UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 5, 2020

Mr. Bryan C. Hanson
Senior Vice President
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President and Chief Nuclear Officer
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4300 Winfield Road
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SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 & 3 – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2019-JLD-0016)

Dear Mr. Hanson:

The purpose of this letter is to document the staff's evaluation of the Dresden Nuclear Power Station, Units 2 & 3 (Dresden), seismic probabilistic risk assessment (SPRA) which was submitted in response to Near-Term Task Force (NTTF) Recommendation 2.1 "Seismic." The U.S. Nuclear Regulatory Commission (NRC) has concluded that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required for Dresden.

By letter dated March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12053A340), the NRC issued a request for information under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(f) (hereafter referred to as the 50.54(f) letter). The request was issued as part of implementing lessons learned from the accident at the Fukushima Dai-ichi nuclear power plant. Enclosure 1 to the 50.54(f) letter requested that licensees reevaluate seismic hazards at their sites using present-day methodologies and guidance. Enclosure 1, Item (8), of the 50.54(f) letter requested that certain licensees complete an SPRA to determine if plant enhancements are warranted due to the change in the reevaluated seismic hazard compared to the site's design-basis seismic hazard.

By letter dated October 30, 2019 (ADAMS Accession No. ML19304B567), Exelon Generation Company, LLC (Exelon, the licensee) provided an SPRA submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter, for Dresden. The October 30, 2019, report was supplemented by letter dated March 6, 2020 (ADAMS Accession No. ML20066K784). The NRC staff assessed the licensee's implementation of the Electric Power Research Institute's Report 1025287, "Seismic Evaluation Guidance - Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML12333A170), as endorsed by NRC letter dated February 15, 2013 (ADAMS Accession No. ML12319A074), through the completion of the reviewer checklist in Enclosure 1 to this letter. As described below, the NRC has concluded that the Dresden SPRA submittal meets the intent of the SPID guidance and that the results and risk insights provided by the

SPRA support the NRC's determination that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required.

BACKGROUND

The 50.54(f) letter requested, in part, that licensees reevaluate the seismic hazards at their sites using updated hazard information and current regulatory guidance and methodologies. The request for information and the subsequent NRC evaluations have been divided into two phases:

> **Phase 1**: Issue 50.54(f) letters to all operating power reactor licensees to request that they reevaluate the seismic and flooding hazards at their sites using updated seismic and flood hazard information and present-day regulatory guidance and methodologies and, if necessary, to request they perform a risk evaluation.
>
> **Phase 2**: Based upon the results of Phase 1, the NRC staff will determine whether additional regulatory actions are necessary (e.g., updating the design basis and structures, systems, and components important to safety) to provide additional protection against the updated hazards.

By letter dated March 31, 2014 (ADAMS Accession No. ML14091A012), the licensee submitted the reevaluated seismic hazard information for Dresden. The NRC performed a staff assessment of the submittal and issued a response letter on April 27, 2015 (ADAMS Accession No. ML15097A519). The NRC's assessment concluded that the licensee conducted the hazard reevaluation using present-day regulatory guidance and methodologies, appropriately characterized the site, and met the intent of the guidance for determining the reevaluated seismic hazard.

By letter dated October 27, 2015 (ADAMS Accession No. ML15194A015), the NRC documented a determination of which licensees were to perform: (1) an SPRA; (2) applicable limited scope evaluations; or (3) no further actions based on, among other factors, a comparison of the reevaluated seismic hazard and the site's design-basis earthquake. As documented in that letter, Dresden was expected to complete an SPRA with an estimated completion date of June 30, 2019, which would also assess high frequency ground motion effects. By letter dated May 14, 2018 (ADAMS Accession No. ML18134A224), the licensee requested to extend the SPRA submittal from June 30, 2019, to December 31, 2019. The staff responded approving this extension in a letter dated September 17, 2018 (ADAMS Accession No. ML18236A262). The Dresden SPRA report was submitted to the NRC in a letter dated October 30, 2019 (ADAMS Accession No. ML19304B567). Dresden was expected to complete a limited-scope evaluation for the spent fuel pool, and it was submitted by letter dated August 31, 2016 (ADAMS Accession No. ML16244A801). The staff provided its assessment of the limited-scope evaluation in a letter dated November 8, 2016 (ADAMS Accession No. ML16291A021).

