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January 17, 2019

10 CFR 50.69
10 CFR 50.90

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

Serial No.: 18-145
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Docket No.: 50-336
License No.: DPR-65

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
APPLICATION TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS"

In accordance with the provisions of 10 CFR 50.69 and 10 CFR 50.90, Dominion Energy Nuclear Connecticut, Inc. (DENC) requests an amendment to the license of Millstone Power Station Unit 2 (MPS2).

The proposed amendment would modify the MPS2 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 provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

Enclosure 1 to this letter provides the basis for the proposed change to the MPS2 Operating License. The categorization process being implemented through this change is consistent with NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, dated July 2005, which was endorsed by the NRC in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, dated May 2006. Attachment 1 of Enclosure 1 provides a list of categorization prerequisites. Use of the categorization process on a plant system will only occur after these prerequisites are met.

The NRC has recently reviewed the technical adequacy of the MPS2 Probabilistic Risk Assessment (PRA) model identified in this application during their review of a proposed revision to the integrated leak rate test Type A and Type C test intervals. The results of this review are contained in the NRC safety evaluation associated with License Amendment No. 335 dated September 29, 2018, "Millstone Power Station, Unit No. 2 –

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Issuance of Amendment No. 335 Regarding Revision to the Integrated Leak Rate Type A and Type C Test Intervals (EPID L-2017-LLA-0316)," [ADAMS Accession No. ML18246A007]. Since then, the MPS2 PRA has undergone another focused scope peer review to ensure: 1) modeling elements are consistent with ASME/ANS RA-Sa-2009, as endorsed by Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009, and 2) there are no PRA upgrades that have not been peer reviewed. Previously the NRC had reviewed the PRA technical adequacy for the adoption of TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b." The results of this review are contained in the NRC safety evaluation associated with License Amendment No. 324 dated October 29, 2015, "Millstone Power Station, Unit No. 2 - Issuance of Amendment Re: Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program, Adoption of TSTF-425, Revision 3 (TAC No. MF5096)" [ADAMS Accession No. ML15280A242].

DENC requests approval of the proposed license amendment by December 31, 2019, with the amendment being implemented within 60 days.

In accordance with 10 CFR 50.91(b), a copy of this application, with attachments, is being provided to the State of Connecticut.

This letter contains a proposed License Condition and the regulatory commitments described in Attachment 1 to Enclosure 1, as well as the additional regulatory commitment noted below.

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

VICKI L. HULL
NOTARY PUBLIC
REG. # 140542
COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES 5/31/2022

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 Dominion Energy Nuclear Connecticut, Inc. 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 17TH day of JANUARY, 2019.

My Commission Expires: 5-31-22.

Vicki L. Hull
Notary Public

Commitments made in this letter:

1. The categorization prerequisites specified in Attachment 1 to Enclosure 1.

2. Prior to implementation of the MPS2 10 CFR 50.69 categorization program, the MPS2 PRA internal events model of the steam generator tube rupture (SGTR) accident sequence will be revised to remove credit for achieving safe and stable conditions at 32 hours.

Enclosures:
1. Description and Assessment of the Proposed Change
2. Proposed Marked-up MPS2 License Page

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Enclosure 1

DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGE

**Dominion Energy Nuclear Connecticut, Inc.
(DENC)
Millstone Power Station Unit 2**

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LIST OF ATTACHMENTS

1 SUMMARY DESCRIPTION

The proposed amendment modifies the licensing basis of Millstone Power Station Unit 2 (MPS2) 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 provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

2 DETAILED DESCRIPTION

2.1 CURRENT REGULATORY REQUIREMENTS

The Nuclear Regulatory Commission (NRC) has established a set of regulatory requirements for commercial nuclear reactors to ensure that a reactor facility does not impose an undue risk to the health and safety of the public, thereby providing reasonable assurance of adequate protection to public health and safety. The current body of NRC regulations and their implementation are largely based on a "deterministic" approach.

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBEs). The deterministic approach then requires that the facility include safety systems capable of preventing or mitigating the consequences of those DBEs to protect public health and safety. The structures, systems and components (SSCs) necessary to defend against the DBEs are defined as "safety-related," and these SSCs are the subject of many regulatory requirements, herein referred to as "special treatments," designed to ensure that they are of high quality and high reliability, and have the capability to perform during postulated design basis conditions. Treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The distinction between "treatment" and "special treatment" is the degree of NRC specification as to what must be implemented for particular SSCs or for particular conditions. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: "safety-related," "important to safety," or "basic component." The terms "safety-related "and "basic component" are defined in the regulations, while "important to safety," used principally in the general design criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

2.2 REASON FOR PROPOSED CHANGE

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, Probabilistic Risk Assessments (PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner.

To take advantage of the safety enhancements available through the use of PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The rule contains requirements on how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and to place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, as described by NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" (Reference 1), which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability, and is a function of the SSC categorization results and associated bases. Finally, periodic assessment activities are conducted to make adjustments to the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, the rule enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as high safety significant, existing treatment requirements are maintained or enhanced. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows an alternative risk-informed approach to treatment that provides a

reasonable, though reduced, level of confidence that these SSCs will satisfy functional requirements.

Implementation of 10 CFR 50.69 will allow DENC to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 DESCRIPTION OF THE PROPOSED CHANGE

DENC proposes the addition of the following condition to the renewed operating license of MPS2 to document the NRC's approval of the use of 10 CFR 50.69:

(15) The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of 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; and the Appendix R program to evaluate fire risk; and 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 the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, 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 Unit 2 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 a seismic margins approach to a seismic probabilistic risk assessment approach).

3 TECHNICAL EVALUATION

10 CFR 50.69 specifies the information to be provided by a licensee requesting adoption of the regulation. This request conforms to the requirements of 10 CFR 50.69(b)(2), which states:

> A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:
>
> (i) A description of the process for categorization of RISC–1, RISC–2, RISC–3 and RISC–4 SSCs.
>
> (ii) 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 probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.
>
> (iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i).
>
> (iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

Each of these submittal requirements are addressed in the following sections.

The NRC has recently reviewed the technical adequacy of the MPS2 Probabilistic Risk Assessment (PRA) model identified in this application during their review of a proposed revision to the integrated leak rate test (Type A) and Type C test intervals. The results of this review are contained in the NRC safety evaluation associated with License Amendment No. 335 dated September 29, 2018, "Millstone Power Station, Unit No. 2 – Issuance of Amendment No. 335 Regarding Revision to the Integrated Leak Rate Type A and Type C Test Intervals (EPID L-2017-LLA-0316," (Reference 21). Since then, the MPS2 PRA has undergone another focused scope peer review to ensure all modeling elements are consistent with ASME/ANS RA-Sa-2009, as endorsed by RG.1200, Revision 2 and there are no PRA upgrades that have not been peer reviewed. Previously the NRC had reviewed the PRA technical adequacy for the adoption of TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - RITSTF Initiative 5b." The results of this review are contained in the NRC safety evaluation associated with License Amendment No. 324 dated October 29, 2015, "Millstone Power Station, Unit No. 2 - Issuance of Amendment Re: Risk-Informed

Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program, Adoption of TSTF-425, Revision 3 (TAC No. MF5096)" (Reference 20).

