From: Klett, Audrey
Sent: Wednesday, October 20, 2021 1:28 PM
To: Stewart, Glenn H:(Exelon Nuclear)
Cc: thomas.loomis@exeloncorp.com
Subject: Supplement to Limerick 50.69 Audit Plan dated October 1, 2021 (L-2021-LLA-0042)
Attachments: Enclosure 1 (Agenda) to Limerick Audit Plan Supplement.docx; Enclosure 2 (Questions) to Limerick Audit Plan Supplement.docx

Glenn,

By letter dated March 11, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21070A412), as supplemented by letter dated May 5, 2021 (ADAMS Accession No. ML21125A215), Exelon Generation Company, LLC (the licensee) requested a license amendment to Renewed Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generation Station, Units 1 and 2 (Limerick) related to adoption 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."

This email serves as a supplement to NRC's Audit Plan dated October 1, 2021 (ADAMS Accession No. ML21263A248). The audit will now occur from November 15-17, 2021. The audit kick-off meeting will be held at 9:30 AM (Eastern Time) on November 15, 2021. A detailed schedule is provided in Enclosure 1 to this email. The NRC's audit will focus on the questions and requests contained in Enclosure 2 to this email. The NRC staff requests that any documentation to support the audit discussions be uploaded to an online portal at least 10 calendar days before the kick-off meeting.

The list of audit team members from the plan dated October 1, 2021, is updated as follows:

- Audrey Klett, Senior Project Manager (Audrey.Klett@nrc.gov) (new)
- Jeff Circle, APLA (Jeff.Circle@nrc.gov) (new)
- Shilp Vasavada, APLC (Shilp.Vasavada@nrc.gov (new)
- De (Wesley) Wu, APLC (De.Wu@nrc.gov) (new)
- ~~Alissa Neuhausen, APLC, NRC (Alissa.Neuhausen@nrc.gov)~~ (removed)

The staff is also requesting that the NRC staff and contractors' access to the online portal be extended to May 19, 2022.

Thank you,

Audrey Klett
Senior Project Manager
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Operating Reactor Licensing
Plant Licensing Branch 1
301-415-0489

Audrey Klett

Senior Project Manager
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Operating Reactor Licensing
Plant Licensing Branch 1
301-415-0489

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Recipients:
"thomas.loomis@exeloncorp.com" <thomas.loomis@exeloncorp.com>
Tracking Status: None
"Stewart, Glenn H:(Exelon Nuclear)" <Glenn.Stewart@exeloncorp.com>
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Enclosure 1: Updated Agenda for NRC Audit of Limerick 10 CFR 50.69 LAR

November 15-17, 2021

Day 1	9:30a-10a	Welcome/introductions
	10 a – 12 p	Licensee presentations: • Walk through the PWROG-20015-NP methodology for core damage alternate defense-in-depth • Presentation of pilot categorization (include discussion of rationale for system selection, and modeling in the probabilistic risk assessment (PRA)). • Walk through the categorization results for core damage alternate defense-in-depth • Review of PRA cutsets for core damage frequency Discussion of core damage alternate defense-in-depth Questions 11, 12, 13
	12 – 1 pm	**Break**
	1-4 p	Licensee presentations: • Walk through the PWROG-20015-NP methodology for containment alternate defense-in-depth • Walk through the categorization results for containment alternate defense-in-depth • Review of PRA cutsets for large early release frequency (licensee presentation) Discussion of containment alternate defense-in-depth Question 14
	4-4:15 pm	Daily summary and wrap-up with licensee.
Day 2	9:30a – 12 pm	Summary of previous day
		Licensee presentation: • Walk through the alternate passive categorization in Electric Power Research Institute topical report • Discussion of internal flooding PRA readiness • Walk through pilot categorization for passive categorization, and review of results • Review of internal flooding PRA cutsets Discussion of alternate passive methodology Discussion of Questions 4 through 10
	12 – 1 pm	**Break**
	1-4 pm	Alternate seismic discussion (Questions 1 through 3)
	4-4:15 pm	Daily summary and wrap-up with licensee.
Day 3	9:30a – 9:45 am	Summary of previous day
	9:45 a – 12 pm	Follow up on alternate seismic discussion (Questions 1 through 3), if needed
	12 – 1 pm	**Break**
	1-3:30 pm	Discussion of Question 15. Follow up questions and discussions, as needed
	3:30 pm	Exit meeting

Enclosure 2: NRC Audit Questions for Limerick 10 CFR 50.69 LAR

Questions on Alternative Seismic Approach

Question 01 – Completeness of Information Describing the Proposed Alternate Seismic Approach

In the LAR,[1] the licensee proposes to address seismic hazard risk using an alternative seismic approach based on the Electric Power Research Institute (EPRI) report 3002017583[2] (so-called Tier 1 approach in the EPRI report). The NRC staff understands that EPRI 3002017583 is an updated version of EPRI 3002012988[3] that was reviewed by the NRC staff in conjunction with the NRC's review of the Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2, LAR[4] for adoption of 10 CFR 50.69. The staff has not endorsed EPRI 3002012988 as a topical report for generic use. As such, the NRC staff performs a plant-specific review of the licensee's basis for the applicability of its proposed alternative seismic approach.

The NRC staff reviewed and approved CCNPP's alternative seismic approach based on the information in EPRI 3002012988 and information provided in the supplements to the CCNPP LAR[5,6,7]. The staff's safety evaluation[8] for the CCNPP LAR is identified as the precedent for the licensee's proposed approach in Section 2.1 of Enclosure 1 to the LAR.

In Section 2.1 of Enclosure 1 of the LAR, the licensee states that the Clinton Power Station (CPS) RAI response[9] is incorporated by reference into the LAR. The staff notes that the Limerick 10 CFR 50.69 LAR submittal and the incorporated-by-reference CPS response to DRA/APLC RAI 03 do not sufficiently describe the overall categorization process as it applies to

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[1] Rafferty-Czincila, S., Exelon Generation Company, LLC letter to U.S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Application to Implement an Alternate Defense-in-Depth Categorization Process, an Alternate Pressure Boundary Categorization Process, and an Alternate Seismic Tier 1 Categorization Process in Accordance with the Requirements of 10 CFR 50.69, 'Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors,'" dated March 11, 2021 (ADAMS Accession No. ML21070A412).

[2] Electric Power Research Institute (EPRI) 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization, Technical Update," February 2020.

[3] EPRI 3002012988, "Alternative Approaches for Addressing Seismic in 10CFR50.69 Risk-Informed Categorization," issued July 2018.

[4] Barstow, J., Exelon Generation Company, LLC letter to U.S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors," dated November 28, 2018 2019 (ADAMS Accession No. ML18333A022).

[5] Barstow, J., Exelon Generation Company, LLC letter to U.S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,'" dated July 1, 2019 (ADAMS Accession No. ML19183A012).

[6] Barstow, J., Exelon Generation Company, LLC letter to U.S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,'" dated July 19, 2019 (ADAMS Accession No. ML19200A216).

