAMS Accession No. ML090410014).

To complete the staff's review of the FLEX strategies modeled in the PRA, the licensee is requested to provide the following information for the internal events and internal flooding PRAs, as appropriate:

a. *Conclusion 5 of NRC memorandum dated May 30, 2017 states, "[t]he NRC staff does not agree with crediting spare portable equipment not modeled in the PRA in lieu of using appropriate failure rates, because this approach is not consistent with the ASME/ANS PRA Standard [ASME/ANS RA-Sa-2009] and RG 1.200. Furthermore, the potential impact of underestimating failure rates could be larger than the unquantified risk benefits of spare equipment not modeled in PRAs." Conclusion 6 of the memorandum states, "[t]he failure rates of permanently installed equipment cannot be used for portable equipment even if sensitivity analyses are performed. Licensees should use plant-specific o[r] generic data collected and analyzed using acceptable approaches to estimate the failure rates for portable equipment."*

   Attachment 6 of the LAR describes the FLEX strategies modeled in the PRA, including use of a portable diesel-driven generator (PDG) to restore power to specific equipment. The failure rates for the PDG were developed using failure rates for an emergency diesel-driven generator (EDG) from NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S.

Commercial Nuclear Power Plants (2015 update)." The licensee's basis for this assumption includes: (1) there are multiple spare portable generators on site that can be used if a single generator failed, and (2) the PDG failure rates will be considered as a source of uncertainty and a sensitivity study will be performed using the 5th and 95th percentile values. The basis for using EDG failure rates to represent that of the PDG is not consistent with Conclusions 5 and 6 of NRC memorandum dated May 30, 2017. To address the above observations, the licensee is requested to provide the following additional information:

i. *A detailed justification for using the EDG failure rates/probabilities to characterize the parameter estimates of the PDG and its uncertainties, and discuss how the PDG failure rates/probabilities are consistent with ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, and the NRC staff positions in memorandum dated May 30, 2017. Include in this discussion any plant-specific operational experience (e.g., number of failures, number of demands, operational hours) of the PDG, and discuss any screening or disregarding of plant-specific data (e.g., design modifications, changes in operating practices). Discuss how the failure rates/probabilities assumed in the PRA for the PDG is consistent with the relevant plant-specific evidence/operational experience.*

OR

ii. *Alternatively to part (i), propose a mechanism to ensure that prior to implementation of the 10 CFR 50.69 risk categorization process the appropriate failure rates/probabilities for PDG (including common cause failures, as applicable) that meet the supporting requirements (SRs) in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, are incorporated into the PRA. Also, describe how these failure rates will be developed/estimated consistent with the applicable SRs under HLR-DA-C and HLR-DA-D in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200.*

b. *Section 7.5 of NEI 16-06 states that the maintenance procedures for the portable equipment should be reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event. High-level requirement (HLR) HR-D in ASME/ANS RA-Sa-2009 describes the requirement assessing probabilities of the pre- initiator human failure events (HFEs) using a systematic process that addresses the plant-specific and activity-specific influences on human performance. Conclusion 13 of the NRC memorandum dated May 30, 2017 states, "[u]ntil acceptable guidance is provided for identifying and assessing unique aspects of pre-initiator HFEs for mitigating strategies, the staff may request additional information regarding assessment of those human failure events."*

Attachment 6 of the LAR discusses the PRA modeling of FLEX operator

actions as detailed in the emergency operating procedures. However, there is no discussion related to the process performed for the identification and assessment of the pre-initiator HFEs for mitigating strategies. To address the above observation, the licensee is requested to provide the following additional information:

i. A detailed discussion on how pre-initiator HFEs for mitigating strategies were identified and assessed to meet HLR-HR-D of ASME/ANS RA-Sa-2009, as endorsed by RG 1.200. This discussion should also address whether test and maintenance procedures for the portable equipment were reviewed for possible pre-initiator HFEs that renders the equipment unavailable during an event, and how the probabilities of those pre-initiator HFEs identified were assessed consistent with HLR-HR-D, as endorsed by RG 1.200.

ii. If pre-initiator HFEs for mitigating strategies were not assessed, then propose a mechanism to ensure that prior to implementation of the 10 CFR 50.69 risk categorization process the appropriate pre-initiator HFEs for mitigating strategies that meet the SRs in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, are incorporated into the PRA.

c. Condition (a) under SR HR-G3 of ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, requires evaluation of the impact of the quality of operator training/experience when estimating human error probabilities (HEPs). Attachment 6 of the LAR describes the use of portable FLEX equipment for mitigating strategies. However, it is not clear to the NRC staff as to how operator training/experience was reflected in the HEPs associated with deployment and installation of the portable FLEX equipment (e.g., deployment could utilize non-trained personnel that do not have the same training given to operators). To address the above observation, the licensee is requested to provide the following additional information:

i. Confirm whether or not non-trained operators (e.g., maintenance or security personnel) are used in the deployment and installation of the portable FLEX equipment.

ii. If non-trained operators are used in the deployment and installation of the portable FLEX equipment, then discuss how the impact of the quality of operator training/experience was evaluated in estimating the associated HEPs to meet SR HR-G3, as endorsed by RG 1.200.

Dominion Energy Virginia Response

a. ii. Prior to implementation, draft generic data from PWROG-18043-P R1, "FLEX Equipment Data Collection and Analysis," will be used to update the failure

probabilities for the credited FLEX equipment. Additionally, modeling will be performed to credit the availability of multiple redundant equipment while also considering common cause terms, as well as adequate timing to perform the task. Plant specific information will also be used as appropriate to inform the results. The requirements of HLR-DA-C and HLR-DA-D will be met. The draft generic data will be considered a source of uncertainty until the information is finalized. A sensitivity study will be performed per NEI 00-04 to increase the component common cause events to their 5th and 95th percentile values as part of the required 50.69 PRA categorization sensitivity cases. Additionally, a sensitivity study will be performed on the independent FLEX failures using the 5th and 95th percentile values.

b. ii. Prior to categorization, pre-initiator HFEs will be assessed to meet HLR-HR-D in ASME/ANS Ra-Sa-2009 or screened per HR-A1 and HR-B1.

c. i. Qualified security personnel are used to move portable FLEX equipment into place.

ii. The Systematic Approach to Training (SAT) was used to determine the appropriate level of training for the security personnel including the frequency for continued training. The required training included operation of the equipment used to move the portable FLEX equipment to the appropriate locations. The qualified personnel continue to receive training to ensure their skill level is maintained. The training performed, level of complexity, time required to perform the actions, etc., per SR HR-G3 were met. Additionally, a focused scope peer review was performed in October 2016 on the FLEX modeling with no current open F&Os.