The completion of the April 27, 2015, NRC staff assessment for the reevaluated seismic hazard and the scheduling of Dresden SPRA submittal described in the NRC's October 27, 2015, letter marked the fulfillment of the Phase 1 process for Dresden.

In its October 30, 2019, letter, the licensee provided the SPRA submittal that initiated the NRC's Phase 2 decisionmaking process for Dresden.

The NRC described this Phase 2 decisionmaking process in a guidance memorandum from the Director of the Division of Operating Reactor Licensing to the Director of the Office of Nuclear Reactor Regulation (NRR) dated March 2, 2020 (ADAMS Accession No. ML20043D958). This memorandum details a Senior Management Review Panel (SMRP) consisting of three NRR Division Directors that are expected to reach a screening decision for each plant submitting an SPRA. The SMRP is supported by appropriate technical staff who are responsible for consolidating relevant information and developing the recommendation for the screening decisions for consideration by the panel. In presenting recommendations to the SMRP, the supporting technical staff is expected to recommend placement of each SPRA plant into one of three groups:

1) **Group 1** includes plants for which available information indicates that further regulatory action is not warranted. For seismic hazards, Group 1 includes plants for which the mean seismic core damage frequency (SCDF) and mean seismic large early release frequency (SLERF) clearly demonstrate that a plant-specific backfit would not be warranted.

2) **Group 2** includes plants for which further regulatory action should be considered under the NRC's backfit provisions. This group may include plants with relatively large SCDF or SLERF, such that the event frequency in combination with other factors results in a risk to public health and safety for which a regulatory action is expected to provide a substantial safety enhancement.

3) **Group 3** includes plants for which further regulatory action may be needed, but for which more thorough consideration of both qualitative and quantitative risk insights is needed before determining whether a formal backfit analysis is warranted.

The evaluation performed to provide the basis for the staff's grouping recommendation to the SMRP for Dresden is described below. Based on its evaluation, the staff recommended to the SMRP that Dresden be classified as a Group 1 plant and therefore, no further regulatory action was warranted.

EVALUATION

Upon receipt of the licensee's October 30, 2019, SPRA submittal, a technical team of staff performed a completeness review to determine if the necessary information to support Phase 2 decisionmaking had been included in the licensee's submittal. The technical team performing the review consisted of staff experts in the fields of seismic hazards, fragilities evaluations, and plant response/risk analysis. On November 29, 2019 (ADAMS Accession No. ML19333B896), the technical team determined that sufficient information was available to perform the detailed technical review in support of the Phase 2 decisionmaking.

As described in the 50.54(f) letter, the staff's detailed review focused on verifying the technical adequacy of the licensee's SPRA such that an appropriate level of confidence could be placed in the results and risk insights of the SPRA to support regulatory decisionmaking associated with the 50.54(f) letter. As stated in its October 30, 2019, submittal, the licensee developed and documented the SPRA in accordance with the SPID guidance, including performing a peer review against Part 5 of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS), "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications."

The peer review of the Dresden SPRA was against Part 5 of Addendum B of the PRA Standard (RA-Sb-2013). Appendix A of the licensee's submittal provided a summary of the full-scope peer review, including excerpts from the corresponding peer review report. Appendix A included the open SPRA finding level facts and observations (F&Os) along with licensee's dispositions which were reviewed by NRC staff in the context of the regulatory decisionmaking associated with the 50.54(f) letter.

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the 50.54(f) letter. The list of applicable licensees in Enclosure 1 of the July 6, 2017, letter included Exelon as the licensee for Dresden. The staff exercised the audit by reviewing licensee documents via an electronic reading room (ePortal) as documented in Enclosure 3 to this letter.

The staff developed questions to verify information in the licensee's submittal and to gain an understanding of non-docketed information that supports the docketed SPRA submittal. The staff's clarification questions (ADAMS Accession Nos. ML19333B896 and ML20035F145, respectively), were sent to the licensee to support the audit. The licensee subsequently provided answers to the questions in the ePortal, which the staff reviewed.