[7] Barstow, J., Exelon Generation Company, LLC letter to U.S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Response to Request for Additional Information Regarding the Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,'" dated August 5, 2019 (ADAMS Accession No. ML19217A143).

[8] Marshall, M.L., U.S. NRC letter to Hanson, B.C., Exelon Generation Company, LLC, "Calvert Cliffs Nuclear Power Plant, Units 1 and 2- Issuance of Amendment Nos. 332 and 310 Re: Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors (EPID L-2018-LLA-0482)," dated February 28, 2020 (ADAMS Accession No. ML19330D909).

[9] Simpson, P.R., Exelon Generation Company, LLC letter to U.S. Nuclear Regulatory Commission (NRC) Document Control Desk, "Response to Request for Additional Information Regarding License Amendment Requests to Adopt TSTF-505, Revision 2 and 10 CFR 50.69," dated November 24, 2020 (ADAMS Accession No. ML20329A433).

seismic considerations for Limerick. Specifically, Table 2 and Figure 1 in the LAR do not include sufficient description for NRC staff to make a finding on the plant-specific application of the proposed alternative seismic approach.

Explain whether the information in EPRI 3002012988 and in the supplements to the CCNPP LAR used to support the staff's review and approval of the CCNPP's alternative seismic approach is fully represented in EPRI 3002017583 and the licensee's LAR. If there is any gap between the two sets of information, then identify such gaps and justify their appropriateness for the licensee's proposed alternative seismic approach.

Question 02 – Relative Contribution of Seismic Risk for 10 CFR 50.69 Categorization

In Section 2.1 of Enclosure 1 to the LAR, the licensee identifies EPRI report 3002017583 as the basis for the proposed alternative seismic approach. Section 2.2.2, "Technical Basis for Approach," of EPRI report 3002017583 provides the underlying basis for the proposed alternative seismic approach (so-called Tier 1 approach in the EPRI report). As part of the technical basis, Section 2.2.2.1, "Integral Assessment," of EPRI Report 3002017583 states that:

> In addition, since the seismic hazards for these sites are low, the seismic CDF would also be expected to be low. **This is important** because the NEI 00-04 … categorization process includes an Integral Assessment that **weights the importance from each risk contributor** (for example, internal events, fire, SPRAs) by the fraction of the total core damage frequency contributed by that contributor …. This would further **reduce the likelihood of identifying a unique seismic condition** that would cause an SSC [system, structure, or component] to be designated HSS. (emphasis added)

In addition, Section 2.1 of Enclosure 1 to the LAR cites the SE for Calvert Cliffs Nuclear Power Plant, Units 1 and 2, as the precedent for this proposed approach. In Section 3.5.3.1 of that SE, under the section titled, "Evaluation of Applicability of Criteria for the Proposed Alternative Seismic Approach to Calvert Cliffs," the staff makes its finding that "the NRC staff concludes that the seismic risk contribution for Calvert Cliffs would not solely result in an SSC being categorized as HSS" after reviewing the relative contribution of seismic risk compared to the total plant risk for CCNPP.

The licensee's supplement dated May 5, 2021, provides nine points of consideration to support that seismic risk will not solely result in an HSS determination based on integrated importance measures. The staff identified the following concerns during its review:

- Item 2 discusses "limited unique seismic insights" identified by the case studies in EPRI report 3002017583. As noted by the licensee, the EPRI report did identify unique seismic insights, albeit limited, that were not captured by either the internal events PRA or the internal fire PRA.
- Item 3 does not address the relative contribution of seismic risk to the plant risk.
- Item 5 does not address the relative contribution of seismic risk to the plant risk.
- Item 9 identifies plant improvements that "are not explicitly captured in the previously discussed RICT LAR SCDF and SLERF estimates," but does not provide any information on the extent of reduction in SCDF and SLERF from those items and the resulting impact on the relative contribution of seismic risk to the plant risk for categorization under 10 CFR 50.69. In addition, the discussion in item 9 appears to be independent of the impact of a seismic event on the mitigation features listed in that item.

- The LAR and items 4, 6, 7, and 8 in the supplement dated May 5, 2021, do not to provide sufficient information to justify that the plant-specific seismic risk has a low relative contribution to the total plant risk such that the seismic risk contribution would not solely result in an SSC being categorized as HSS. Therefore, the appropriateness of the proposed alternative seismic approach for use in the licensee's previously approved categorization process remains unclear.

Please address the following:

A. In Item 4 of the supplement dated May 5, 2021, the licensee identifies four-division systems for AC and DC power, emergency diesel generators, residual heat removal pumps, and core spray pumps as one of the reasons to "support that seismic risk will not solely result in an HSS determination based on integrated importance measures." One of the unique concepts of seismic risk is correlated failures where redundant systems have the same conditional probability of failure given a seismic event of a particular magnitude. Based on the results of the case studies, Section 4 of the EPRI report 3002017583 includes the conclusion that "the seismic risk insights provided only limited unique insights into the 50.69 categorization process…those unique insights were generally associated with SSCs that would be treated as seismically correlated failures…." The discussion in item 4 does not include any mention of correlated failures of the identified systems or justification for why correlated failures of those systems are unlikely. Table 6 in the supplement dated May 5, 2021, provides the same HCLPF for all four emergency diesel generators (EDGs), thereby indicating correlated failure. Therefore, it is unclear to the NRC staff how the discussion in item 4 supports the low contribution of seismic risk to the total plant risk and the applicability of the proposed alternative seismic approach.

   i. Demonstrate that the four-division systems for AC and DC power, emergency diesel generators, residual heat removal pumps, and core spray pumps are not susceptible to seismically induced correlated failures. Include justification for why correlated failures are unlikely when the HCLPF values in Table 6 for these components are the same across the divisions.

   ii. If the four-division systems for AC and DC power, emergency diesel generators, residual heat removal pumps, and core spray pumps cannot be shown to be seismically uncorrelated, provide detailed justification for why seismic risk and its relative contribution to total plant risk would decrease compared to the available estimate in spite of the correlated failure of four-division systems.