RAI 04 - Dispositions of Key Sources of Uncertainty (APLA)

Paragraphs (c)(1)(i) and (ii) of 10 CFR 50.69 require that a licensee's PRA be of sufficient quality and level of detail to support the SSC categorization process, and that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience. The guidance in NEI 00-04 specifies sensitivity studies to be conducted for each PRA model to address uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask importance of components. NEI 00-04 guidance states that additional "applicable sensitivity studies" from characterization of PRA adequacy should be considered.

NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," (ADAMS Accession No. ML17062A466), provides guidance for how to treat uncertainties associated with PRA models in risk-informed decisionmaking. Additionally, Section 3.3.2 of RG 1.200, Revision 2, defines key assumptions and sources of uncertainty as follows:

A key assumption is one that is made in response to a key source of model uncertainty in the knowledge that a different reasonable alternative assumption would produce different results, or an assumption that results in an approximation made for modeling convenience in the knowledge that a more detailed model would produce different results. For the base PRA, the term "different results" refers to a change in the risk profile (e.g., total core damage frequency (CDF) and total large early release frequency (LERF), the set of initiating events and accident sequences that contribute most to CDF and to LERF) and the associated changes in insights derived from the changes in the risk profile. A "reasonable alternative" assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being challenged.

A key source of uncertainty is one that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an impact on the risk profile (e.g., total CDF and total LERF, the set of initiating events and accident sequences that contribute most to CDF and to LERF) such that it influences a decision being made using the PRA. Such an impact might occur, for example, by introducing a new functional accident sequence or a change to the overall CDF or LERF estimates significant enough to affect insights gained from the PRA.

As part of its audit (ADAMS Accession No. ML20058A010) of the licensee's LAR dated December 6, 2019, the NRC staff reviewed the PRA assumptions and sources of uncertainty associated with the internal events and internal flooding PRA models. The dispositions of some PRA assumptions/sources of uncertainty were unclear to the NRC staff as to why they were not considered "key" for this application and evaluated under LAR Attachment 6. To address the above observation, the licensee is requested to provide the following additional information:

a. In order to confirm that the PRA assumptions/sources of uncertainty were properly assessed for this application, provide a more detailed justification for characterizing the following PRA assumptions/sources of uncertainty as not impacting the 10 CFR 50.69 application. If this justification considers the impact on plant risk (i.e., CDF and LERF), address why this impact is an appropriate substitute for assessing SSC risk achievement worth (RAW) and Fussell-Vesely (F-V) impacts. If sensitivity studies were performed, describe these sensitivity studies and their results.

i. A number of PRA assumptions/sources of uncertainty were dispositioned based on not impacting plant risk and that any potential impact will be handled when the system is categorized (i.e., identifier I10).

The NRC staff notes that minimal changes in plant risk can impact the categorization for some SSCs since RAW and F-V values, not plant risk values, determine the high-safety-significant (HSS)/LSS designations. Also, it is unclear to the NRC staff how any "potential impact" will be addressed during the categorization process.

ii. Environmental qualification (EQ) equipment in containment: Non-EQ equipment could be credited in the PRA. This may be a non-conservative bias for loss of coolant accident (LOCA) sequences where the licensee recommended a sensitivity study to determine its impact on certain applications.

iii. Direct current (DC) bus load shed failure during a station blackout (SBO) event: SBO events can be a large contributor to risk. The conservative modeling decision of failure to load shed the DC bus during an SBO event could have a relatively large overall effect on risk, and therefore, the categorization results.

iv. The scenario cutoff time of two hours for the internal flooding analysis: The licensee's characterization of this assumption recommended a sensitivity study. However, the associated disposition did not indicate a sensitivity study was performed nor was a basis presented for concluding that this assumption has no significant impact on the 10 CFR 50.69 SSC categorization.

v. Loss of offsite power (LOOP) recovery curves for human reliability analysis: The NRC staff observed that a sensitivity study was performed and determined the impact was not significant with regards to plant risk. However, minimal changes in plant risk can impact the categorization for some SSCs since RAW and F-V values, not plant risk values, determine the HSS/LSS designations.

OR

b. Alternatively, propose a mechanism to ensure that these PRA assumptions/sources of uncertainty will be appropriately addressed during the implementation of the 10 CFR 50.69 risk categorization process. Also, describe how these will be addressed during the 10 CFR 50.69 risk categorization.

<u>Dominion Energy Virginia Response</u>

a. i. Procedure ER-AA-RIE-101, *Active Component Risk Significance Insights*, will be revised to communicate to the IDP any PRA level of detail modeling simplifications

(i.e., PRA assumption/sources of uncertainty disposition identifier I10) that impact plant risk for the system being categorized.

ii. Equipment survivability under adverse environmental conditions is documented in the Dominion Energy PRA Level 2 Supporting Analysis notebook. The review did not identify any non-EQ equipment required to mitigate core damage under adverse environmental conditions (e.g., LOCAs) in the containment.

iii. A recent SPS PRA model revision has included DC bus load shedding for SBO events. Therefore, this previous conservative modeling decision has been addressed.

iv. The Dominion Energy internal flooding models are built with the assumption that in scenarios with at least 2 hours available for operators to isolate a flood (time from receiving a flood alarm to time of equipment damage), isolation can be assumed successful because of high reliability of the operator action. Based on discussions and feedback on this assumption from industry peers and the author of the EPRI internal flooding reports, each flooding scenario screened based on this assumption is further analyzed to ensure no potentially risk significant cutsets are screened out of the model results.

v. The loss of offsite power (LOOP) recovery curves are based on a robust dataset from NUREG/CR-INEEL-04-023261, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1986 – 2003 (Draft)," and EPRI TR-1025749, "Losses of Offsite Power at US Nuclear Power Plants – 2011." Any variability in the LOOP recovery curves would not have a significant impact on plant risk. The major SSCs mitigating a LOOP event (e.g., Emergency Diesel Generators, Auxiliary Feedwater Turbine Driven Pump) already have RAW and/or F-V values greater than the risk significant threshold criteria. Other mitigating SSCs would be lower down in the cutsets and would not experience any significant change in RAW and/or F-V values.