The staff determined that the answers to the questions provided in the ePortal served to verify statements that the licensee made in its October 30, 2019, SPRA submittal and March 6, 2020, supplement. The findings from the licensee's internal events PRA, which form the base for the SPRA, were not provided in the submittal. As part of the audit, the NRC staff requested information on the internal events PRA findings. Based on the information provided by the licensee, the staff identified a set of findings which had the potential to impact the SPRA. The NRC staff requested, via the audit, additional information to verify the appropriateness and impact of the dispositions on this submittal. In response, the licensee provided relevant information, including the results of sensitivity studies. Based on its review, the staff determined that its decision on the SPRA in the context of this submittal would not be impacted by the dispositions of the relevant findings from the internal events PRA. The licensee also provided the results of a review to assess the disposition of SPRA findings and concluded that the proposed resolutions would not impact the conclusions of the SPRA staff review.

The staff's review process included the completion of the SPRA Submittal Technical Review Checklist (SPRA Checklist) contained in Enclosure 1 to this letter. As described in Enclosure 1, the SPRA Checklist is a document used to record the staff's review of licensees' SPRA submittals against the applicable guidance of the SPID in response to the 50.54(f) letter. The SPRA Checklist also focuses on areas where the SPID contains differing guidance from standard industry SPRA guidance. Enclosure 1 contains the staff's application of the SPRA Checklist to Dresden's submittal. As documented in the checklist, the staff concluded that the Dresden SPRA met the intent of the SPID. The staff further concluded that the peer review was done in accordance with the ASME/ANS Standard RA-Sb-2013 process.

Based on the staff's review, the NRC staff concluded that the technical adequacy of the licensee's SPRA submittal was sufficient to support regulatory decisionmaking associated with Phase 2 of the 50.54(f) letter.

Following the staff's conclusion on the technical adequacy of the SPRA, the staff reviewed the risk and safety insights contained in the Dresden SPRA submittal. The staff also used the screening criteria described in the August 29, 2017 (ADAMS Accession No. ML17146A200), staff memorandum titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic," as part of its review and recommendation to the SMRP. The criteria in the staff's guidance document include thresholds to assist in determining whether to apply the backfit screening process described in Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," dated October 9, 2013 (ADAMS Accession No. ML12059A460), to the SPRA submittal review. The Dresden SPRA submittal demonstrated that the plant SLERF was not below the initial screening value in the August 29, 2017, staff memorandum. As a result, the NRC staff utilized the Dresden SPRA submittal and other available information in conjunction with the guidance in the August 29, 2017, memorandum to complete a detailed screening with respect to SLERF for Dresden. The detailed screening concluded that Dresden should be considered a Group 1 plant because:

- Sufficient reductions in SLERF cannot be achieved by potential modifications considered in this evaluation, to constitute substantial safety improvements based upon importance measures, available information, and engineering judgement;

- Additional consideration of containment performance, as described in NUREG/BR-0058, does not identify a modification that would result in a substantial safety improvement; and

- The staff did not identify any potential modifications that would be appropriate to consider necessary for adequate protection or compliance with existing requirements.

A discussion of the detailed screening evaluation completed by the NRC staff is provided in Enclosure 2 to this letter.

Based on its review of the Dresden SPRA submittal, including the detailed screening evaluation, the technical team determined that recommending Dresden be classified as a Group 1 plant was appropriate and additional review and/or analysis to pursue a plant-specific backfit was not warranted.

As a part of the Phase 2 decisionmaking process for SPRAs, the NRC formed the Technical Review Board (TRB), a board of senior-level NRC subject matter experts, to ensure consistency of review across the spectrum of plants that would be providing SPRA submittals. The technical review team provided the results of the Dresden review to the TRB with the Phase 2 recommendation that Dresden be categorized as a Group 1 plant. The TRB members assessed the information presented by the technical team and agreed with the team's recommendation for classification of Dresden as a Group 1 plant.

Subsequently, the technical review team consulted the SMRP and provided the results of the review including the recommendation for Dresden to be categorized as a Group 1 plant. The SMRP members asked questions about the review, as well as the risk insights and provided input to the technical review team. The SMRP approved the staff's recommendation that Dresden should be classified as a Group 1 plant.