3.1 CATEGORIZATION PROCESS DESCRIPTION (10 CFR 50.69(b)(2)(i))

3.1.1 Overall Categorization Process

DENC will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Reference 2). NEI 00-04, Section 1.5, states, "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety-significant." A separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The process to categorize each system will be consistent with the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201. RG 1.201 states that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, the 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. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as LSS by all other elements. Similarly, NEI 00-04 only requires Item 4 to be completed for safety related active components/functions categorized as LSS by all other elements.

1. PRA-based evaluations (i.e., the internal events including internal flooding PRA).
2. Non-PRA approaches (i.e., fire safe shutdown equipment list (SSEL), seismic SSEL, other external events screening, and shutdown assessment).
3. Seven qualitative criteria in Section 9.2 of NEI 00-04.
4. The defense-in-depth assessment.
5. The passive categorization methodology

Categorization of SSCs will be completed per the NEI 00-04 process, as endorsed by RG 1.201, which includes the determination of safety significance through the various elements identified above. The results of these elements are used as inputs to arrive at a preliminary component categorization (i.e., High Safety Significant (HSS) or Low Safety Significant (LSS)) that is presented to the Integrated Decision-making Panel (IDP). Note: the term "preliminary HSS or LSS" is synonymous with the NEI 00-04 term

"candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 3-1 below. The safety significance determination of each element, identified above, is independent of each other and therefore the sequence of the elements does not impact the resulting preliminary categorization of each component or function. Consistent with NEI 00-04, the categorization of a component or function will only be "preliminary" until it has been confirmed by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final RISC category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04 Section 10.2. The IDP may always elect to change a preliminary LSS component or function to HSS; however, the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 and endorsed by RG 1.201. Table 3-1 summarizes these IDP limitations in NEI 00-04. The steps of the process are performed at either the function level, component level, or both. This is also summarized in the Table 3-1. A component is assigned its final RISC category upon approval by the IDP.

Table 3-1: Categorization Evaluation Summary

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Risk (PRA Modeled)	Internal Events Base Case – Section 5.1	Component	Not Allowed	Yes
	Fire, Seismic and Other External Events Base Case		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment – Section 5.6		Not Allowed	Yes
Risk (Non-modeled)	Fire, Seismic and Other External Hazards –	Component	Not Allowed	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable[1]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

Note:
[1] The assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2. In some cases, a 50.69 categorization team may provide preliminary assessments of the seven considerations for the IDP's consideration; however, the final assessments of the seven considerations are the direct responsibility of the IDP.

The seven considerations are addressed preliminarily by the 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

The mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., Internal Events PRA or Integral PRA assessment) or defense-in-depth evaluation will be initially treated as HSS. However, NEI 00-04, Section 10.2 allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS. In addition, Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with an HSS function but which do not support the critical attributes of that HSS function. Note that certain steps of the categorization process are performed at a component level (e.g., Passive, Non-PRA-modeled hazards - see Table 3-1). These components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Therefore, if an HSS component is mapped to an LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component may be driven HSS based on Table 3-1 above, or may remain LSS.

The following are clarifications to be applied to the NEI 00-04 categorization process:

- The IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and probabilistic risk assessment. At least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in the modeling and updating of the plant-specific PRA.

- The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address at a minimum the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the defense-in-depth philosophy and requirements to maintain this philosophy.

- The decision criteria for the IDP for categorizing SSCs as safety significant or low safety-significant pursuant to § 50.69(f)(1) will be documented in MPS2 procedures. Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. However, a simple majority of the panel is sufficient for final decisions regarding safety significant and LSS.

- Passive characterization will be performed using the processes described in Section 3.1.2. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

- An unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04. The factor of 3 was chosen as it is representative of the typical error factor of basic events used in the PRA model.

- NEI 00-04, Section 7, requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5, but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. This requirement is further clarified in the Vogtle SER (Reference 18), which states "...if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS."

- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The IDP must intervene to assign any of these HSS Function components to Low Safety Significant (LSS).

- With regard to the criteria that consider whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, MPS2 will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

The following are the exceptions taken to the NEI 00-04 categorization process:

- NEI 00-04, Section 5.2 states that the fire safety significance process takes one of two forms. Either the use of Fire Induced Vulnerability Evaluation (FIVE) or a Fire PRA. However, Section 3.2.2 of this LAR describes an alternate approach, which implements the Appendix R Safe Shutdown analysis that will be used in the MPS2 categorization process to evaluate safety significance related to the fire hazard.

The risk analysis to be implemented for each hazard is described below:

- <u>Internal Event Risks</u>: Internal events including internal flooding PRA model version MPS2-R05g, April 2018. This model was accepted by the NRC in Reference 21.

- <u>Fire Risks</u>: The MPS2 Appendix R SSEL will be used. The MPS2 Appendix R SSEL is a living program that reflects the current as-built, as-operated plant.

- <u>Seismic Risks</u>: Seismic SSEL from the Individual Plant Evaluation of External Events (IPEEE) seismic analysis as approved by NRC Safety Evaluation Report (SER) dated January 12, 2001 (Reference 16).

- <u>Other External Risks (e.g., tornados, external floods)</u>: The IPEEE screening process as approved by NRC SER dated January 12, 2001 (Reference 16) determined other external hazards were insignificant contributors to plant risk.

- <u>Low Power and Shutdown Risks</u>: Qualitative defense-in-depth (DID) shutdown model for shutdown configuration risk management (CRM) based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" (Reference 3), which provides guidance for assessing and enhancing safety during shutdown operations.

A change to the categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach) will not be used without prior NRC approval. The SSC categorization process documentation will include the following elements:

1. Program procedures used in the categorization

2. System functions, identified and categorized with the associated bases

3. Mapping of components to support function(s)
4. PRA model results, including sensitivity studies
5. Hazards analyses, as applicable
6. Passive categorization results and bases
7. Categorization results including all associated bases and RISC classifications
8. Component critical attributes for HSS SSCs
9. Results of periodic reviews and SSC performance evaluations
10. IDP meeting minutes and qualification/training records for the IDP members

3.1.2 Passive Categorization Process

For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components will be evaluated using the Arkansas Nuclear One (ANO) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology contained in Reference 5 consistent with the related Safety Evaluation (SE) issued by the Office of Nuclear Reactor Regulation.