B. In Item 6 of the supplement dated May 5, 2021, the licensee states that a review of the Limerick Severe Accident Risk Assessment (SARA) SPRA was performed to illustrate the margin beyond high confidence of low probability of failure (HCLPF) of 0.3g peak ground acceleration (PGA) used for the latest docketed seismic risk estimate for the plant. Table 6 in the supplement provides HCLPF values based on SARA for the SSCs in the preferred and alternate success paths from the licensee's Individual Plant Examination for External Events (IPEEE). Based on Table 6, the licensee states that a case can be made for a plant level seismic fragility of 0.4g PGA HCLPF. As identified by the licensee in the supplement, an assessment of the 1983 SARA is documented in NUREG/CR-3493 (ADAMS Accession No. ML071650308). Section 4.0 of NUREG/CR-3493 documents concerns raised regarding the seismic fragility analysis in the 1983 SARA and recommendations for resolving the concerns. Recommendations 5 through

11 in Section 4.1.3 of NUREG/CR-3493 raise concerns about the lack of plant specificity in the fragility analysis for several items listed in Table 6 of the supplement dated May 5, 2021, such as the electrical components, hydraulic control unit, nitrogen accumulator, diesel generator heating and ventilation, and the standby liquid control (SLC) test tank. In addition, Table 6 indicates that the HCLPF from the 1983 SARA for all four EDG and all four DC components is 0.4g PGA, which is the same as that proposed as the plant level seismic fragility for this application. It appears that there is no margin between the proposed and identified HCLPF values to account for unquantified variability in the fragility development. Therefore, it is unclear to the NRC staff that the values listed in Table 6 provide a defensible basis for a plant level seismic fragility of 0.4g PGA HCLPF.

i. As noted by the licensee, the 1983 SARA SPRA does not contain sufficient information to identify a plant-level HCLPF. The NRC staff's review of the information and insights from several contemporary SPRAs encompassing different locations, designs, and vintages (commencement of construction and operation) indicates plant-level fragilities in the range of 0.17g – 0.3g. It is unclear what differences in the plant's seismic design and construction support a higher plant-level HCLPF for the licensee compared to detailed SPRAs. Provide technical justification to support a higher plant-level HCLPF compared to the insights from a variety of contemporary SPRAs.

ii. Provide detailed justification for addressing the concerns raised in recommendations 1, 2, 3, 5, 6, and 7 in Section 4.1.3 of NUREG/CR-3493. Include a discussion of any plant-specific analysis performed after the 1983 SARA to address the concerns or justification for not performing such analysis.

iii. Provide detailed justification to support the fragility values from the 1983 SARA for the SLC tanks, nitrogen accumulator, and diesel generator heating and ventilation (HVAC) listed in Table 6 given the concerns raised in Section 4.1.3 of NUREG/CR-3493 (recommendations 8 and 10 for the SLC tanks; recommendation 9 for the nitrogen accumulator; and recommendation 11 for the diesel generator HVAC). Include a discussion of plant-specific fragility analysis performed after the 1983 SARA for these components or justification for not performing such analysis.

iv. Provide justification that the list of SSCs in Table 6 represents all the SSCs, including support systems, necessary to achieve safe shutdown in a seismic event and that all the SSCs, including support systems, have HCLPF above 0.4g PGA.

v. Table 6 indicates that the HCLPF from the 1983 SARA for all four EDGs and all four DC components is 0.4g PGA, which is the same as that proposed as the plant level seismic fragility for this application. Justify the proposed plant-level HCLPF of 0.4g PGA to account for unquantified variabilities in the fragility development.

vi. Justify the appropriateness of the proposed alternative seismic approach in the licensee's approved 10 CFR 50.69 program given the relative contribution of SCDF to the total plant CDF considering any changes to SCDF based on the responses to items (a) and (b) above.

C. In Item 7 of the supplement dated May 5, 2021, the licensee discusses conservatisms in the seismic conditional large early release probability (SCLERP) determination made for the licensee's RICT application. The licensee states that lower SCLERP values (in one case reduced by a factor of 4) would be more appropriate than those applied directly to core damage, unmitigated anticipated transient without scram (ATWS) accident sequences, and loss of containment heat removal sequences. However, the licensee has not provided any justification for the lower SCLERP values mentioned in item 7. For example, the licensee identifies scrubbing in the wetwell (i.e., suppression pool) and radionuclide filtration in the reactor enclosure. However, technical basis for the extent and effectiveness of the scrubbing and filtration is not provided to support the claim that the SCLERF should be reduced by a factor of 2. In addition, the high uncertainty in the scrubbing and reactor enclosure filtration phenomena is not addressed. Further, the impact of seismic events, especially on the timing of release, is not included in the discussion such as that for the loss of containment heat removal sequences.

i. Provide the technical basis for arriving at the estimate of 0.8 for SCLERP to reflect that some portion of the direct to core damage SCDF sequences do not directly result in a large and early release.

ii. Provide detailed justification for the reduction by a factor of 4 in the SCLERP for unmitigated anticipated transient without scram (ATWS). The justification should (i) describe the basis, including any analysis, for the extent of scrubbing in the wetwell and radionuclide filtration in the reactor enclosure, (ii) address the uncertainty in the scrubbing and reactor enclosure filtration phenomena, and (iii) explain why it is appropriate to qualitatively credit filtration for these sequences in this application when it is not credited in the licensee's internal events PRA.

iii. Provide detailed justification for the reduction by a factor of 2 in the SCLERP for loss of containment heat removal sequences. The justification should (i) describe the basis, including any analysis, for the extent of scrubbing in the wetwell and radionuclide filtration in the reactor enclosure, (ii) address the uncertainty in the scrubbing and reactor enclosure filtration phenomena, (iii) explain why it is appropriate to qualitatively credit scrubbing and filtration for these sequences in this application when it is not credited in the licensee's internal events PRA, and (iv) how the 5 percent probability of delay in the declaration of General Emergency is conservative for the impact of a seismic event on the timing of the release (i.e., late release sequences can become early due to seismically-induced failures).

D. In Item 8 of the supplement dated May 5, 2021, the licensee states that approximately one third of the calculated SLERF involves seismically induced severe damage states and that for the majority of the plant SSCs, this portion of the calculated SLERF is not influenced by whether or not an SSC is categorized as HSS or LSS. However, the basis for the fraction of sequences involving seismically induced severe damage states is not provided. In addition, the staff notes that its concern is related to HSS and LSS categorization being influenced by the relative contribution of SLERF rather than the reverse case, which appears to be discussed in Item 8.

Justify that "approximately one third of the calculated SLERF" involves seismically induced severe damage states and the discuss the resulting impact on SSC categorization in the proposed alternative seismic approach.

E. Item 9 of the supplement dated May 5, 2021, identifies plant improvements that “are not explicitly captured in the previously discussed RICT LAR SCDF and SLERF estimates,” but it does not provide any information on the extent of reduction in SCDF and SLERF from those items and the resulting impact on the relative contribution of seismic risk to the plant risk for categorization under 10 CFR 50.69. In addition, the staff notes that the discussion in item 9 appears to be independent of the impact of a seismic event on the mitigation features listed in that item. Justify the ability of mitigation features discussed in item 9 to reduce seismic risk considering the impact of a seismic event and the resulting reduction in SCDF and SLERF.

F. It appears that the relative contribution of SLERF to the total plant LERF would continue to remain high even after considering the licensee’s discussion in Items 7 and 8 in its supplement dated May 5, 2021. Justify the appropriateness of the proposed alternative seismic approach in the licensee’s approved 10 CFR 50.69 program given the relative contribution of SLERF appears to remain high to the total plant LERF, considering any changes due to the responses to Questions 3 through 5 above. Include justification or demonstration not included in the responses to Questions 3 through 5 above.