AUDIT REPORT

The July 6, 2017, generic audit plan describes the NRC staff's intention to issue an audit report that summarizes and documents the NRC's regulatory audit of licensee's SPRA submittals associated with their reevaluated seismic hazard information. The NRC staff's audit for Dresden's SPRA response to the 50.54(f) letter included a review of licensee documents through an electronic reading room. An audit summary document is included as Enclosure 3 to this letter.

CONCLUSION

Based on the staff's review of the Dresden submittal against the endorsed SPID guidance, the NRC staff concludes that the licensee responded appropriately to Enclosure 1, Item (8) of the 50.54(f) letter. Additionally, the staff's review concluded that the SPRA is of sufficient technical adequacy to support Phase 2 regulatory decisionmaking in accordance with the intent of the 50.54(f) letter. Based on the results and risk insights of the SPRA submittal, the NRC staff also concludes that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required.

Application of this review is limited to the review of the 10 CFR 50.54(f) response associated with NTTF Recommendation 2.1 "Seismic" review. The staff notes that assessment of the SPRA for use in other licensing applications would warrant review of the SPRA for its intended application. The NRC may use insights from this SPRA assessment in its regulatory activities, as appropriate.

If you have any questions, please contact Milton Valentin at (301) 415-2864 or via e-mail at Milton.Valentin@nrc.gov.

Sincerely,

/RA/

Mohamed K. Shams, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosures:
1. NRC Staff SPRA Submittal Technical Review Checklist
2. NRC Staff SPRA Submittal Detailed Screening Evaluation
3. NRC Staff Audit Summary

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NRC Staff SPRA Submittal Technical Review Checklist

Several nuclear power plant licensees are performing seismic probabilistic risk assessments (SPRAs) as part of their required submittals to satisfy Near-Term Task Force (NTTF) Recommendation 2.1: Seismic. These submittals are prepared according to the guidance in the Electric Power Research Institute – Nuclear Energy Institute (EPRI-NEI) Screening, Prioritization, and Implementation Details (SPID) document (EPRI-SPID, 2012), which was endorsed by the staff for this purpose (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12319A074). The SPRA peer reviews are also expected to follow the guidance in NEI 12-13 (NEI, 2012).

The SPID indicates that an SPRA submitted to satisfy NTTF Recommendation 2.1: Seismic must meet the requirements in the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS), Probabilistic Risk Assessment (PRA) Methodology Standard (hereafter the ASME/ANS Standard). Either the “Addendum A version” (ASME/ANS Addendum A, 2009) or the “Addendum B Version” (ASME/ANS Addendum B, 2013) of the ASME/ANS Standard can be used.

Tables 6-4, 6-5, and 6-6 of the SPID also provide a comparison of each of the Supporting Requirements (SRs) of the ASME/ANS Standard to the relevant guidance in the SPID. For most SRs, the SPID guidance does not differ from the requirement in the ASME/ANS Standard. However, because the guidance of the SPID and the criteria of the ASME/ANS Standard differ in some areas, or the SPID does not explicitly address an SR, the staff developed this checklist, in part, to help staff members to address and evaluate the differences.

In general, the SPID allowed departures or differed from the ASME/ANS Standard in the following ways:

(i) In some technical areas, the SPID’s requirements tell the SPRA analyst “how to perform” one aspect of the SPRA analysis, whereas the ASME/ANS Standard’s requirements generally cover “what to do” rather than “how to do it.”

(ii) For some technical areas and issues, the requirements in the SPID differ from those in the ASME/ANS Standard.

(iii) The SPID has some requirements that are not in the ASME/ANS Standard.

The technical positions in the SPID have been endorsed by the U.S. Nuclear Regulatory Commission (NRC) staff, subject to certain conditions concerning peer review outlined in the staff’s November 16, 2012, letter to NEI (ADAMS Accession No. ML12286A029).

The following checklist is comprised of the 16 “Topics” that require additional staff guidance because the SPID contains specific guidance that differs from the ASME/ANS Standard or expands on it. Each is covered below under its own heading, “Topic 1,” “2,” etc. The checklist was discussed during a public meeting held on December 7, 2016 (ADAMS Accession No. ML16350A181).