The RI-RRA methodology is a risk-informed safety classification and treatment program for repair/replacement activities (RI-RRA methodology) for pressure retaining items and their associated supports. In this method, the component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed. It additionally applies deterministic considerations (e.g., DID, safety margins) in determining safety significance. Component supports are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model.

The use of this method was previously approved for a 10 CFR 50.69 application by the NRC in the Vogtle SER dated December 17, 2014 (Reference 18). The RI-RRA method as approved for use at Vogtle for 10 CFR 50.69 does not have any plant specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, this RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment. The passive categorization process is intended to apply the same risk-informed process accepted in the ANO2-R&R-004 (Reference 5)

for the passive categorization of Class 2, 3, and non-class components. This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662 as published in RG 1.147, Revision 18. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/ replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned HSS for passive categorization, which will result in HSS for its risk-informed safety classification and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at MPS2 for 10 CFR 50.69 SSC categorization.

3.2 TECHNICAL ADEQUACY EVALUATION (10 CFR 50.69(b)(2)(ii))

The following sections demonstrate the quality and level of detail of the processes used in categorization of SSCs are adequate. All technical elements of the PRA model described below has been peer reviewed consistent with ASME/ANS RA-Sa-2009, as endorsed by RG.1200, Revision 2 and there are no PRA upgrades that have not been peer reviewed. The PRA model is the same one described in the MPS2 license amendment request to revise the integrated leak rate test (Type A) and Type C test intervals (Reference 21). Previously the NRC had reviewed the PRA technical adequacy for adoption of TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control- RITSTF Initiative 5b" (Reference 20) with routine maintenance updates applied.

3.2.1 Internal Events and Internal Flooding

The MPS2 categorization process for the internal events and flooding hazard will use the plant-specific PRA model. The Dominion Energy risk management process ensures the PRA model used in this application reflects the as-built and as-operated plant for the MPS2 unit. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

3.2.2 Fire Hazards

The MPS2 categorization process will use the Fire SSEL for evaluation of safety significance related to fire hazards. The Fire Safe Shutdown paths identify the safety functions and associated sets of equipment credited to achieve and maintain safe shutdown under postulated fire conditions as defined by 10 CFR 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," and NRC Branch Technical Position CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power" (Reference 19). The Fire SSEL identifies the credited equipment on these Fire Safe Shutdown Paths. This approach also considers regulatory exemptions related to the Fire Safe Shutdown program and fire-induced Multiple Spurious Operations (MSOs) to identify any additional equipment.

The use of the Fire SSEL is a screening approach. There are no importance measures used in determining safety significance related to the fire hazard. Instead, using the Fire SSEL would identify all credited equipment as HSS regardless of their fire damage susceptibility or frequency of challenge. This approach ensures the SSCs that are credited to establish and maintain safe shutdown capability are retained as safety-significant. If a component is credited on the Fire SSEL, it is considered HSS. As stated in NEI 00-04, an SSC identified as HSS by a non-PRA method to address fire "may not be re-categorized by the IDP."

Furthermore, regulatory exemptions related to the Fire Safe Shutdown program and previously identified fire-induced MSOs were reviewed and it was concluded that no equipment in addition to the components on the Fire SSEL are relied upon to establish and maintain safe shutdown. Therefore, no additional components will be identified as HSS with regard to the fire hazard. The results of this review have been documented by the site and are available for NRC audit. Figure 3-1 illustrates how the Fire SSEL is reviewed to determine if the component being evaluated is HSS.

This approach is an alternate process from the NEI 00-04 endorsed approaches. Similar to the NEI 00-04 FIVE approach, this approach uses the SSEL as a screening tool. However, the development of the Fire SSEL is not based on a successive screening methodology and is the starting point for the FIVE methodology. Therefore, industry assessments have shown that this Fire SSEL approach leads to many additional SSCs being identified as HSS making it more conservative in determining safety significance than the NEI 00-04 FIVE approach or a Fire PRA.

The MPS2 Fire Safe Shutdown program is an active regulatory program that is routinely inspected by the NRC. It was confirmed that this program ensures that the Fire SSEL and the identification of additional equipment relied upon to establish and maintain safe shutdown reflects the current as-built, as-operated plant and that changes to the plant will be evaluated to determine their impact to the equipment list and the categorization process.

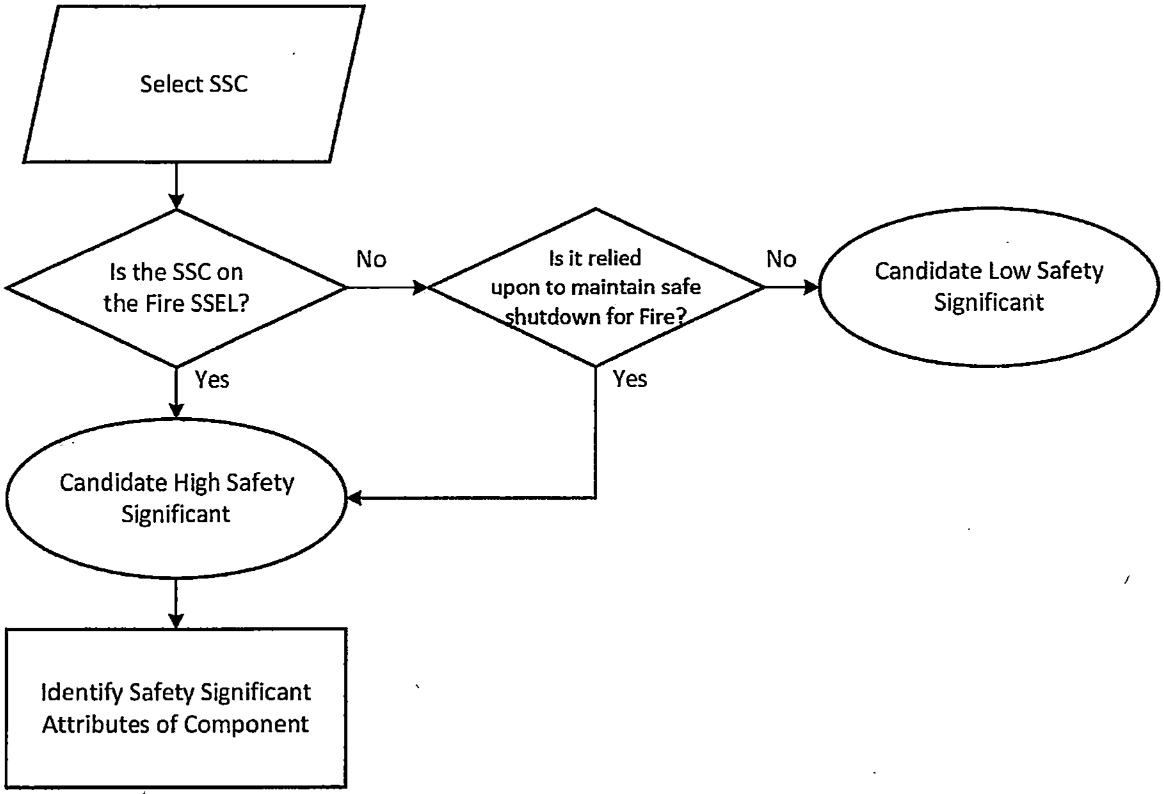

Figure 3-1: Safety Significance Process for Systems and Components Addressed in Fire Safe Shutdown Program