RAI 05 – Interim PRA Updates (APLA)

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires, for a license amendment, a description of measures taken to assure the level of detail of the systematic processes that evaluate the plant. RG 1.201, Revision 1, provides guidance for categorizations of SSCs.

Section 4.2 of RG 1.200 states the LAR should include a discussion of the resolution of the peer review facts and observations (F&Os) that are applicable to the parts of the PRA required for the application. This discussion should take the following forms:

- *A discussion of how the PRA model has been changed, and*

- *A justification in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted (remained the same) by the particular issue.*

For F&O QU-F2-01 in Attachment 3 of the LAR, the peer review team identified where the licensee's PRA update process does not address all aspects of SRs QU-F2 and QU-F3 in ASME/ANS RA-Sa-2009 when reviewing interim model updates. In the disposition for F&O QU- F2-01, the licensee explains that reverifying SRs QU-F2 and QU-F3 after interim PRA updates would not impact the application. It is not clear to the NRC staff how review of the new results is sufficient to conclude that repeating all the sensitivity studies related to the QU requirements is not expected to impact SSC categorization. The NRC staff notes improper truncation values that do not achieve convergence could potentially impact the 10 CFR 50.69 categorization results.

To address this observation, the licensee is requested to provide the following additional information:

a. *A detailed justification that describes why not performing the truncation level sensitivity study for the PRA model used for 10 CFR 50.69 categorization does not impact the risk categorization of any SSC.*

b. *Alternatively, propose a mechanism to ensure that a truncation level sensitivity study is performed that confirms the truncation value(s) used to quantify the PRA does not impact SSC categorization prior to implementation.*

Dominion Energy Virginia Response

Dominion Energy has modified the model update procedure to require performing a truncation level sensitivity study that meets PRA Standard requirements with each model change.

RAI 06 – Categorization Process (APLA)

Paragraph (c)(1)(iv) of 10 CFR 50.69 requires that the SSC categorization process includes evaluations that provide reasonable assurance that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment permitted by implementation of 50.69(b)(1) and (d)(2) are small. The guidance in NEI 00-04 states, "[t]he purpose of the IDP [Integrated Decision-making Panel] is to ensure that the appropriate considerations from plant design and operating practices and experience are reflected in the categorization input." It is further discussed in Section 8.1 of NEI 00-04 that the cumulative sensitivity study should provide the IDP with both the overall assessment of the potential risk implications and the relative

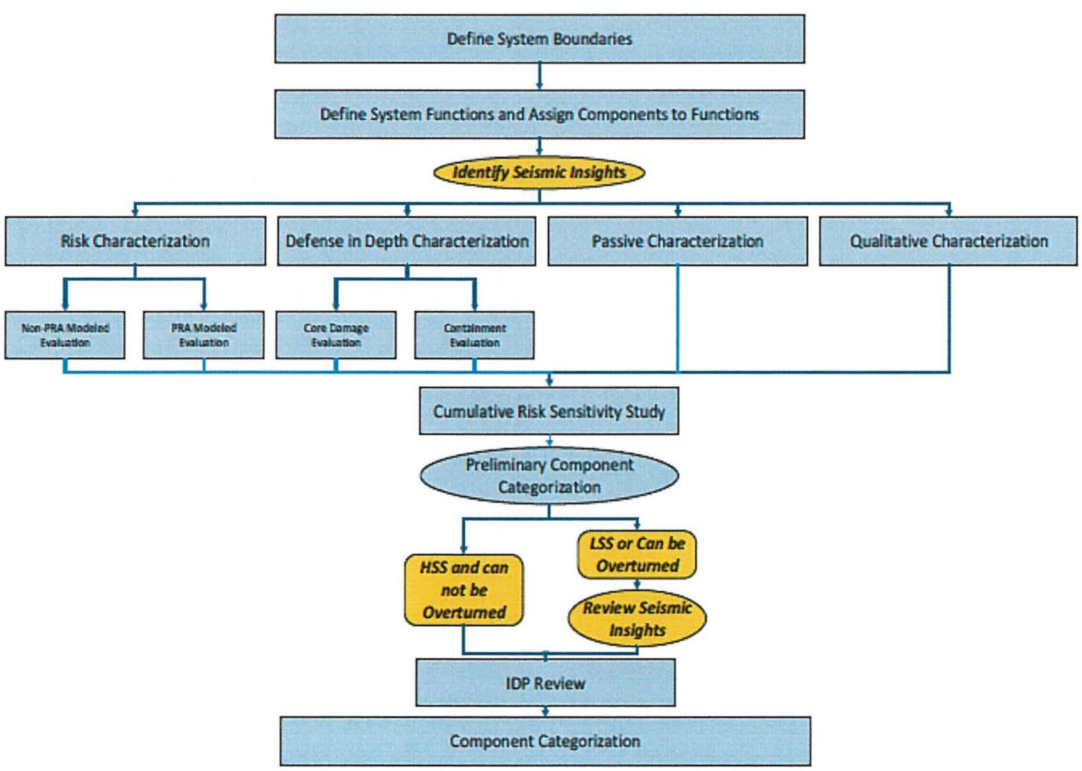
contribution of each system. Lastly, Figures 1-2 and 2-1 of NEI 00-04 indicate that the risk sensitivity study should be provided to the IDP prior to SSC categorization.