- Topic 1: Seismic Hazard (SPID Sections 2.1, 2.2, and 2.3)
- Topic 2: Site Seismic Response (SPID Section 2.4)
- Topic 3: Definition of the Control Point for the Safe Shutdown Earthquake (SSE) to Ground Motion Response Spectrum (GMRS) Comparison Aspect of the Site Analysis (SPID Section 2.4.2)
- Topic 4: Adequacy of the Structural Model (SPID Section 6.3.1)
- Topic 5: Use of Fixed-Based Dynamic Seismic Analysis of Structures for Sites Previously Defined as “Rock” (SPID Section 6.3.3)
- Topic 6: Use of Seismic Response Scaling (SPID Section 6.3.2)
- Topic 7: Use of New Response Analysis for Building Response, In-Structure Response Spectra (ISRS), and Fragilities
- Topic 8: Screening by Capacity to Select Structures, Systems, and Components (SSCs) for Seismic Fragility Analysis (SPID Section 6.4.3)
- Topic 9: Use of the Conservative Deterministic Failure Margin (CDFM)/Hybrid Methodology for Fragility Analysis (SPID Section 6.4.1)
- Topic 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)
- Topic 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)
- Topic 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables Methodology (SPID Section 6.4.1)
- Topic 13: Evaluation of Seismic Large Early Release Frequency (SLERF) (SPID Section 6.5.1)
- Topic 14: Peer Review of the SPRA, Accounting for NEI 12-13 (SPID Section 6.7)
- Topic 15: Documentation of the SPRA (SPID Section 6.8)
- Topic 16: Review of Plant Modifications and Licensee Actions

TOPIC 1: Seismic Hazard (SPID Sections 2.1, 2.2, and 2.3)

The site under review has updated/revised its Probabilistic Seismic Hazard Analysis (SHA) from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.	No
Notes from staff reviewer: None. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the Probabilistic SHA requirements in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The guidance in the SPID was followed for developing the probabilistic seismic hazard for the site.	Yes
• An alternate approach was used and is acceptable on a justified basis.	N/A

TOPIC 2: Site Seismic Response (SPID Section 2.4)

The site under review has updated/revised its site response analysis from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.	Yes
Notes from staff reviewer: The licensee updated its site response analysis to calculate foundation input response spectra (FIRS) for plant structures for the analysis used in the SPRA submittal (ADAMS Accession No. ML19304B567). This update uses the same inputs as was used in the reevaluated seismic hazard report submitted March 31, 2014 (ADAMS Accession No. ML14091A012), and results in no change to the seismic hazard assessment at the facility. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs SHA-E1 and E2 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's development of PSHA inputs and base rock hazard curves meets the intent of the SPID guidance or another acceptable approach.	Yes
• The licensee's development of a site profile for use in the analysis adequately meets the intent of the SPID guidance or another acceptable approach.	Yes
• Although the licensee's development of a shear velocity (V_s) profile for use in the analysis does not meet the intent of the SPID guidance, it is acceptable on another justified basis.	N/A

TOPIC 3: Definition of the Control Point for the SSE to GMRS Comparison Aspect of the Site Analysis (SPID Section 2.4.2)

The issue is establishing the control point where the SSE is defined. Most sites have only one SSE, but some sites have more than one SSE, for example one at rock and one at the top of the soil layer. This control point is needed because it is used as part of the input information for the development of the seismic site-response analysis, which in turn is an important input for analyzing seismic fragilities in the SPRA. The SPID (Section 2.4.1) recommends one of two criteria for establishing the control point for a logical SSE-to-GMRS comparison:	
A) If the SSE control point(s) is defined in the final safety analysis report (FSAR), it should be used as defined.	No
B) If the SSE control point is not defined in the FSAR, one of three criteria in the SPID (Section 2.4.1) should be used.	Yes
C) An alternative method has been used for this site.	N/A
The control point used as input for the SPRA is identical to the control point used to establish the GMRS.	Yes
If <u>yes</u>, the control point can be used in the SPRA and the NRC staff's earlier acceptance governs.	
If <u>no</u>, the NRC staff's previous reviews might not apply. The staff's review of the control point used in the SPRA is acceptable.	N/A
Notes from staff reviewer: None. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this topic.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's definition of the control point for site response analysis adequately meets the intent of the SPID guidance.	Yes
• The licensee's definition of the control point for site response analysis does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 4: Adequacy of the Structural Model (SPID Section 6.3.1)