3.2.3 Seismic Hazards

The MPS2 categorization process will use the seismic margins analysis (SMA) performed for the IPEEE in response to GL 88-20 (Reference 6) for evaluation of safety significance related to seismic hazards (i.e., the SSEL). No plant specific approaches were utilized in development of the SSEL. NEI 00-04 approved use of the SSEL as a screening process results in the identification of all system functions and associated SSCs that are involved in the seismic margin success path as HSS. Since the analysis is being used as a screening tool, importance measures are not used to determine safety significance. The NEI 00-04 approach using the SSEL identifies credited equipment as HSS regardless of its capacity, frequency of challenge, or level of functional diversity.

An evaluation was performed of the as-built, as-operated plant against the SSEL. The evaluation compared the as-built, as-operated plant to the plant configuration originally assessed by the SMA. Differences were reviewed to identify any potential impacts to the equipment credited on the SSEL. Appropriate changes to the credited equipment were identified and documented. This documentation is available for audit. The MPS2

risk management program ensures that future changes to the plant will be evaluated to determine their impact on the SMA and risk categorization process.

3.2.4 Other External Hazards

All other external hazards (i.e., not seismic or fire hazards) were screened for applicability to MPS2 per a plant-specific evaluation in accordance with GL 88-20 (Reference 6) 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. Attachment 5 provides a summary of the progressive screening approach for external hazards.

As part of the categorization assessment of other external hazard risk, 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. All remaining hazards were screened from applicability and considered insignificant for every SSC and, therefore, will not be considered during the categorization process.

3.2.5 Low Power & Shutdown

Consistent with NEI 00-04, the MPS2 categorization process will use the shutdown safety management plan described in NUMARC 91-06 for evaluation of safety significance related to low power and shutdown conditions. The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04.

NUMARC 91-06 specifies that a defense-in-depth approach should be used with respect to each defined shutdown key safety function. The key safety functions defined in NUMARC 91-06 are evaluated for categorization of SSCs.

SSCs that meet the two criteria (i.e., considered part of a "primary shutdown safety system" or a failure would initiate an event during shutdown conditions) described in Section 5.5 of NEI 00-04 will be considered preliminary HSS.

3.2.6 PRA Maintenance and Updates

The MPS2 risk management process ensures the applicable PRA model used in this application continues to reflect the as-built and as-operated plant for the MPS2 unit. The process delineates the responsibilities and guidelines for updating the PRA models, and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, and industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process will assess the impact of these changes on the plant PRA model in a timely manner but no

longer than once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated.

In addition, DENC will implement a process that addresses the requirements in NEI 00-04, Section 11, "Program Documentation and Change Control." The process will review the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model upgrades will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.2.7 PRA Uncertainty Evaluations

Uncertainty evaluations associated with any applicable baseline PRA model(s) used in this application were evaluated during the assessment of PRA technical adequacy and confirmed through peer review process as discussed in Section 3.3 of this enclosure.

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in Section 8 of NEI 00-04 and in the prescribed sensitivity studies discussed in Section 5.

In the overall risk sensitivity studies, DENC will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference 18. Consistent with the NEI 00-04 guidance, Dominion Energy will perform both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in all identified PRA models for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study.

The detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 7.2 of NUREG-1855 and Section 3.1.1 of EPRI TR-1016737 (Reference 9). The process in these references was mostly developed to evaluate the uncertainties associated with the internal events PRA model; however, the approach can be applied to other types of hazard groups.

The list of assumptions and sources of uncertainty were reviewed to identify those which would be significant for the evaluation of this application. If the MPS2 PRA model used a non-conservative treatment, or methods that are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine its impact on

this application. Only those assumptions or sources of uncertainty that could significantly impact the risk calculations were considered key for this application.

Key MPS2 PRA model specific assumptions and sources of uncertainty for this application were identified and dispositioned in Attachment 6. The conclusion of this review is that no additional sensitivity analyses are required to address MPS2 PRA model specific assumptions or sources of uncertainty except for the following:

1) ECCS sump blockage probability is currently based on data from the mid-1990s, whereas more recent data is available from WCAP-16882, Rev. 1, "PRA Modeling of Debris-Induced Failure of Long Term Core Cooling via Recirculation Sumps." A sensitivity study will be performed using the newer sump blockage probabilities.

2) Thermally-induced SGTR is based on conservative NUREG-1570 analysis, whereas less conservative data is available from EPRI TR-107623-V1, Rev. 1, "Steam Generator Tube Integrity Risk Assessment." A sensitivity study will be performed using less conservative data from the aforementioned EPRI report.

3.3 PRA REVIEW PROCESS RESULTS (10 CFR 50.69(b)(2)(iii))

The PRA model described in Section 3.2 has been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference 7) consistent with NRC Regulatory Information Summary (RIS) 2007-06.

The internal events PRA model, including internal flooding, was subject to several focused scope peer reviews covering all supporting requirements that were conducted in accordance with RG 1.200, Revision 2 in September 2012, March 2018 and July 2018. No PRA upgrades as defined by the ASME PRA Standard RA-Sa-2009 (Reference 10) have occurred to the Internal Events PRA model since the July 2018 focused peer review.

A finding closure review was conducted on the identified PRA models the week of March 19, 2018. Resolved findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Close-out of Facts and Observations" (F&Os) (Reference 11) as accepted by NRC in the letter dated May 3, 2017 (ADAMS Accession No. ML17079A427) (Reference 12). The results of this review have been documented and are available for NRC audit.

Attachment 3 provides a summary of the remaining findings and open items, including:

- Open findings and disposition of the MPS2 peer reviews.
- Identification of and basis for any sensitivity analysis needed to address open findings.

The attachment identified above demonstrates the PRA is of sufficient quality and level of detail to support the categorization process and has been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC as required 10 CFR 50.69(c)(1)(i).