Section 3.1.1 of the LAR states that "[t]he order in which each of the elements of the categorization process (listed below) is completed is flexible, and, as long as they are all completed, they may even be performed in parallel." For certain elements, this does not appear to be the case (i.e., many of the elements must be completed prior to the IDP review). LAR Figure 3-1, "Categorization Process Overview," depicts the cumulative risk sensitivity study following the IDP review. However, based on the above discussion, it seems this sensitivity study should appear before the IDP review in LAR Figure 3-1. Thus, the licensee is requested to confirm whether or not the IDP review will consider information from the cumulative risk sensitivity study. If the IDP review will not consider this information, provide a detailed justification as to why the IDP review does not need to consider information from the cumulative risk sensitivity study, and how the requirements of 10 CFR 50.69 are met by the overall process depicted in LAR Figure 3-1.

Dominion Energy Virginia Response

The IDP will consider information from the cumulative risk sensitivity study. In the LAR for Surry Power Station, Figure 3-1 was incorrect. It should be replaced with the following Figure 3-1:

Figure 3-1: Categorization Process Overview



RAI 07 – SSCs Categorization Based on Other External Hazards (APLC)

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires that the quality and level of detail of the systematic processes that evaluate the plant for external events during operation is adequate for the categorization of SSCs. Section 3.2.4 of the LAR states that, "all other external hazards were screened for applicability to SPS per a plant-specific evaluation in accordance with GL 88- 20 and updated to use the criteria in ASME PRA Standard RA-Sa-2009. Attachment 4 provides a summary of the other external hazards screening results."

LAR Attachment 4, "External Hazards Screening," states that Dominion Energy is in the process of reevaluating the external flooding hazard and the tornado missile hazard, and that any identified discrepancies will be tracked in the corrective action program. In light of these observations, the licensee is requested to address the following:

a. *Provide a summary of the reevaluation of external flooding and tornado missile hazard including the basis for the reevaluation, potential differences between the current knowledge of those hazards at the site, and potential impact on SSC categorization under 10 CFR 50.69.*

b. *Discuss the licensee's approach, such as an implementation item completed prior to implementation of the 10 CFR 50.69 program and controlled by the proposed license condition, to ensure that any impacts on SSC categorization under 10 CFR 50.69 identified after the completion of the updated external flood reevaluation and the updated tornado missile reevaluation are included in the program consistent with the guidance in NEI 00-04.*

Dominion Energy Virginia Response

a. External Flooding Hazard

Dominion Energy Virginia's responses to the NRC March 2012 Near-Term Task Force (NTTF) 10 CFR 50.54(f) request for information are captured in a Flooding Focused Evaluation/Integrated Assessment submitted by letter dated October 1, 2019 (ADAMS Accession No. ML19291B034). The conclusion of the evaluations for re-evaluated flood hazards indicated that three reevaluated flood-causing mechanisms (local intense precipitation (LIP), storm surge, and intake canal failure) were not bounded by the current licensing basis and required further evaluation. A Focused Evaluation was performed on the LIP and intake canal failures in accordance with the guidance of NEI 16-05, Rev. 1. A Flooding Integrated Assessment was performed on the reevaluated storm surge.

Storm Surge

The re-evaluated storm surge was determined to potentially affect the low level intake structure Emergency Service Water pumps and create a temporary loss of service water for an event not bounded by the current licensing basis at a frequency less than 1E-4. A bounding analysis was performed which determined the CCDP for this event was approximately 5E-3 with the dominant sequence being a loss of cooling to the reactor coolant pump seals and a consequential seal LOCA. (Surry has replaced the reactor coolant pump seals with low leakage seals). Station core cooling would be maintained during the event by three auxiliary feedwater pumps (one steam driven pump, and two motor driven pumps) feeding steam generators. The two motor driven pumps would be powered by the site air cooled emergency diesel generators. The bounding analysis was determined to be less than 1E-6 and therefore screened out in accordance with ASME/ANS PRA Standard RA-Sa-2009 Part 6. Station procedures are being developed to implement FLEX strategies to mitigate the loss of service water; however, they are not required to support the bounding analysis.

LIP Event

Plant modifications were recommended to address the re-evaluated LIP event. However, the LIP event frequency was estimated to be less than 1E-6 and therefore screened out in accordance with ASME/ANS PRA Standard RA-Sa-2009 Part 6 for bounding analysis.

Intake Canal Failures

The detection and mitigation of intake canal failures were evaluated, and no plant modifications were recommended or required. Procedure changes were recommended to implement the site response strategy to ensure timely detection and mitigation response. The external hazard bounding analysis for the canal failure relies on the procedure change to be implemented.

The conclusions of the screening evaluations for external flooding hazard remain unchanged after considering the re-evaluated flooding evaluations once the procedure changes for the intake canal failure response are implemented.

Tornado Missile Hazard

Pursuant to Regulatory Issue Summary (RIS) 2015-06, Surry Power Station

performed multiple reviews to identify nonconformances between the physical plant configuration/design basis and the current licensing basis. Non-conformances were identified and entered into the correction action program for resolution.

The non-conformances were addressed through the Tornado Missile Risk Evaluator (TMRE) methodology (NEI 17-02), which demonstrated the risk is below acceptable limits.

b. Prior to implementation, the procedure changes for intake canal failure described above will be implemented. Additionally, per NEI 00-04, as part of the categorization assessment of other external hazard risk (e.g., external flooding, high winds, tornado missiles), an evaluation is performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS.

<u>RAI 08 – Risk Contribution of a Seismic Event (APLC)</u>

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires a description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific PRA, margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs. 10 CFR 50.69(b)(2)(iv) requires a description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv). The Statement of Consideration (SoC) on 50.69(b)(2)(iv) of the rule states that the licensee is required to include information about the evaluations they intend to conduct to provide reasonable confidence that the potential increase in risk would be small. The SoC further clarifies that a licensee must provide sufficient information to the NRC, describing the risk sensitivity study and other evaluations and the basis for their acceptability as appropriately representing the potential increase in risk from implementation of the requirements in the rule.