Question 03 – Use of Different Approaches for Seismic Risk Consideration in 10 CFR 50.69 Categorization

The NRC staff notes that the proposed license condition in Section 2 of the LAR states that categorization of systems can be performed using either the previously approved SMA approach or the proposed alternative seismic approach. Based on the information in the LAR, it is unclear to the NRC staff whether one approach will be used for categorizing an entire system. Clarify whether the licensee proposes to use a single approach (i.e., either SMA or, if approved, the proposed alternative seismic approach) for categorization of an entire system. If the licensee proposes to use different approaches for different components of the same system, justify the validity of such an implementation given the differences in the underlying basis for the approaches and the guidance in NEI 00-04 which indicates the use of a single approach for categorizing a system.

<u>Questions on Alternate Categorization Methodology for Pressure Boundary Components</u>

Question 04 – Results of the Alternate Pressure Boundary 10 CFR 50.69 Method

Section 50.69(b)(2)(ii) of 10 CFR requires that a LAR include 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. Section 50.69(b)(2)(iv) of 10 CFR requires that a LAR include 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 10 CFR 50.69(b)(2)(iv) of the Final 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 1 of EPRI 3002015999[10] states that a second report is anticipated, for 2020, that will provide background on additional investigations and providing implementation guidance. The NRC staff is unaware if this report has been published.

Section 3.1.3 of Enclosure 1 of the LAR states that the alternate EPRI approach was piloted in April 2020 for the entire plant and the results were reasonable and consistent. The NRC staff requests the following information:

A. Results of the alternate pressure boundary categorization. Include the specific difference in results between the existing approved method and the alternate EPRI 3002015999 method.

B. Discussion and justification that support the licensee's conclusion that the pilot categorization results are reasonable and consistent.

C. Identify any passive SSCs categorized by the existing approved method that have changed by applying the alternate method.

D. Provide the number and percentage of SSCs determined to be HSS based on criteria 1 through 10 and, separately, criteria 11 through 14 provided in Section 4.2 of EPRI 3002015999.

E. Provided details of any follow-up EPRI documentation related to the alternate pressure boundary categorization method. Include in this discussion if the Limerick, and any other plant, pilot results have been reviewed by EPRI to assess impact on the categorization results.

F. If additional relevant EPRI documents have been published (e.g., the second report identified in EPRI 3002015999), then provide them on the docket for NRC staff review of its applicability to this LAR.

Question 05 – Prerequisite No. 1 – PRA Technical Adequacy Internal Flooding Model

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 structures, systems, components (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

[10] EPRI 3002015999, "Enhanced Risk-Informed Categorization Methodology for Pressure Boundary Components," November 2019.

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 3.1.3 of Enclosure 1 of the LAR states that the EPRI 3002015999 methodology is proposed as an alternate to the NRC-accepted ANO-2 R&R-004[11] method for SSC categorization. Section 4.1 of the EPRI document specifies prerequisites or requirements that must be met prior to implementing the methodology, including the requirement of a robust internal events and internal flooding PRA (IFPRA). Section 3.1.3 of Enclosure 1 of the LAR states, for Prerequisite No. 1, that Limerick Generating Station (LGS) has a risk-informed in-service inspection (RI-ISI) program that is sufficient for use in categorization based on the 'gap' assessment provided in Section 4.1 of the EPRI report (6 clarifications) and the analysis provided in EPRI 1021467[12]. The NRC staff notes the analysis of EPRI 1021467 was performed against the 2008 ASME/ANS PRA Standard. It is unclear to the NRC staff which PRA model (i.e., the 2009 ASME/ANS PRA Standard[13] peer-reviewed IFPRA model (in accordance with RG 1.200[14]) or the RI-ISI application model) will be utilized in the LGS alternate passive categorization process.

A. Clarify which LGS PRA model (i.e., RG 1.200 or RI-ISI) will be utilized in the alternate passive categorization process.

B. If the LGS passive categorization process utilizes the RI-ISI PRA model, then:

   i. Provide a gap assessment of the RI-ISI PRA model to the ASME/ANS PRA Standard and the clarifications and qualifications in RG 1.200, Revision 3.

   ii. Provide details how the RI-ISI model meets the PRA update requirements of 10 CFR 50.69(e), "Feedback and Process Adjustment," including updating the model to address peer review Finding-level findings and observations (F&Os) and independent closure reviews performed in accordance with the Appendix X process.

   iii. Provide details and justification that all RI-ISI model key sources of uncertainty and assumptions have been identified and will be addressed with sensitivity studies in accordance with NEI 00-04.

   iv. Provide details and justification how the RI-ISI model will address non-piping SSCs, such as tanks, gaskets, and fittings, in the passive categorization process.

   v. Provide details and justification how the RI-ISI model will address inter-area propagation and barriers to inter-area propagation in the passive categorization process.

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[11] Markley, Michael, U.S. Nuclear Regulatory Commission, letter to Vice President, Operation, Arkansas Nuclear One, Entergy Operations, Inc., "Arkansas Nuclear One, Unit 2 – Approval of Request for Alternative AN0-2 R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 & 3 Moderate and High Energy Systems," dated April 22, 2009 (ADAMS Accession No. ML090930246).
[12] EPRI 1021467, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs," July 2010.
[13] American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) PRA standard ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", February 2009, New York, NY (Copyright).
[14] U.S. Nuclear Regulatory Commission, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," RG 1.200, Revision 3, December 2020 (ADAMS Accession No. ML20238B871).

vi. Provide details and justification how the RI-ISI model will address human-induced floods in the passive categorization process.

C. Appendix A to the LAR provides open F&Os related to IFPRA. The LAR appears to disposition the IFPRA F&O only with regard to the alternate defense-in-depth methodology, and there are no dispositions with regard to the alternate passive categorization method. For each open IFPRA F&O, provide a disposition for the alternate passive categorization method.

D. Provide a summary of changes, if any, performed in the IFPRA to support the alternate passive categorization method.

E. Discuss how comprehensive the IFPRA analysis is to model indirect effects (e.g., pipe whip, jet impingement, spray, inventory losses, etc.) and drain capacity and credited operator actions.

F. Discuss what pipe rupture frequency methodology is employed in the Limerick IFPRA, and justify why it is believed to be adequate to support the alternate passive categorization.

Question 06 – Prerequisite No. 2 – Integrity Management

Section 50.69(b)(2)(ii) of 10 CFR requires that a LAR include 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. Section 50.69(b)(2)(iv) of 10 CFR requires that a LAR include 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 10 CFR 50.69(b)(2)(iv) of the Final 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 3.1.3 of Enclosure 1 of the LAR states that the EPRI 3002015999 methodology is proposed as an alternate to the NRC-accepted ANO-2 R&R-004 method for SSC categorization. Section 4.1 of the EPRI document specifies prerequisites or requirements that must be met prior to implementing the methodology, including the requirement to have robust programs that address localized corrosion, flow-accelerated corrosion (FAC), and erosion. Section 3.1.3 of Enclosure 1 of the LAR states that for Prerequisite No. 2, LGS programs follow the guidance of all the references listed in the EPRI document. The NRC staff notes that there are other degradation methods (e.g., mechanical wear, fretting, fatigue, and stress corrosion cracking) that can impact the integrity of passive SSCs.