3.4 RISK EVALUATIONS (10 CFR 50.69(b)(2)(iv))

The MPS2 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions, and meets the requirements of §50.69(b)(2)(iv). Sensitivity studies described in NEI 00-04, Section 8 will be used to confirm that the categorization process results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF). The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, and human errors). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data and provide timely insights into the need to account for any important new degradation mechanisms.

3.5 FEEDBACK AND ADJUSTMENT PROCESS

If significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can (or actually did) prevent a safety significant function from being satisfied, an immediate evaluation and review will be performed prior to the normally scheduled periodic review. Otherwise, the assessment of potential equipment performance changes and new technical information will be performed during the normally scheduled periodic review cycle.

Scheduled periodic reviews once every two refueling outages will evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process will be updated. This review will include:

- A review of plant modifications since the last review that could impact the SSC categorization,
- A review of plant specific operating experience that could impact the SSC categorization,
- A review of the impact of the updated risk information on the categorization process results,
- A review of the importance measures used for screening in the categorization process, and

- An update of the risk sensitivity study performed for the categorization.

In addition to the normally scheduled periodic reviews, if a PRA model or other risk information is upgraded, a review of the SSC categorization will be performed.

4 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The following NRC requirements and guidance documents are applicable to the proposed change.

- The regulations in 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."
- NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, April 2015.
- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

The proposed change is consistent with the applicable regulations and regulatory guidance.

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

Dominion Energy Nuclear Connecticut Inc. (DENC) proposes to modify the licensing basis to allow for the voluntary 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 provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

DENC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of structures, systems and components (SSCs) subject to special treatment requirements and to implement alternative treatments per the regulations. The process used to evaluate SSCs for changes to special treatment requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not significantly affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC. Under the proposed change, no additional plant equipment will be installed.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not affect any Safety Limits or operating parameters used to

establish the safety margin. The safety margins included in analyses of accidents are not affected by the proposed change. The regulation requires that there be no significant effect on plant risk due to any change to the special treatment requirements for SSCs and that the SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, DENC concludes the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 CONCLUSIONS

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment could change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6 REFERENCES

1. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute, July 2005.

2. NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.

3. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.

4. EPRI TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure, Final Report," Revision B-A, January 2000.

5. ANO SE Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative ANO2-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems (TAC NO. MD5250) (Accession No. ML090930246), April 22, 2009.

6. Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4," USNRC, June 1991.

7. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

8. NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, Revision 1, March 2017.

9. EPRI TR-1016737, Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, December 2008.

10. ASME/ANS RA-Sa-2009, Standard for Level I/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, dated February 2009.

11. NEI Letter to NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," February 21, 2017, Accession Number ML17086A431.

12. NRC Letter to Mr. Greg Krueger (NEI), "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 7-12, and 12-13, Close Out of Facts and Observations (F&Os)," May 3, 2017, Accession Number ML17079A427.

13. NRC Letter to Mr. Oliver Martinez, "U.S. Nuclear Regulatory Commission (NRC) Comments on 'Addenda to a Current ABS: ASME RA-SB - 20XX, Standard for

Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications."

14. NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," Revision 0, Nuclear Energy Institute, August 2012.
15. NRC Letter to Mr. Biff Bradley (NEI), "U.S. Nuclear Regulatory Commission Comments on Nuclear Energy Institute 12-13, 'External Hazards PRA Peer Review Process Guidelines,' Dated August 2012," November 16, 2012, Accession Number ML12321A280.
16. NRC Letter to Mr. R. G. Lizotte, "Millstone Nuclear Power Station, Unit No. 2 – Individual Plant Examination of External Events (IPEEE) (TAC No. M83642)," January 12, 2001, Accession Number ML010120072.
17. Dominion Nuclear Connecticut, Inc., "Proposed License Amendment Request to Revise Integrated Leak Rate Test (Type A) and Type C Test Intervals," October 4, 2017, Accession Number ML 17284A179.
18. NRC letter to TVA, "Vogtle Electric Generating Plant, Units 1 and 2 -Issuance of Amendments RE: Use of 10 CFR 50.69 (TAC Nos. ME9472 AND ME9473)," dated December 17, 2014 (Accession No. ML14237A034).
19. Branch Technical Position CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," Revision 3, dated July 1981 Accession Number ML070660454.
20. NRC Letter to Mr. David A. Heacock, "Millstone Power Station, Unit No. 2 – Issuance of Amendment RE: Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program, Adoption of TSTF-425, Revision 3 (TAC No. MF5096)," October 29, 2015, Accession Number ML 15280A242.
21. NRC Letter to Mr. Daniel G. Stoddard, "Millstone Power Station, Unit No. 2 – Issuance of Amendment No. 335 Regarding Revision to the Integrated Leak Rate Type A and Type C Test Intervals (EPID L-2017-LLA-0316," September 25, 2018, ML18246A007.

Attachment 1: List of Categorization Prerequisites

DENC will establish procedure(s) prior to the use of the categorization process on a plant system. The procedure(s) will contain the elements/steps listed below.

- Integrated Decision-Making Panel (IDP) member qualification requirements

- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significant (HSS) or Low Safety Significant (LSS) based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.2 of this enclosure). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting an LSS function are categorized as preliminary LSS.

- Component safety significance assessment. Safety significance of active components is assessed through a combination of Probabilistic Risk Assessment (PRA) and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.

- Assessment of defense-in-depth (DID) and safety margin. Safety-Related components that are categorized as preliminary LSS are evaluated for their role in providing DID and safety margin and, if appropriate, upgraded to HSS.

- Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.

- Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174.

- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.

- Documentation requirements per Section 3.1.1 of the enclosure.

Attachment 2: Description of PRA Models Used in Categorization

Units	Model	Baseline CDF	Baseline LERF	Comments
1	MPS2 Integrated Model for Internal Events and Internal Flooding Hazards – CAFTA MPS2-R05g, April 2018	1.95E-05/yr	1.27E-06/yr	M209A Model, July 2012 Focused Scope Peer Review Against R.G. Rev. 2 (SFCP Reference 20) MPS2-R05e (ILRT LAR Reference 17) MPS2-R05f, March 2018 Upgrade and F&O Closure Peer Against R.G. 1.200 Rev. 2. MPS2-R05g Model, July 2018, Focused Scope Peer Reviews (ILRT Reference 21) The above peer reviews cover all ASME/ANS PRA Standard RA-Sa-2009 supporting requirements conducted in accordance with RG 1.200, Revision 2.