Section 3.2.3, "Seismic Hazards," of the enclosure to the LAR states that "the seismic risk (CDF/LERF) will be low such that seismic hazard risk is unlikely to influence an HSS decision." Section 2.2.2 of the EPRI report 3002012988 identifies the contribution of seismic to total plant risk as a basis for the use of the proposed alternate seismic approach. Further, the insights in the EPRI report are derived from the full spectrum of the seismic hazard (i.e., the entire hazard curve). The NRC staff noted that the LAR does not provide sufficient information to support the claim that the plant-specific seismic risk is a small percentage contribution to the total plant risk such that an integral importance measure for a

component would not result in an overall HSS determination, and thereby, the applicability of the proposed alternate seismic approach to the licensee.

The licensee is requested to provide a technical justification that supports the claim that the plant-specific seismic risk is low relative to the overall plant risk such that the categorization results will not be significantly impacted to support the applicability of the proposed alternate seismic approach.

Dominion Energy Virginia Response

The most direct method to justify that the plant-specific seismic risk is low relative to the overall plant risk would be the use of a modern seismic PRA (SPRA) to calculate the seismic core damage frequency (SCDF). However, Tier 1 plants such as Surry have not developed an SPRA with which to calculate the SCDF value.

An estimate of SCDF for Surry is calculated for use in this LAR. The calculation of the Surry SCDF estimate is based on a simple convolution of the Surry Ground Motion Response Spectra (GMRS) seismic hazard curve, in units of peak ground acceleration (PGA), and the Surry Individual Plant Examination of External Events (IPEEE)-based plant level seismic fragility, as reported by the NRC in Table C-2 of Reference 1. These seismic fragilities developed for the Generic Issue (GI) - 199 program were very approximate and have some inherent conservatisms, as well as some inconsistencies with licensee submitted IPEEE data. In addition, these estimates were intended to be used as indicators of changes in seismic risk based on consideration of new seismic hazard information. As such, a qualitative assessment to support the applicability of the proposed alternate 50.69 seismic approach is followed by an estimate of SCDF.

Qualitative Discussion of Surry Seismic Risk

For the Surry site, the GMRS-to-SSE comparison taken from the LAR (Reference 2) demonstrates that Surry qualifies as a Tier 1 plant under the criteria in the EPRI report. These criteria are similar to the grading process used in EPRI 1025287 (Reference 3) for the Fukushima 50.54(f) letter responses and in NEI 12-06 (Reference 4) for the Mitigation Strategy Assessment. This comparison also confirms that the expected seismic risk at Surry would be very low. Structures, systems, and components (SSCs) designed to the SSE level would have seismic capacity margins beyond the GMRS seismic level, which would result in high median capacities, and would therefore produce low seismic risk values.

Additionally, recent activities have been performed to ensure seismic risk at Surry remains low. During the post-Fukushima seismic walkdowns, Surry verified the plant configuration with the current seismic licensing basis; addressed degraded, nonconforming, or unanalyzed seismic conditions; and verified the adequacy of monitoring and maintenance programs for protective features (Reference 5). The justification that the plant-specific seismic risk for Surry is low, such that the 10 CFR 50.69

categorization results will not be significantly impacted, includes the following three key conclusions taken from the EPRI report:

1. Most SSCs that would be categorized as HSS for seismic risk would already be categorized as HSS for other reasons under 50.69 (e.g., internal events PRA insights, other external hazard risk insights, pressure boundary insights, Defense-in-Depth considerations, qualitative risk considerations).

2. Comparisons of the Surry SSE to GMRS show considerable seismic margin in the plant.

3. SPRA test cases conducted by EPRI demonstrate that even for plants at sites with high seismic ground motions compared to their design basis, very few if any SSCs would be designated HSS only due to seismic risks.

The text quoted in the question above from EPRI 3002012988 [or 3002017583] Section 2.2.2, while accurate, is simply reflecting a supplemental justification referring to the 50.69 Integral Assessment portion of the categorization process. The primary technical basis for the Tier 1 approach is described in Section 2.2.2 as the following.

> The test cases described in Section 3 showed that even for plants with high seismic ground motions compared to their design basis, there would be very few if any SSCs designated HSS for seismic unique reasons. At the low seismic hazard sites in Tier 1, the likelihood of identifying a unique seismic condition that would cause an SSC to be designated HSS is very low.

This conclusion holds true even for cases where the seismic risks are not considerably lower than internal events risks. For example, even if the internal events risks are low, the relative risk importance measures applied in 50.69 would still identify the appropriate HSS SSCs. This HSS vs LSS determination, based on internal events risk insights, cannot be changed by the IDP or the integral importance assessment process.

Consistent with Regulatory Guide 1.174 (Reference 6) and Regulatory Guide 1.201 (Reference 7), while for some risk-informed applications a quantitative assessment of all plant operating modes and hazard groups may be appropriate, for other risk-informed applications such as Surry 50.69 implementation, it is not necessary to have a PRA of such scope. A qualitative treatment of the missing modes and hazard groups is sufficient when those modes or hazards would not affect the decision. As such, a key premise of the EPRI approach is that most seismic-related SSCs that would be categorized as HSS in accordance with NEI 00-04 would also be categorized as HSS for other reasons (e.g., internal events PRA insights, other external hazard risk insights, pressure boundary insights, Defense-in-Depth considerations, qualitative risk considerations). Therefore, the EPRI report shows this qualitative treatment of seismic risk is sufficient to demonstrate the hazard does not affect the 50.69 HSS vs LSS categorization decision.

SCDF Estimation

Additional insights may be obtained by convolving the Surry GMRS hazard curve with the plant level high-confidence-of-low-probability-of-failure (HCLPF)[1]. A seismic PRA was developed for Surry in response to the IPEEE (Reference 8) where an SCDF of 8.2E-06/yr was calculated, and a plant level HCLPF of 0.16g was determined. This plant level HCLPF is also documented in Table C-2 in Reference 1. The dominant contributor was a seismic failure of the Turbine Building which was assumed to result in direct core damage since canal isolation was failed by failure of the 96-inch Circulating Water motor-operated valves (MOVs). The intake canal is the source of Service Water (SW), and loss of the intake canal results in loss of SW cooling to the Charging pumps and Component Cooling (CC) which results in loss of RCP seal cooling and RCP seal failure.