A. Provide justification that other passive component degradation mechanisms should not be addressed with regards to the component's integrity.

B. Alternatively to Part (A), provide justification that the exclusion of these degradation mechanisms does not impact the passive categorization process.

Question 07 – Prerequisite No. 3 - Protective Measures for Internal Flooding

Section 50.69(b)(2)(ii) of 10 CFR requires that a LAR include 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. Section 50.69(b)(2)(iv) of 10 CFR requires that a LAR include 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 10 CFR 50.69(b)(2)(iv) of the Final 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 3.1.3 of Enclosure 1 of the LAR states for Prerequisite No. 1, that internal flooding protective measures, such as floor drains and sumps, *<u>will be</u> considered LSS* (emphasis added) unless other evaluations determine the failure of these measures invalidate the LSS determination.

The NRC staff notes Section 4.1 of EPRI 3002015999, regarding internal flood (IF) protective measures (Prerequisite No. 3), states these measures *shall <u>not</u> be categorized as LSS* (emphasis added) unless they are evaluated and shown to not invalidate the HSS determinations provided in Section 4.2, “Predetermined HSS Passive SSCs.” It is unclear to the NRC staff how the Limerick approach is in alignment with the proposed EPRI alternate method.

A. Provide clarification how internal flooding protective measure SSCs will be categorized in the Limerick 10 CFR 50.69 categorization process. Include in this discussion how the Limerick approach is in alignment with the proposed EPRI alternate methodology.

B. If the Limerick process is not in alignment with the EPRI guidance, then provide justification that the Limerick approach does not impact categorization.

Question 08 – Mapping of All Functions for Passive SSCs

Section 50.69(b)(2)(ii) of 10 CFR requires that a LAR include 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. Section 50.69(b)(2)(iv) of 10 CFR requires that a LAR include 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 10 CFR 50.69(b)(2)(iv) of the Final 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 4 of NEI 00-04, regarding mapping of component to functions, states that for passive components, the pathway for each function is to be identified and analyzed for categorization.

Section 3.1.3 of Enclosure 1 of the LAR states that the EPRI 3002015999 methodology is proposed as an alternate to the NRC-accepted ANO-2 R&R-004 method for SSC categorization. Section 4.2 of the EPRI document provides 14 criteria for determining which passive SSCs are predetermined to be categorized as HSS. The final sentence of this section states, "All other safety-related and non-safety-related systems, subsystems, and components not classified as HSS in accordance with the preceding list shall be categorized LSS." It is not apparent to the NRC staff how the NEI 00-04 requirement of evaluating each function associated with a passive SSC is met in the proposed methodology given that not considering all functions in the categorization process could result in passive SSCs being prematurely categorized as LSS.

A. Provide details of how the proposed methodology addresses the categorization evaluation of each function associated with a passive SSC. Include in this discussion how the alternate method meets the function mapping requirements of NEI 00-04.

B. If the alternate method does not meet the function mapping requirements of NEI 00-04, then provide justification that the proposed method does not adversely impact categorizations of passive SSCs.

Question 09 – Pre-determined HSS SSCs Criteria

Section 50.69(b)(2)(ii) of 10 CFR requires that a LAR include 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. Section 50.69(b)(2)(iv) of 10 CFR requires that a LAR include 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 10 CFR 50.69(b)(2)(iv) of the Final 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 3.1.3 of Enclosure 1 of the LAR states that the EPRI 3002015999 methodology is proposed as an alternate to the NRC-accepted ANO-2 R&R-004 method for passive SSC categorization. Section 4.2 of the EPRI document provides criteria for identifying predetermined HSS SSCs.

A. Part (a) of the Criterion No. 1 specifies that the Class 1 portions of the reactor coolant pressure boundary (RCPB) can be classified LSS when the component failure allows the reactor to be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system. It is unclear to the NRC staff what the LGS definition/criteria for an orderly shutdown and cooldown (e.g., hot or cold shutdown) to be used in determining the applicability of Part (a).

   Provide details on what constitutes an acceptable shutdown and cooldown in an orderly manner for the alternate pressure boundary method. Include in this discussion the long-term requirement of the reactor coolant makeup system to achieve safe shutdown conditions.

B. Criterion No. 9 implies those passive heat exchangers whose failure does not allow reactor coolant to bypass primary containment can be categorized as LSS. It is unclear to the NRC staff if this criterion applies to all LGS plant modes of operation.

i. Provide clarification on the LGS plant modes that will be included in the passive categorization process.

ii. If not all LGS plant modes are included in the passive categorization process, then provide justification that the exclusion of those LGS plant modes does not adversely impact the categorization process.

C. Criterion Nos. 11, 12, and 13 provide several risk threshold values to determine when a passive SSC can be designated LSS. These three criteria refer to piping segments in determining the risk impact of piping failures.

It appears the calculation of initiating event frequency (IEF) when estimating the risk contribution of piping components will be determined based on pipe segments. However, the methodology does not explain how piping segments[15] will be defined when estimating its risk contribution. Consequently, the staff is unclear if the risk calculation will include the entire length of a passive system, those passive segments not screened as HSS (e.g., 'candidate' LSS) based on the first ten criteria of the alternate method, or a smaller segment.

In addition, with regards to Criterion Nos. 12 and 13, no justification is provided supporting the risk threshold values delineated in Figures 4-1 and 4-2. Hence, the NRC staff is unclear of the basis in choosing the risk threshold values for Criterion Nos. 12 and 13 and why they are appropriate for categorization of pressure-boundary SSCs.

Lastly, it appears that the risk calculations only utilize the internal flooding PRA model. The staff notes that other pressure boundary failure events, such as loss of coolant accidents (LOCAs), steam generator tube ruptures (SGTRs), main steam line breaks (MSLBs), and main feedwater line breaks (MFLBs) are addressed in the internal events (IE) PRA model. It is unclear to the NRC staff how the internal flooding PRA model can address 'candidate' LSS SSCs associated with internal event pressure boundary failures. Therefore:

i. Provide details of what constitutes a piping segment to be evaluated for Criterion Nos. 11, 12, and 13. If the definition differs from the NEI 00-04 endorsed definition, then include in this discussion a justification that the determination of piping segments does not adversely impact the categorization process.

ii. Provide details of how the IEF will be determined for the passive SSCs being evaluated in either Criterion Nos. 11, 12, or 13. If the analysis includes the entire length of a passive system, then include in this discussion only those portions screened as LSS by Criterion Nos. 1 through 10 of the alternate categorization method, or a smaller segment.

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[15] American Society of Mechanical Engineers (ASME) Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1," July 2002, defines a piping segment for categorization purposes

iii. Regarding Criterion Nos. 12 and 13, explain the basis for the risk threshold values of the proposed alternate pressure boundary categorization method. Include in this discussion justification that these threshold values do not adversely impact the categorization process.

iv. Provide clarification on what PRA hazard models are used to determine pressure boundary failure initiators associated with the three risk value criteria.