The NRC staff review of the structural model finds an acceptable demonstration of its adequacy.	Yes
Used an existing structural model	No
Used an enhancement of an existing model	No
Used an entirely new model	Yes
Criteria 1 through 7 (SPID Section 6.3.1) are all met.	Yes

Notes from staff reviewer:

Section 4.3 of the SPRA submittal, describes the analysis of structures which support the safety-related components and systems. Table 4.3-1 of the submittal provides a summary of the foundation condition, structural modeling and the analysis methods used for the Reactor Building-Turbine Building (RB-TB) complex, Station Blackout Building (SBO) and Crib House (CB). The RB-TB complex consists of the Reactor and Turbine Buildings for Units 2 and 3, the Main Control Room, and the HPCI Building.

New 3D finite element models were developed for the RB-TB complex, SBO and CB. The licensee stated in the submittal that the TBs are structurally connected to each other and to the RBs, and a coupled structural model was developed for the RB-TB complex. Using the audit review process, the staff learned that the reactor pressure vessel and the internals in the RBs were modeled using a lumped mass stick model.

Soil-structure interaction (SSI) analysis was performed for the RB-TB and SBO buildings, while fixed-base analysis was used for the CB. The structural response from these models was used to develop the ISRS at specific locations. The ISRS was used for fragility evaluations of mechanical and electrical components housed in these structures.

The NRC staff used the audit process to assess the structural modeling and response analyses. The review of the supporting documents confirms that the 3D-finite element structural modeling captures structural responses that include torsional effects resulting from eccentricities, out-of-plane floor response, and in-plane floor diaphragm stiffness. The cut-off frequency was found in the range of 20 to 50 Hertz (Hz). The licensee stated in its submittal that the SSI analyses were performed in three-spatial directions simultaneously and the seismic hazard level for the dynamic analyses were selected consistent with the risk contribution from the quantification analysis. The NRC staff audit review indicates that appropriate modes of vibration of the structures were considered in the analysis and the modeling approach used is consistent with the requirements of ASCE/SEI 4-16. Thus, NRC staff finds that SPID (Section 6.3.1) Criteria 1 through 7 were met and that Dresden used realistic mathematical models to represent the three-dimensional dynamic characteristics of the building structures for seismic response calculations is in accordance with ASME/ANS Code Case SFR-C1, C5 and C6 requirements.

In addition to the buildings listed in Table 4-3-1, the Dresden SPRA report also identified the Isolation Condenser Pump House, Ventilation Chimneys, and Dresden Island Lock and Dam in the Seismic Equipment List (SEL). The licensee developed fragilities for these structures.

The NRC audit confirmed that the licensee addressed potential effects of soil-liquefaction, lateral spreading, and settlement at the site and precluded secondary hazards arising from these soil failures because major structures are either found on rock or on soil which is not susceptible to liquefaction.

In response to F&O 19-13 and 24-1 (SFR-C1), the licensee stated that three times the GMRS was used as the reference earthquake for the structural response analysis and ISRS for fragility evaluations of all components in the RB-TB complex because most components housed in the complex are top risk contributors. Seismic ground motion at hazard exceedance 10^{-5}/yr was used for SBO and Crib House. The SBO building contains relatively few risk significant components and these components fail at lower ground motions than three times GMRS. The Crib House houses limited SSCs (vertical and horizontal pumps), which are not risk significant. In response to F&O 21-1 (SFR-C1), the licensee stated that V/H ratios consistent with three times the GMRS and 10^{-5}/yr hazard exceedance level were developed and the response analysis for RB-TB complex and SBO were updated.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs Seismic Fragility Analysis (SFR)-C1 through C6 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's structural model meets the intent of the SPID guidance.	Yes
• The licensee's structural model does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 5: Use of Fixed-Based Dynamic Seismic Analysis of Structures for Sites Previously Defined as "Rock" (SPID Section 6.3.3)