Convolving the GMRS with the plant level HCLPF results in a SCDF of approximately 1.9E-06/yr (includes 0.9 capacity factor). However, the seismic PRA developed for the IPEEE included conservatisms and did not account for some recent plant changes as listed below.

- Flowserve RCP seals – These low leakage RCP seals are designed to be more reliable and have a low failure probability if there is a loss of seal cooling. Therefore, the dominant seismic contributor of Turbine Building failure being a direct core damage sequence would be multiplied by this seal failure probability.

- FLEX - FLEX mitigating strategies were implemented in response to the Fukushima accident and are implemented in station blackout (SBO) or losses of the emergency switchgear room (ESGR) due to flooding. Credit for FLEX has been included in the Surry PRA which has reduced the CDF contributions for SBO and ESGR floods. The FLEX strategies are designed to reduce the CDF SBO sequences.

- Alternate AC Diesel Generator (AAC) – The AAC diesel generator was not credited. This diesel generator is of a different design and location from the emergency diesel generators (EDGs) and therefore would not be correlated with the EDGs. The seismic capacity of the diesel generator may be somewhat limited due to the battery racks; however, crediting the diesel generator would provide some reduction in the SCDF for loss of offsite power (LOOP and SBO sequences.)

- Both pressurizer power-operated relief valves (PORVs) required for bleed and feed – Recent thermal hydraulic analyses have determined that bleed and feed is successful with one of two PORVs instead of both PORVs as assumed in the

[1] The plant level HCLPF contains inherent conservatisms since the HCLPF is used to estimate a SCDF using a single SSC with a seismic capacity that reflects the most limiting seismic risk contributor; whereas, core damage scenarios typically require failure of multiple SSCs and Operator actions in diverse safety functions in order to result in core damage. And, as discussed further in this response, the plant level HCLPF does not reflect recent plant changes and model refinements; therefore, the estimated SCDF derived using the plant level HCLPF is considered overly conservative.

IPEEE seismic PRA. This would reduce the seismic sequences that contain bleed and feed and seismic damage to the PORVs or their power supplies (including the non-seismic failures of these SSCs).

The actual SCDF when using the GMRS and crediting the above changes is expected to result in a SCDF that is at least an order of magnitude less than the IPEEE SCDF (i.e., in the 8E-07/yr range). Based on this, the seismic risk contribution for Surry is considered to be low relative to the overall plant risk; therefore, the categorization results are not expected to be significantly impacted using the alternate seismic approach.

In summary:

- The Surry plant-specific seismic risk is low relative to the overall plant risk, and
- The Surry 50.69 categorization results will not be significantly or adversely impacted by proposed alternate seismic approach.

References:

1. USNRC memorandum, "Safety/Risk Assessment Results for Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," September 2, 2020 (ML100270582).

2. Surry GMRS – Surry response to Recommendation 2.1 in letter titled "Virginia Electric and Power Company Surry Power Station Units 1 and 2 Response to March 12, 2012 Information Request Seismic Hazard and Screening Report (CEUS Sites) for Recommendation 2.1," (ML14092A414).

3. EPRI 1025287, "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," Electric Power Research Institute (EPRI), February 2013.

4. NEI 12-06, Rev 4, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," December 2016.

5. Letter to NRC titled, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Report in Response to March 12, 2012 Information Request Regarding Seismic Aspects of Recommendation 2.3," dated November 27, 2012 (ML13017A002).

6. NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018 (ML17317A256).

7. NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006.

8. Surry IPEEE - letter titled, "Virginia Electric and Power Company, Surry Power Station Units 1 and 2, Summary Report for Individual Plant Examination of External Events (IPEEE) – Seismic," dated November 26, 1997.

RAI 09 – Configuration Control Process (APLC)

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires that the quality and level of detail of the systematic processes that evaluate the plant for external events during operation is adequate for the categorization of SSCs. Revision 3 of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (ADAMS Accession No. ML17317A256), establishes the need for an implementation and monitoring program to ensure that proposed changes do not degrade operational safety over time and that no adverse degradation occurs due to unanticipated degradation or common cause mechanisms. An implementation and monitoring program ensures that proposed changes continue to reflect the reliability and availability of impacted SSCs.

The staff's review of the LAR did not identify a description of the configuration control process to ensure that changes to the plant, including physical changes and changes to documents, are evaluated to determine the impact on design bases, licensing documents, programs, procedures, and training, or how the configuration control program is implemented.

The licensee is requested to provide a description of the configuration control process and a list for implementation of the configuration control program that includes:

a. A review of the impact on the System Categorization Document for configuration changes that may impact a categorized system under 10 CFR 50.69.

b. Steps to be performed if redundancy, diversity, or separation requirements are identified or affected. These steps include identifying any potential seismic interaction between added or modified components and new or existing safety-related or safe shutdown components or structures.

c. Review of impact to seismic loading, safe shutdown earthquake (SSE) seismic requirements, as well as the method of combining seismic components.

d. Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

Dominion Energy Virginia Response

a. A Periodic Review process assesses system/component performance changes and plant operation or design changes that have occurred for categorized systems on a frequency no longer than once every two refueling outages, as required by 10 CFR 50.69(e) to review the impact of plant changes on RISC1, RISC2, RISC3, and RISC4 SSCs. The review is implemented by station procedure.

b. The Dominion Energy design process includes the following in the design effects and consideration review:

   - Requirements for redundancy, diversity, and separation of structures, systems, and components are met including seismic interactions.
   - Review of impact to seismic loading, safe shutdown earthquake (SSE) seismic requirements, as well as the method of combining seismic components.
   - Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

RAI 10 – Alternate Seismic Approach (APLC)

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires that the quality and level of detail of the systematic processes that evaluate the plant for external events during operation is adequate for the categorization of SSCs. The proposed alternate seismic approach is based on insights of the EPRI report 3002012988, which derives risk insights from four case studies. Those case studies compare the HSS SSCs determined based on a seismic PRA (SPRA) against HSS SSCs determined from other PRAs used for categorization. Each of the cases studies included a full power internal events (FPIE) PRA, but only two of the four case studies used information from a fire PRA. Sections 3.3 through 3.5 of the EPRI report provide general information about the peer reviews conducted for the PRAs used for in each of the four case studies. However, the level of information is insufficient to determine whether the PRAs used in the case studies have been performed in a technically acceptable manner to support this application.