1. If the internal flooding PRA model is only used, then provide details and justification how the risk values of pressure boundary initiators not included in the internal flooding PRA will be calculated.

2. Alternatively to Part (1), provide justification that excluding the IE PRA from the alternate method does not adversely impact the categorization process.

D. Criterion No. 14 provides three considerations related to restraints or supports. The first consideration, 14.a, states that supports (for example, component support, hanger, or snubber) may remain uncategorized until a need is identified.

Section 3.1.3 of Enclosure 1 to the LAR states that the improved methodology represents a more efficient process since it is performed for all piping segments in the plant and not the system-by-system approach of the currently approved approaches.

It is unclear to the NRC staff how the first consideration is in alignment with the 'entire plant' analysis approach specified in the alternate method and what constitutes an identified 'need' (for example, whether the associated supports of a passive non-safety-related SSC categorized as HSS by the alternate method would be required to be categorized).

i. Provide the criteria to determine when categorization of supports is required.

ii. Provide justification that the exclusion of support categorization does not adversely impact plant safety.

E. NRC-accepted ANO-2 R&R-004 method for SSC categorization provides additional qualitative and defense-in-depth considerations, as summarized in Attachment 2 of ANO-2 RAI response dated January 12, 2009 (ADAMS Accession No. ML090120620). Section I-3.1 of ANO-2 R&R-004 method provides guidance in performing a hybrid qualitative/quantitative consequence evaluation to determine safety significance. Sections I-3.1.1 and I-3.1.2 of the ANO-2 R&R-004 method provide additional guidance with regards to postulated pressure boundary failure size, isolability of the break, including credit for automatic isolation and allowed operator actions, and evaluation for shutdown operations and external hazards. Further, Section I-3.2.2 provides several qualitative considerations for potential (e.g., candidate) LSS SSCs and, if met, would result in an HSS designation.

The proposed alternate method does not address all these qualitative considerations. Therefore, provide justification that the exclusion of the ANO-2 R&R-004 qualitative criteria does not adversely impact the categorization process. In the response, specifically discuss how each of these criteria are addressed by the proposed alternate method.

Question 10 – Pressure Boundary D-I-D

Section 50.69(b)(2)(ii) of 10 CFR requires that a LAR include 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. Section 50.69(b)(2)(iv) of 10 CFR requires that a LAR include 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 10 CFR 50.69(b)(2)(iv) of the Final 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 2.2 of Enclosure 1 to the LAR states that the licensee will either use the ANO-2 passive categorization method for Class 2 and 3 SSCs or will use the alternate method described in EPRI 3002015999. Section 3.1.2 of the same enclosure states that certain pressure boundary failure events will not be addressed by the alternate defense-in-depth method described in PWROG-20015-NP[16], but rather will be assessed by the pressure boundary categorization process (e.g., either by the ANO-2 method or by the EPRI 3002015999 method).

Section 3.3.3 of the enclosure to the NRC safety evaluation of the ANO-2 alternate pressure boundary categorization process states licensee personnel will verify that assigning each segment to the LSS category is not contrary to maintaining defense-in-depth. While EPRI 3002015999 states that the alternate pressure boundary method addresses defense-in-depth, insufficient information is provided in either the EPRI report or the LAR that explains how defense-in-depth is maintained for pressure boundary components classified as LSS.

The EPRI 3002015999 alternate pressure boundary method does not completely address D-I-D for all HSS considerations (e.g., passive candidate LSS SSCs). Therefore,

A. For utilizing the alternate EPRI 3002015999 alternate pressure boundary method, provide justification for how each pressure boundary component that is categorized LSS is evaluated maintaining the defense-in-depth philosophy.

B. If not all passive SSCs categorized as LSS are evaluated for defense-in-depth considerations, then provide justification that the exclusion of these SSCs from defense-in-depth consideration does not adversely impact the categorization process.

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[16] PWROG-20015-NP, "Alternate 10 CFR 50.69 Defense-in-Depth Categorization Process," PA-RMSC-1769, Revision 0, March 2021.

Enclosure 2: NRC Audit Questions for Limerick 10 CFR 50.69 LAR

<u>Questions on Alternate Defense-in-Depth Categorization Process</u>

Question 11 – PRA Technical Adequacy Prerequisites

Section 50.69(b)(2)(ii) of 10 CFR requires, in part, that a LAR include 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.

Step 1.b of PWROG-20015-NP states the following:

> "Findings related to the following ASME/ANS RA-Sa-2009 PRA Standard […] technical areas must be closed or dispositioned as not impacting the categorization process:
>
> 1) Accident sequence analysis
> 2) Success Criteria
> 3) Initiating Event Frequencies
> 4) Truncation
> 5) Common Cause Groupings

A. The ASME/ANS 2009 PRA standard has other areas related to FPIE PRA, such as: system analysis (SY), data analysis (DA), human reliability analysis (HR) and large early release (LE). Explain why the other technical elements in the PRA standard are not considered a pre-requisite for the applying the proposed alternate defense-in-depth approach.

B. With regards to items 1 and 2, explain whether all supporting requirements under AS (Accident Sequence) and SC (Success Criteria) technical elements of the ASME/ANS 2009 PRA standard are considered a pre-requisite. If not, then explain why not.

C. With regards to item 3, "initiating event frequencies," explain whether all supporting requirements under the initiating events (IE) technical element of the PRA standard are considered a pre-requisite.

D. Discuss whether and how future changes to the PRA standard will be taken into account.

Question 12 – Core Damage Defense in Depth steps

A. Step 7.a of PWROG-20015-NP states the following with regards to cutsets filtering:

> "Filter to only cutsets that have an initiating event and a single basic event representing a failure of an SSC, including an independent failure, a common cause failure, or a human failure event (HFE) which leads to core damage. Ensure cutsets that include flags, split fractions, and other house or special events with an initiating event and a single basic event are not discarded."

i. Explain how the HFEs are taken into account and whether a recovery action could preclude a cutset from being screened in. If so, explain how that conforms with the RG 1.174 defense-in-depth guidance of not overly relying on human actions.

ii. Provide a review of Limerick PRA cutsets that meet the HSS thresholds for core damage defense in depth.

B. Step 7.a.2 of PWROG-20015-NP states the following with regards to common cause failures: "Common cause failure groups that are greater than or equal to four can be screened out of the filtered cutsets." Justify the rationale for excluding common cause failure groups of 4 or more.

C. Step 7.b of PWROG-20015-NP states the following with regard to cutset quantitative screening: "Cutsets with initiating events with frequencies that are less than 1E-04/yr are not included in the alternate core damage defense-in-depth categorization process and can be screened out of the filtered cutsets."

i. Discuss whether and how uncertainty in initiating event frequency is taken into account in the alternate defense-in-depth categorization.

ii. Discuss how it is assured that initiating events would not be split into multiple initiating events of lower frequencies.

iii. List all initiating events in the Limerick PRA that have an initiating event frequency less than 1E-4/year.