Attachment 3: Disposition and Resolution of Open Peer Review Findings

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
LE-C3-01 (2012)	LE-C3	Met	<u>2012 Peer Review Comment:</u> MPS2 did not review significant accident progression sequences resulting in a large early release to determine if repair of equipment can be credited. To move up to CCII, perform and document a review of significant accident progression sequences resulting in a large early release to determine if repair of equipment can be credited. If any such actions are identified, provide the basis for their feasibility. <u>Dominion disposition of 2012 Comment:</u> MPS2 LE.1 R5 Significant accident progression sequences have now been reviewed to determine if repair can help reduce LERF. However, no credit for repair is taken. <u>2018 Peer Review Comment:</u> The section has been provided and assessment performed to address the F&O. However, the current documentation does not address the current model results and is not completely closed. The process developed is sufficient to meet Category II for SR LE-C3.	Resolved. The significant accident progression sequences for the current model, MPS2-R05g, have been reviewed and documented in the LERF documentation to determine if repair can be credited to reduce LERF. Resolution does not impact the 50.69 LAR.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			Proposed Resolution: To close this F&O, the documentation of each interim update should include an evaluation that concludes that the current Model of Record results documentation reasonably represents the results documentation expected for the current interim model. The evaluation should also conclude that there are no new insights or conclusions to be obtained based on the changes that would otherwise require fully documenting the results to meet the necessary SRs.	
LE-C12-01 (2012)	LE-C12	Met	2012 Peer Review Comment: There is no evidence that MPS2 performed a review of accident progression accident sequences to determine if it was possible for continued operation of equipment or personnel that would reduce LERF. Dominion disposition of 2012 Comment: Significant accident progression sequences have now been reviewed to determine if continued equipment operation or operator actions after containment failure can help reduce LERF. The review is documented (see above referenced PRA notebooks). However, no credit is taken for additional equipment or	Resolved The significant accident progression sequences for the current model, MPS2-R05g, have been reviewed and documented in the LERF documentation for continued operation of equipment or personnel that would reduce LERF. Resolution does not impact the 50.69 LAR.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			operator actions because the significant LERF contributors are bypass scenarios, not a failure of containment isolation. 2018 Peer Review Comment: Documentation is provided and addresses the F&O by adding the appropriate Section and at that time there was a discussion addressing the top 97% of contributions to LERF. This process is still pending for the most current interim release, but the process exists and therefore the F&O as generated is considered partially closed. The associated SR LE-C12 requires an approach to be developed and is considered now met. Proposed Resolution: To close this F&O, the documentation of each interim update should include an evaluation that concludes that the current Model of Record results documentation reasonably represents the results documentation expected for the current interim model. The evaluation should also conclude that there are no new insights or conclusions to be obtained based on the changes that would otherwise require fully documenting the results to meet the necessary SRs.	

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
LE-F1-01 (2012)	LE-F1	Met	2012 Peer Review Comment: QU.2 Section 2.3.2 provided LERF by initiating event, Section 2.3.6 presented the dominant LERF cutsets, Section 2.3.9 presents the LERF importance analysis, and Table 15 presents the system contribution to LERF. Attachment 4 to QU.2 also presents the containment failure mode contribution to LERF. However, there is no identification of the contributors to LERF by plant damage states. 2018 Peer Review Comment: N/A Proposed Resolution: MPS2 needs to calculate and present the dominant LERF contributors to LERF by plant damage state.	Resolved. The LERF documentation has been updated for the current model, MPS2-R05g, to include the relative contribution to LERF from the plant damage states. Resolution does not impact the 50.69 LAR.
LE-F2-01 (2012)	LE-F2	Not Met	2012 Peer Review Comment: There is no evidence that MPS2 reviewed the LERF contributors for reasonableness. While MPS2 did group and present the LERF results by initiating event, system contribution and containment failure mode, there is no evidence that MPS2 reviewed the results for reasonableness.	Resolved to meet Capability Category I/II/III. The LERF contributors for the current model, MPS2-R05g, were reviewed for reasonableness and documented in the PRA LERF documentation.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			2018 Peer Review Comment: N/A Proposed Resolution: MPS2 needs to perform and document a review of the LERF results to show that they are explainable based on what would be expected for the plant.	Resolution does not impact the 50.69 LAR.
LE-G5-01 (2012)	LE-G5	Not Met	2012 Peer Review Comment: MPS2 has not reviewed their Level 2 analysis to identify any limitations that may have the potential to impact applications. MPS2 needs to review their Level 2 analysis to identify any limitations that may have the potential to impact applications. MPS2 should document the process used for the review and the results of the review. If no limitations are identified, this needs to be clearly stated. Dominion disposition of 2012 Comment: MPS2 LE.1 R5, section 7.0 PRA Notebook MPS2 LE.1 has been updated to include documentation of LERF limitations in Section 7.0 of the notebook. 2018 Peer Review Comment: The current section provides only general discussion and does not provide any support	Resolved to meet Capability Category I/II/III. This issue is associated with identifying the limitations associated with the LERF analysis for applications. Limitations have been identified for the LERF analysis, including applicable modes, the scope of hazards, and limitations with use of MAAP for Level 2. The Level 2 analysis contains sufficient scope and detail for this application, including consideration of various containment

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			for limitations when applying the LERF model. Limitations in MAAP should be included as well as any plant-specific modeling considerations. The SR LE-G5 remains not met. <u>Proposed Resolution:</u> The current section on limitations should be updated to reflect any new or additional limitations identified for the current model of record and the quantification results. The documentation should capture the insights gained from the uncertainty and sensitivity section and highlight possible limitations with respect to the use of the PRA for applications.	failure modes and plant damage state modeling, which allows the core damage sequences to be binned to the EPRI accident classes. A more thorough documentation of LERF analysis limitations that accounts for the items identified in the proposed resolution would not impact the conclusions of this assessment. Resolution does not impact the 50.69 LAR.
SC-A5-01 (July 2018)	SC-A5	Met	<u>2018 Peer Review Comment:</u> The success path for SGTR includes throttling HPSI to maintain RWST inventory until SDC can be achieved for safe and stable conditions and requires 32 hours to achieve. The PRA model is based on a 24 hour mission time. Also, it is not clear how long the RWST can provide inventory <u>Assessment Basis</u> MPS2-SC.1 identifies the base mission time	Open Before implementation, the MPS2 PRA internal events model of the SGTR accident sequence will be revised to remove credit for achieving safe and stable conditions at 32 hours.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			as safe and stable within 24 hours. No events were identified requiring longer than 24 hours, however MPS2-SC.2 identifies that one SGTR sequence requires 32h to reach safe and stable and includes an evaluation that the impact on CDF is not significant. Proposed Resolution Provide additional analysis to demonstrate that safe and stable conditions can be achieved by 24 hours, revise the model to account for the extra time, or add a recovery action such as refilling the RWST.	
IFQU-A7-1 (March 2018)	IFQU-A7 IFQU-B1	Met Met	2018 Peer Review Comment: Detailed quantification notebook is not available for the MC documentation. To meet this requirement, a detailed QU notebook will be required. Assessment Basis: The quantification process met the requirements of QU SR groups A-E. Because these results were not contained in the model update documentation, a finding was given to create a detailed QU that addresses HLR F SRs. Proposed Resolution: Generate a detailed QU notebook either for	Resolved. The MPS2 PRA documentation has been updated to address the supporting requirements in IFQU-A7 and IFQU-B1 based on the latest model, MPS2-R05g. Resolution does not impact the 50.69 LAR.