The NRC staff has previously requested and reviewed information to support its decision on the technical acceptability of the PRAs used in the case studies as well as details of the conduct of the case studies. This information is included in the supplements to the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, LAR for adoption of 10 CFR 50.69. The supplement to the 10 CFR 50.69 by Calvert Cliffs Nuclear Power Plant LAR, dated May 10, 2019 (ADAMS Accession No. ML19130A180), contained additional information related to the alternate seismic approach including incorporation by reference docketed information related to

case study Plants A, C, and D; the supplement dated July 1, 2019 (ADAMS Accession No. ML19183A012) further clarified the information related to the alternate seismic approach (see response to RAI 4); the supplement dated July 19, 2019 (ADAMS Accession No. ML19200A216) provided responses to support the technical acceptability of the PRAs used for the Plant A, C, and D case studies as well as technical adequacy of certain details of the conduct of the case studies; the supplement dated August 15, 2019 (ADAMS Accession No. ML19217A143) clarified a response in the July 19, 2019 supplement. The supplement dated July 19, 2019 included modifications to the content of the EPRI report.

Since the above-mentioned information was requested and reviewed by the staff for Calvert Cliffs Nuclear Power Plant's LAR for adoption of 10 CFR 50.69, the staff is unable to use it for the licensee's docket unless it is incorporated in the licensee's LAR. The above-mentioned information is necessary for the staff to make its regulatory finding on the licensee's proposed alternate seismic approach and has not been provided by the licensee. Thus, the licensee is requested to address the following:

a. *Provide the above-mentioned information to support the staff's regulatory finding on the alternate seismic approach by either incorporating the information by reference or responding to the RAIs in the identified supplements as well as providing information in the docketed documents related to case study Plants A, C, and D that were included by Calvert Cliffs Nuclear Power Plant in their supplement dated May 10, 2019 (ADAMS Accession No. ML19130A180).*

b. *If differences exist between the licensee's proposed alternate seismic approach and the information in the supplement to the 10 CFR 50.69 by Calvert Cliffs Nuclear Power Plant LAR dated May 10, 2019 (ADAMS Accession No. ML19130A180), identify such differences and either incorporate them in the licensee's proposed approach or justify their exclusion.*

Dominion Energy Virginia Response

a. The information related to EPRI Report 3002012988 case study Plants A, C, and D pertaining to the technical acceptability of the PRAs used, as well as the technical adequacy of certain details of the conduct of the case studies included in the supplement to the 10 CFR 50.69 Calvert Cliffs Nuclear Power Plant LAR, dated May 10, 2019 (ADAMS Accession No. ML19130A180); the supplement dated July 1, 2019 (ADAMS Accession No. ML19183A012); the supplement dated July 19, 2019 (ADAMS Accession No. ML19200A216); and the supplement dated August 15, 2019 (ADAMS Accession No. ML19217A143), are applicable to Surry Power Station and included in Surry's proposed seismic approach by reference. Additionally, EPRI Report 3002017583, which incorporated changes in response to the referenced supplements, is applicable to Surry.

b. The differences in the proposed alternate seismic approach between Surry and the supplement to the 10 CFR 50.69 Calvert Cliffs Nuclear Power Plant LAR, dated May 10, 2019 (ADAMS Accession No. ML19130A180), are as follows:

   1) The information referencing site specific Calvert Cliffs information from other Calvert Cliffs licensing responses that describe seismic capacity and other seismic evaluations do not apply to Surry. Surry-specific seismic capacity information is described, in part, in the response to RAI 8 above. Surry will provide the Integrated Decision-Making Panel (IDP) additional insights from previous seismic evaluations at Surry as described in the Surry LAR.

   2) The configuration control checklist described in the Calvert Cliffs response implies that a specific checklist was developed for 50.69 reviews. Surry configuration control is described in the response to RAI 9 above.

RAI 11 – Implementation Items (APLA/APLC)

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires that a licensee's application contain a description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown are adequate for the categorization of SSCs. If any of the responses to the RAIs associated with this LAR require follow-up actions prior to implementation of the 10 CFR 50.69 categorization process, the licensee is requested to provide a list of those actions and any PRA modeling changes including any items that will not be completed prior to issuing the amendment but must be completed prior to implementing the 10 CFR 50.69 categorization process.

Additionally, the licensee is requested to propose a mechanism that ensures these activities and changes will be completed and appropriately reviewed and any issues resolved prior to implementing the 10 CFR 50.69 categorization process (for example, a license condition that includes all applicable implementation items and a statement that they will be completed prior to implementation of the 10 CFR 50.69 categorization process).

Dominion Energy Virginia Response

Follow-up actions required to be completed prior to implementation stated in response to RAIs 2, 3, 4 and 7 are provided in Attachment 4. These actions are considered regulatory commitments, will be entered into the station's licensing commitment tracking program, and will be completed prior to implementation of the Surry Power Station 10 CFR 50.69 categorization process.

Attachment 2

REVISED MARKED-UP SPS UNITS 1 AND 2 LICENSE PAGES

**Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Power Station Units 1 and 2**

T. (Continued)

16. For the applicable UFSAR Chapter 14 events, Surry 1 will re-analyze the transient consistent with VEPCO's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A. If NRC review is deemed necessary pursuant to the requirements of 10 CFR 50.59, the accident analyses will be submitted to the NRC for review prior to operation at the uprate power level. These commitments apply to the following Surry 1 UFSAR Chapter 14 DNBR analyses that were analyzed at 2546 MWt consistent with the Statistical DNBR Evaluation Methodology in VEP-NE-2-A: • Section 14.2.7 - Excessive Heat Removal due to Feedwater System Malfunctions (Full Power Feedwater Temperature Reduction case only); • Section 14.2.8 - Excessive Load Increase Incident; • Section 14.2.9 - Loss of Reactor Coolant Flow; and • Section 14.2.10 - Loss of External Electrical Load	Prior to operating above 2546 MWt (98.4% RP).