D. Step 7.f of PWROG-20015-NP states the following: "Consistent with the existing NEI 00-04 defense-in-depth process, SSCs and functions outside the scope of the PRA do not need to be evaluated for core damage defense-in-depth since the level of defense-in-depth is based on the success criteria in the PRA."

i. Per NEI 00-04, defense-in-depth is to be applied to all SSCs, not only those modeled in the PRA. Therefore, please justify the statements in Step 7.f.

ii. For Limerick's categorization performed using NEI 00-04 guidance, discuss whether there were any SSCs not modeled in the PRA but marked HSS by the NEI 00-04 defense-in-depth approach. If so, describe them.

Question 13 – Defense-in-Depth First Order Core Damage Cutset Approach

Section 50.69(b)(2)(ii) of 10 CFR requires that a LAR include 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. Section 50.69(b)(2)(iv) of 10 CFR requires that a LAR include 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 10 CFR 50.69(b)(2)(iv) of the Final 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.

Sections 2.1.1.2 and 2.1.2.3 of RG 1.174[17] provide seven considerations of proposed licensing changes regarding defense-in-depth. The first consideration, “Preserve a reasonable balance among the layers of defense,” states the licensing change should not significantly reduce the effectiveness of a layer of defense. The fifth consideration, “Maintain multiple fission product barriers,” states the change should not significantly reduce the effectiveness of these barriers.

Section 3.1.2, “Alternate Defense-in-Depth Categorization Process,” of the LAR states that cutsets having an initiating event and a single basic event (BE) representing either an SSC, common cause failure (CCF), or human failure event (HFE) are to be evaluated for defense-in-depth categorization. It continues by stating that this process meets defense-in-depth guidance as discussed in PWROG-20015-NP.

Section 2.2.5 of the PWROG guidance states that a reasonable balance among the layers of defense is achieved in this method in that there is a reasonable confidence that SSCs will remain capable of performing their safety-related functions. Regarding fission product barriers, the guidance implies that the method maintains reasonable confidence that the barriers will perform their safety-related functions.

The defense-in-depth approach is to ensure that there are multiple layers (e.g., alternate success paths) of mitigation in responding to an event and that the failure of one layer is usually represented by a single BE. To provide a reasonable balance assessment, a licensee should evaluate all the layers in context of responding to an event, such as redundancy and diversity (no common cause failures across the layers of defense). The single BE approach (first order cutset) would either identify single point failures that represent a lack of defense-in-depth or common cause failure of several components that represent a lack of diversity. The layered approach of defense-in-depth would be represented by cutsets with two or more BEs. It is unclear to the NRC staff the basis for why this approach is adequate for defense-in-depth categorizations.

Table 4 of the LAR supplement dated May 5, 2021,[18] regarding functions identified as HSS by the alternate method, lists the low-pressure core injection (LPCI) mode and suppression pool cooling mode of the residual heat removal (RHR) system and providing air or gas to the automatic depression system (ADS) relief valves or other steam relief valves. These are backup functions to the other functions (e.g., feedwater, reactor core isolation cooling (RCIC), and high-pressure injection (HPCI)) that would normally require failure first. Therefore, the associated accident sequences for these functions would be represented in two or more ordered cutsets. It is unclear to the staff how these functions were determined to be HSS by the first order method of the alternate approach.

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[17] U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Revision 3, January 2018 (ADAMS Accession No. ML17317A256).

[18] Helker, D.P., Exelon Generation Company, LLC, letter to U.S. Nuclear Regulatory Commission (NRC) Document Control Desk, “Supplement - Application to Implement an Alternate Defense-in-Depth Categorization Process, an Alternate Pressure Boundary Categorization Process, and an Alternate Seismic Tier 1 Categorization Process in Accordance with the Requirements of 10 CFR 50.69, ‘Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors,’" dated May 5, 2021 (ADAMS Accession No. ML21125A215).

Therefore, the NRC staff requests the following:

A. Clarify if the intent of the alternate defense-in-depth method is to assign 'candidate' HSS to SSCs that provide only one layer of defense to an event (e.g., no defense-in-depth exists).

B. Regarding SSCs that have only one back-up function that is not diverse, explain and justify why the SSC should not be categorized as HSS. Include in this discussion how this explanation is in accordance with the seven defense-in-depth considerations of RG 1.174.

C. With regards to the answer to Parts A and B above, provide justification that this approach does not adversely impact the categorization process.

D. Explain and justify how the functions provided in Table 4 of the supplement dated May 5, 2021, were determined by the alternate method using first order cutsets.

E. Tables 3 and 5 of LAR supplement dated May 5, 202,1 provide 11 functions that were categorized as LSS by the proposed alternate defense-in-depth, which were previously identified as HSS based on the NEI 00-04, Chapter 6, "Defense-in-depth Assessment." Three of the functions appear to have changed to LSS because of the alternate core damage defense-in-depth methodology. These include one function for each of the following systems: core spray, reactor enclosure HVAC (REHVAC), and control enclosure HVAC (CEHVAC). The LAR supplement states that sufficient redundancy and diversity exists.

   For each of these three functions, provide a summary of the available redundancy and diversity. Discuss the categorization results based on NEI 00-04. Explain whether the listed diversity is credited in the Limerick design basis.

Question 14 – Defense-in-Depth First Order Large Early Release Cutset Approach and Screening

Section 50.69(b)(2)(ii) of 10 CFR requires that a LAR include 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. Section 50.69(b)(2)(iv) of 10 CFR requires that a LAR include 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 10 CFR 50.69(b)(2)(iv) of the Final 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.

Sections 2.1.1.2 and 2.1.2.3 of RG 1.174 provide seven considerations of proposed licensing changes regarding defense-in-depth. The first consideration, "Preserve a reasonable balance among the layers of defense," states the licensing change should not significantly reduce the effectiveness of a layer of defense. The fifth consideration, "Maintain multiple fission product barriers," states the change should not significantly reduce the effectiveness of these barriers.

Section 3.1.2 of Enclosure 1 to the LAR states that the process used for core damage defense-in-depth is the same as the one used for containment defense-in-depth with the exception that the FPIE LERF PRA model will be used as discussed in PWROG-20015-NP, and each system categorized continues using the guidance in NEI 00-04, Section 6.2, "Long-Term Containment Integrity."

Section 2.2.5 of the PWROG guidance states that a reasonable balance among the layers of defense is achieved in this method because there is a reasonable confidence that SSCs will remain capable of performing their safety-related functions. Regarding fission product barriers, the method is stated to maintain reasonable confidence that the barriers will perform their safety-related functions.

A. Step 8.b.1 of the PWROG guidance states to only filter cutsets with a single BE (e.g., SSC failure, CCF, or HFE) that leads to containment failure. The NRC staff notes that core damage cutsets that lead to plant damage state (PDS) that can proceed to a large early release event usually contains one or more failures. It is unclear to the NRC staff if the containment defense-in-depth approach applies the filtering process to the core damage cutset failures or to the containment-related failures.

    i. Clarify if the alternate containment defense-in-depth filtering in this step includes SSC failures that lead to core damage. Include in this discussion if the alternate method performs any filtering based on the associated core damage cutset.

    ii. If the alternate method containment filter approach includes core damage SSC failure, then provide details of the filtering and justify this approach does not adversely impact the categorization process for containment considerations.