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69
			the integrated PRA model or for the flooding specific model that meets the requirements of IFQU and QU of the standard.	
QU-F2-02 (March 2018)	QU-F2 QU-F3	Met Met	<u>2018 Peer Review Comment:</u> Dominions' PRA update process periodically creates a new "model of record" that addresses the requirements of QU-F2 & QU-F3. However interim updates are performed to maintain the PRA consistent with the as-built/as-operated plant that do not address all of requirements of QU-F2 & QU-F3 to document sensitivities, uncertainty assessments and significant contributors. <u>Proposed Resolution:</u> The full quantification analysis, which addresses the SRs for QU-F2 and QU-F3, should be updated to for use with risk informed submittals or peer reviews, or justification should be included in the documentation for the interim quantification results detailing why elements of the previous quantification analysis still apply to the interim model results.	Resolved. The MPS2 PRA documentation has been updated to address the supporting requirements in QU-F2 and QU-F3 based on the latest model, MPS2-R05g. Resolution does not impact the 50.69 LAR.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Aircraft Impact	Y	PS2	Airport hazard meets 1975 SRP requirements. Airports, military installations and flight corridors around MP2 (including Groton Airport) have been considered. Evaluations of aircraft impact associated with these facilities find that it does not pose a significant hazard.
Avalanche	Y	C3	Not applicable to site because of topology. MP2 is located on the Long Island Sound with no hilly or mountainous terrain near the site. Avalanches are not a viable external initiator.
Biological Event	Y	C1, C5	Plant design accounts for biological growth. Slowly developing growth can be detected and mitigated by surveillance. The circulating water system intake structure incorporates several features to control biological fouling including trash racks and traveling screens, a cutoff wall to prevent ecologically rich surface water from entering the system, exit passages for fish are provided, vertical guides allow individual channels to be drained and a chlorination system for biocide treatments.
Coastal Erosion	Y	C5	Slowly developing event can be detected and mitigated by surveillance.
Drought	Y	C1, C5	Plant design eliminates drought as a concern and event is slowly developing. The UHS is the Long Island Sound which is unaffected by drought since it communicates with the Atlantic Ocean.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
External Flooding	Y	PS2 (Note b)	The IPEEE documented that most external flooding hazards meet the 1975 SRP requirements or the plant is designed against the hazards. As part of the NRC 10 CFR 50.54(f) request on Reevaluation of External Floods, Dominion Energy is in the process of evaluating the external flooding hazard at Millstone, which includes storm surge, water ponding, local intense precipitation, etc. Currently there are no identified plant modifications or deficiencies that would preclude screening of external flooding hazard. As part of the reevaluation, any identified discrepancies will be tracked in the corrective action program.
Extreme Wind or Tornado	Y	C1, PS4	The wind loadings for all structures are based on American Society of Civil Engineers Paper 3269, "Wind Forces on Structures". The basic design wind velocity for MP2 Class 1 structures is 115 mph with gusts up to 140 mph. MP2 structures are designed for tornados having a maximum rotational velocity of 300 mph and a maximum translational velocity of 60 mph. This design basis tornado has a frequency less than 1E-6/yr at MP2. Failure of a service water pump due to missile strike is bounded by 1E-6/yr. Failure of both diesels due to missile strike is bounded by 1E-6/yr. Failure of the EDG Room ventilation due to tornado is bounded by 1E-6/yr. CDF due to tornado-induced failure of the East 480V Switchgear Room

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			ventilation is bounded by 1E-6/yr. Control Room ventilation failure is not a significant contributor to risk because of alternate means of Control Room cooling and the ability to shut down outside the Control Room.
Fog	Y	PS2, C1	Fog can be a contributor to transportation accidents. Transportation accidents meet the criteria of the 1975 SRP. Deep draft boats must stay at least 2 miles offshore to avoid running aground on Bartlett Reef. Therefore, a boat that could cause significant damage to the Intake Structure is highly unlikely to collide with the Intake Structure and would, most likely, run aground first.
Forest or Range Fire	Y	C3	Site is cleared preventing fire from propagating onto the site and is not located in forested or grassland area.
Frost	Y	C4	Frost is covered under snow and ice hazards.
Hail	Y	C2	Loss of offsite power (LOOP) events associated with hail are addressed in the Internal Events PRA and the occurrence frequency is enveloped by the frequency of weather-induced LOOP events. Limited occurrence and bounded by other events for which the plant is designed.
High Summer Temperature	Y	C1, C5	Plant is designed for this hazard. Ventilation systems provide conditioned air in the plant to cool equipment. Weather-induced LOOP events are considered in the Internal Events PRA. Effects on the UHS are slow to develop if they develop at all

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			because of the size of the Long Island Sound.
High Tide, Lake Level, or River Stage	Y	C4	High tide is covered by external flooding considering storm surge.
Hurricane	Y	C4	Hurricane is covered by external flooding and high winds or tornado.
Ice Cover	Y	C1, C4	Plant is designed against freezing temperatures. Protection against ice blocking of roof penetrations will be demonstrated by Dominion. Ice blockage causing flooding is covered under external flooding.
Industrial or Military Facility Accident	Y	PS2	Explosive hazard impacts and control room habitability impacts meet 1975 SRP requirements (RG 1.78 and 1.91). Industrial facilities are too distant to pose a hazard to the safe operation of the plant. Nearby military facilities do not conduct operations that could potentially pose a hazard to the safe operation of the plant.
Internal Flooding	N	None	MPS2 has an internal flooding model.
Internal Fire	N	None	Addressed in Section 3.2.2.
Landslide	Y	C3	Not applicable to the site because of topography.
Lightning	Y	C1, C4	Lightning strikes causing loss of offsite power or turbine trip are contributors to the initiating event frequencies for these events. However, other causes are included. The impacts are no greater than those already modeled in the internal events PRA.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Additionally, MP2 does not have a specific vulnerability to lightning and does not have unique features that would create a high likelihood of failing safety-related systems, structures, or components concurrent with a LOOP.
Low Lake Level or River Stage	Y	C3	Not applicable to site because of location. MP2 is located on the coast of the Long Island Sound which is virtually unaffected by lack of precipitation.
Low Winter Temperature	Y	C1, C5	Plant is designed for this hazard. Potential pipe freezing is addressed by a requirement for heat tracing operability during cold weather. Impacts on the UHS are slow to develop, if it all, due to the size and salinity of the Long Island Sound.
Meteorite or Satellite Impact	Y	C2, PS4	Event occurrence frequency of meteorites greater than 100 lb. striking the plant is 7E-9/yr. This frequency is very low in absolute terms and lower than aircraft impacts. Aircraft impact damage envelops meteorite/satellite impact damage. Site is no more likely to be struck by meteorite/satellite than any other site.
Pipeline Accident	Y	C3	Pipelines are not close enough to significantly impact plant structures.
Release of Chemicals in Onsite Storage	Y	PS2	Plant storage of chemicals meets 1975 SRP requirements (RG 1.78 and 1.91). Control room habitability during postulated chemical releases has been evaluated and it has been determined that habitability is not threatened by this hazard.
River Diversion	Y	C3	Not applicable to the site because of location. There are no diversions near