U. Deleted by Amendment No. 289

Insert new License Condition V here

4. This renewed license is effective as of the date of issuance and shall expire at midnight on May 25, 2032.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Samuel J. Collins, Director

Office of Nuclear Reactor Regulation

Attachment: Appendix A, Technical Specifications

Date of Issuance: March 20, 2003

T. (Continued)

16. For the applicable UFSAR Chapter 14 events, Surry 2 will re-analyze the transient consistent with VEPCO's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A. If NRC review is deemed necessary pursuant to the requirements of 10 CFR 50.59, the accident analyses will be submitted to the NRC for review prior to operation at the uprate power level. These commitments apply to the following Surry 2 UFSAR Chapter 14 DNBR analyses that were analyzed at 2546 MWt consistent with the Statistical DNBR Evaluation Methodology in VEP-NE-2-A: • Section 14.2.7 - Excessive Heat Removal due to Feedwater System Malfunctions (Full Power Feedwater Temperature Reduction case only); • Section 14.2.8 - Excessive Load Increase Incident; • Section 14.2.9 - Loss of Reactor Coolant Flow; and • Section 14.2.10 - Loss of External Electrical Load	Prior to operating above 2546 MWt (98.4% RP).

U. Deleted by Amendment No. 289

Insert new License Condition V here

4. This renewed license is effective as of the date of issuance and shall expire at midnight on January 29, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Samuel J. Collins, Director

Office of Nuclear Reactor Regulation

Attachment: Appendix A, Technical Specifications

Date of Issuance: March 20, 2003

INSERT

V. The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. [XXX] dated [DATE].

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach).

Attachment 3

REVISED PROPOSED SPS UNITS 1 AND 2 LICENSE PAGES

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Power Station Units 1 and 2

T. (Continued)

16. For the applicable UFSAR Chapter 14 events, Surry 1 will re-analyze the transient consistent with VEPCO's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A. If NRC review is deemed necessary pursuant to the requirements of 10 CFR 50.59, the accident analyses will be submitted to the NRC for review prior to operation at the uprate power level. These commitments apply to the following Surry 1 UFSAR Chapter 14 DNBR analyses that were analyzed at 2546 MWt consistent with the Statistical DNBR Evaluation Methodology in VEP-NE-2-A: • Section 14.2.7 - Excessive Heat Removal due to Feedwater System Malfunctions (Full Power Feedwater Temperature Reduction case only); • Section 14.2.8 - Excessive Load Increase Incident; • Section 14.2.9 - Loss of Reactor Coolant Flow; and • Section 14.2.10 - Loss of External Electrical Load	Prior to operating above 2546 MWt (98.4% RP).

U. Deleted by Amendment No. 289

V. The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. [XXX] dated [Amendment Date].

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach.)

4. This renewed license is effective as of the date of issuance and shall expire at midnight on May 25, 2032.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Samuel J. Collins, Director

Office of Nuclear Reactor Regulation

Attachment: Appendix A, Technical Specifications

Date of Issuance: March 20, 2003

T. (Continued)

16. For the applicable UFSAR Chapter 14 events, Surry 2 will re-analyze the transient consistent with VEPCO's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A. If NRC review is deemed necessary pursuant to the requirements of 10 CFR 50.59, the accident analyses will be submitted to the NRC for review prior to operation at the uprate power level. These commitments apply to the following Surry 2 UFSAR Chapter 14 DNBR analyses that were analyzed at 2546 MWt consistent with the Statistical DNBR Evaluation Methodology in VEP-NE-2-A: • Section 14.2.7 - Excessive Heat Removal due to Feedwater System Malfunctions (Full Power Feedwater Temperature Reduction case only); • Section 14.2.8 - Excessive Load Increase Incident; • Section 14.2.9 - Loss of Reactor Coolant Flow; and • Section 14.2.10 - Loss of External Electrical Load	Prior to operating above 2546 MWt (98.4% RP).

U. Deleted by Amendment No. 289

V. The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. [XXX] dated [Amendment Date]

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach.)

4. This renewed license is effective as of the date of issuance and shall expire at midnight on January 29, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Samuel J. Collins, Director

Office of Nuclear Reactor Regulation

Attachment: Appendix A, Technical Specifications

Date of Issuance: March 20, 2003

Attachment 4

<u>LIST OF REGULATORY COMMITMENTS INCLUDED IN THIS CORRESPONDENCE</u>

**Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Power Station Units 1 and 2**

<table>
<tr><th colspan="2">List of Regulatory Commitments Included in this Correspondence</th></tr>
<tr><td>1. Revise Procedure ER-AA-RIE-103, Categorization Process, to describe the categorization of interfacing system components.</td><td rowspan="5">These regulatory commitments will be entered into the station's licensing commitment tracking program will be completed prior to implementation of the Surry Power Station 10 CFR 50.69 SSC risk categorization process.</td></tr>
<tr><td>2. Perform model changes to reflect multiple FLEX equipment available, associated common cause failures, and data update to include PWROG provided generic data.</td></tr>
<tr><td>3. Perform a sensitivity study per NEI 00-04 to increase the component common cause events to their 5th and 95th percentile values as part of the required 50.69 PRA categorization sensitivity cases.

Develop a procedure requirement to perform a sensitivity study on the independent FLEX failures using the 5th and 95th percentile values.</td></tr>
<tr><td>4. Revise Procedure ER-AA-RIE-101, Active Component Risk Significance Insights, to communicate to the IDP any PRA level of detail modeling simplifications (i.e., PRA assumption/sources of uncertainty disposition identifier I10) that impact plant risk for the system being categorized.</td></tr>
<tr><td>5. Develop and/or update applicable station procedures to provide appropriate guidance to station personnel on the actions required to respond to a beyond design basis event associated with response to a failure of the intake canal.</td></tr>
</table>