B. The single BE approach (first order cutset) will either identify single point failures that represent a lack of defense-in-depth or common cause failure of several components that represent a lack of diversity. The layered approach of defense-in-depth would be represented by cutsets with two or more BEs. It is unclear to the NRC staff the basis of why this approach is adequate for defense-in-depth categorizations.

    i. Clarify if the intent of the alternate defense-in-depth method is to assign 'candidate' HSS to SSCs that provide only one layer of defense to an event (e.g., no defense-in-depth exists).

    ii. Regarding SSCs that have only one back-up function that is not diverse, explain and justify why the SSC should not be categorized as HSS. Include in this discussion how this explanation is in accordance with the seven defense-in-depth considerations of RG 1.174.

    iii. With regards to the answer to Parts i and ii above, provide justification that this approach does not adversely impact the categorization process.

C. Step 8.b.2.a of the PWROG guidance states that cutsets with IEFs less than 1E-04 per year may be screened from the alternate defense-in-depth approach. Section 6.2 of NEI 00-04 provides guidance for considering containment bypass, such as intersystem LOCAs (ISLOCAs), considerations in determining passive SSC categorization. Containment bypass events usually have an IEF < 1E-04/year, yet are significant

contributors to LERF risk. It is unclear to the staff how the PWROG alternate approach includes this consideration in the categorization of passive SSCs related to containment bypass. Therefore:

i. Describe how the alternate containment defense-in-depth approach assesses containment bypass events. Include in this discussion the treatment of cutsets that do not contain a containment-related SSC failure that was not part of the core damage cutset.

ii. Provide justification that the screening of a containment bypass or containment isolation failure associated with an IEF < 1E-04/year does not adversely impact the categorization process.

iii. Alternatively to Part (ii), describe and justify another screening mechanism to ensure the appropriate containment bypass or containment isolation failures sequences are assessed in the categorization process.

D. The guidance on containment defense-in-depth in Chapter 6 of NEI 00-04 contains the following questions to decide whether SSCs are to be HSS that address containment isolation:

Containment Isolation

- Does the SSC support containment isolation for containment penetrations that are:
    - Directly connected to containment atmosphere, and
    - > 2" in diameter, and
    - not locked closed or only locally operated?
- Does the SSC support containment isolation for containment penetrations that are:
    - Part of the reactor coolant system pressure boundary, and
    - > 3/8" in diameter, and
    - not locked closed or only locally operated?

Describe how the containment penetrations are modeled in the Limerick PRA. Describe whether and, if yes, how the above considerations on containment isolation defense-in-depth from NEI 00-04 are addressed by the new proposed alternate defense-in-depth.

E. Several of the Table 5 entries of the LAR supplement dated May 5, 2021, regarding percentages categorized as HSS and LSS by the alternate method, state that previously categorized HSS SSCs are now LSS based on their contribution to LERF. It is unclear to the staff how these functions were determined to be LSS by the containment assessment of the alternate approach. Therefore, explain and justify how the functions identified in Table 5 of the supplement dated May 5, 2021, were categorized using the alternate containment defense-in-depth method.

Other Questions

Question 15 – Open Phase Condition

Section C.1.4 of Regulatory Guide (RG) 1.200 states the base (e.g., Model of Record) PRA is to represent the as-built, as-operated plant to the extend needed to support the application. The licensee is to have a process that identifies updated plant information that necessitate changes to the base PRA model.

In response to the January 30, 2012, open phase condition (OPC) event at the Byron Generating Station, the NRC issued Bulletin 2012-01.[19] As part of the initial voluntary industry initiative (VII) for mitigation of the potential for the occurrence of an OPC in electrical switchyards,[20] licensees have added an open phase isolation system (OPIS). Per SRM-SECY-16-0068,[21] the NRC staff was directed to ensure that licensees have appropriately implemented OPIS and that licensing bases have been updated accordingly. Inspections of OPIS by NRC staff are currently underway. From the revised voluntary initiative[22] and resulting industry guidance in NEI 19-02[23] on estimating OPC and OPIS risk, the risk impact of an OPC can vary widely depending on electrical switchyard configuration and design. In light of these observations, provide the following information:

A. Discuss LGS's evaluation of the risk impact associated with OPC events, including the likelihood of OPC initiating plant trips and the impact of those trips on PRA-modeled SSCs. Also, explain whether an OPIS has been installed at LGS and, if it has been installed, then discuss its functionality and any operator actions needed to operate the system or needed in response to the system.

B. Clarify whether any installed OPIS equipment and associated operator actions are credited in the PRAs that support this application. If OPIS equipment and associated operator actions are credited, then provide the following information:

   i. Describe the OPIS equipment and associated actions that are credited in the PRA models.

   ii. Describe the impact that this treatment, if any, has on key assumptions and sources of uncertainty for the categorization process.

   iii. Discuss HRA methods and assumptions used for crediting OPIS alarm manual response.

   iv. Discuss how OPC related scenarios are modelled for non-internal event scenarios such as fire, seismic, flooding, high winds, tornado, and other external events.

[19] U.S. NRC Bulletin 2012-01, "Design Vulnerability in Electric Power System" (ADAMS Accession No. ML12074A115).

[20] Anthony R. Pietrangelo to Mark A. Satorius, Ltr re: "Industry Initiative on Open Phase Condition - Functioning of Important-to-Safety Structures, Systems and Components (SSCs)", dated October 9, 2013 (ADAMS Accession No. ML13333A147).

[21] U.S. NRC SRM-SECY-16-0068, "Interim Enforcement Policy for Open Phase Conditions in Electric Power Systems for Operating Reactors," dated March 9, 2017 (ADAMS Accession No. ML17068A297).

[22] Doug True to Ho Nieh, Ltr re: "Industry Initiative on Open Phase Condition, Revision 3", dated June 6, 2019 (ADAMS Accession No. ML19163A176)

[23] Nuclear Energy Institute (NEI) 19-02, "Guidance for Assessing Open Phase Condition Implementation Using Risk Insights", Revision 0, April 2019 (ADAMS Accession No. ML19122A321).

v. Regarding inadvertent OPIS actuation:

a. Explain whether scenarios regarding inadvertent actuation of the OPIS, if applicable, are included in the categorization process PRA models.

b. If inadvertent OPIS actuation scenarios are not included in the PRA models, then provide justification that the exclusion of this inadvertent actuation does not impact the categorization process.

C. If OPC and OPIS are not included in the application PRA models (whether OPIS equipment is installed or not), then provide justification the exclusion of this failure mode and mitigating system does not impact this categorization process.

D. As an alternative to Part C, describe how the licensee ensures that OPC-related scenarios are incorporated into the application PRA models prior to implementing the categorization process.