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			MP2. Cooling water is supplied directly from the Long Island Sound.
Sand or Dust Storm	Y	C3	Not applicable to the site because of location. MP2 is not subject to sand or dust storms.
Seiche	Y	C3	Not applicable to the site because of location. Seiche in the Long Island Sound or the discharge basin was evaluated not to be a hazard for these bodies of water because of their geometry and locations relative to seiche inducing phenomena.
Seismic Activity	N	None	Addressed in Section 3.2.3.
Snow	Y	C1, C4	Event damage potential is less than other events for which the plant is designed. Potential flooding impacts covered under external flooding.
Soil Shrink-Swell Consolidation	Y	C1	Plant is designed for this hazard. The MP2 FSAR Chapter 2.7 describes the characteristics of the area geology, soil conditions, testing, foundations and backfill. Allowable bearing pressures for soil-supported structures are greater than contact pressures as determined by backfill testing. The potential for this hazard is low.
Storm Surge	Y	C4	Storm surge is covered by external flooding.
Toxic Gas	Y	C4	Toxic gas is covered by industrial or military facility accident, release of chemicals in on-site storage, and transportation accident. Control room habitability during postulated chemical

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			releases has been evaluated and it has been determined that habitability is not threatened by this hazard.
Transportation Accident	Y	PS2, C4	Potential hazards meet the 1975 SRP requirements. The hazards resulting from potential transportation accidents (i.e., highway, waterway, railroad, and air) were found to not contribute significantly to plant risk. Highway – The distance to the nearest highway exceeds the RG 1.91 safe distance criterion. Waterway – Most ships passing MP2 are deep draft and must remain at least 2 miles offshore to avoid running aground. Railroad – Hazardous materials are transported about 0.25 miles from the protected area. Most of the transported hazardous materials (chlorine, anhydrous ammonia, carbon dioxide and carbon disulfide) meet the RG 1.78 screening criterion for low transport frequency. The remaining transported hazardous material (propane) presents negligible potential for damage due to explosion and is not a threat to control room habitability due to toxic gas plume. Air – Covered under aircraft impact.
Tsunami	Y	C4	Covered under external flooding.
Turbine-Generated Missiles	Y	PS4	Bounding analysis is used to show CDF for turbine generated missiles is less than 1E-6/yr.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Volcanic Activity	Y	C3	Not applicable to the site because of location. There are no volcanos within the vicinity of MP2.
Waves	Y	C4	Waves are covered under external flooding.

Note a – See Attachment 5 for descriptions of the screening criteria.
Note b – External Flooding is being reevaluated in support of NRC 10 CFR 50.54(f) request.

Attachment 5: Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is < events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
	PS3. Design basis event mean frequency is < 1E-5/y and the mean conditional core damage probability is < 0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009	
	PS4. Bounding mean CDF is < 1E-6/y.	NUREG-1407 and ASME/ANS	

Event Analysis	Criterion	Source	Comments
		Standard RA-Sa-2009	
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty

Assumption/ Uncertainty	Discussion	Disposition
Containment Sump/Strainer Performance MPS2 currently models plugging of sump strainers based on data from the mid-1990s. Other Dominion models used data based on WCAP-16882, Rev 1, "PRA Modeling of Debris-Induced Failure of Long Term Core Cooling via Recirculation Sumps," Westinghouse Electric Co. Nov 2009.	SSCs involved in sump recirculation scenarios	The modeling is slightly non-conservative and is therefore considered a source of uncertainty. Sensitivity study will be performed using recent sump plug/blockage data.
Thermally induced failure of hot leg/SG tubes: The MPS2 TI-SGTR model is a simplified application of the NUREG-1570 methodology. The MPS2 model uses the conditional probability of TI-SGTR from NUREG-1570.	SSCs supporting mitigation of TI-SGTR scenarios.	The NUREG-1570 analysis has significant conservatisms (e.g., SG depressurization due to leaky MSIVs and large seal LOCAs clearing the loop seal and lower core barrel). Therefore, the NUREG-1570 TI-SGTR analysis is conservative in 50.69 application, since TI-SGTR is a significant contributor to LERF. A sensitivity study will be performed using less conservative data from EPRI TR-107623-V1 Rev. 1 "Steam Generator Tube Integrity Risk Assessment."
Equipment type code data includes successful post maintenance testing (PMT).	Inclusion of successful PMT demands can result in an under-estimation of the failure probability of a component type.	This was addressed by performance of sensitivity study that indicate inclusion of the number of assumed PMT demands has a small impact on CDF and, therefore, will not require any additional sensitivity study for system categorization.
Common cause failures are developed using available industry data.	This uncertainty potentially affects all SSCs evaluated during 50.69 categorization.	As directed by NEI 00-04, common cause basic events are increased to their 95th percentile and also decreased to their 5th

Assumption/ Uncertainty	Discussion	Disposition
		percentile values as part of the required 50.69 PRA categorization sensitivity cases. These results are capable of driving a component and respective functions HSS and, therefore, the uncertainty of the common cause failure probabilities are accounted for in the categorization process.

Serial No. 18-145
Docket No. 50-336
10 CFR 50.69 LAR

Enclosure 2

PROPOSED MARKED-UP MPS2 LICENSE PAGE

Dominion Energy Nuclear Connecticut, Inc.
(DENC)
Millstone Power Station Unit 2

Insert Here

D. This renewed operating license is effective as of its date of issuance and shall expire at midnight July 31, 2035.

FOR THE NUCLEAR REGULATORY COMMISSION

/ RA /

J. E. Dyer, Director

Office of Nuclear Reactor Regulation

Attachment:

1. Appendix A - Technical Specifications

Date of Issuance: November 28, 2005

(15) The licensee is approved to implement 10 CFR 50.69 using the processes for categorization of 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; and the Appendix R program to evaluate fire risk; and 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 the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, 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 Unit 2 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 a seismic margins approach to a seismic probabilistic risk assessment approach).