Fixed-based dynamic seismic analysis of structures was used, for sites previously defined as "rock." If no, this issue is moot. If yes, on which structure(s)? Structure #1: Crib House	Yes
Structures #1, #2, and #3: If used, is V_S > about 5000 feet /second (ft./sec.)?	Yes
If 3500 ft./sec. < V_S < 5000 ft./sec., was peak-broadening or peak shifting used?	N/A
Potential Staff Finding: The demonstration of the appropriateness of using this approach is adequate.	Yes
Notes from staff reviewer: Section 4.3.1 of the submittal states that Unit 2/3 Crib House is founded on hard rock. A review of the supporting documents, during the audit, shows that the shear wave velocity at the bottom of the foundation is approximately 8,000 ft/sec. The licensee performed fixed-base analysis of the 3D finite element model of the structure to develop ISRS for fragility evaluation of the SSCs in the Crib House. Fixed-base analysis was performed on the Unit 1[1] and Unit 2/3 Chimneys to determine if failure of these structures would impact adjacent structures. During the audit the staff determined that this approach is consistent with standard practice. Fixed-base modal analyses were also performed for the RB-TB complex and SBO as a verification step in the development of SSI models. The SPRA submittal stated in Section 4.3.1 that SSI results for these buildings were used to evaluate building responses, ISRS, or fragility assessments. No F&Os related to fixed-base analysis. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	

[1] Remaining structures of the decommissioned Dresden Unit 1 were considered by the licensee in the SPRA. Additional information is available at https://www.nrc.gov/info-finder/decommissioning/power-reactor/dresden-nuclear-power-station-unit-1.html

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this topic.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" adequately meets the intent of the SPID guidance.	Yes
• The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 6: Use of Seismic Response Scaling (SPID Section 6.3.2)

Seismic response scaling was used. If <u>yes</u>, on which structure(s)?	No
<u>Potential Staff Findings</u>: If a new UHS/RLE is used, the shape is approximately like the spectral shape previously used for ISRS generation. If the shape is not similar, the justification for seismic response scaling is adequate. Consideration of non-linear effects is adequate.	 N/A N/A N/A
Notes from staff reviewer: The unit 2/3 Crib House and SBO analyses used the ground motion at 10^{-5}/yr hazard exceedance level, which is considered the hazard range of interest (HROI). The submittal states that the time histories were developed matching GMRS hazard level. To evaluate time history at HROI (i.e., 1E-05 hazard level), the licensee scaled the time history at GMRS level to the 1E-05 level. The scaled time histories were used in the structural response analysis. The licensee stated that the spectral shape of the GMRS and 1E-05 was similar, hence, scaling to develop a hazard compatible time history was justified. The NRC staff notes, based on review of supporting documents during audit, that for this submittal the seismic response scaling, as described in SPID Section 6.3.2, was not used to develop the ISRS. No F&Os related to SFR-C3 Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SR SFR-C3 in the ASME/ANS Standard, as well as to the requirements in the SPID. • Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis. • The licensee's use of seismic response scaling adequately meets the intent of the SPID guidance.	 N/A N/A N/A

• The licensee's use of seismic response scaling does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 7: Use of New Response Analysis for Building Response, ISRS, and Fragilities

The SPID does not provide specific guidance on performing new response analysis for use in developing ISRS and fragilities. The new response analysis is generally conducted when the criteria for use of existing models are not met or more realistic estimates are deemed necessary. The requirements for new analysis are included in the ASME/ANS Standard. See SRs SFR-C2, C4, C5, and C6. One of the key areas of review is consistency between the hazard and response analyses. Specifically, this means that there must be consistency among the ground motion equations, the SSI analysis (for soil sites), the analysis of how the seismic energy enters the base level of a given building, and the in-structure-response-spectrum analysis. Said another way, an acceptable SPRA must use these analysis pieces together in a consistent way. The following are high-level key elements that should have been considered:	
1. The FIRS site response was developed with appropriate building specific soil velocity profiles for the following structures: Structure #1 name: RB-TB Complex Structure #2 name: SBO Building Structure #3 name: Crib House Are all structures appropriately considered?	Yes
2. Are models adequate to provide realistic structural loads and response spectra for use in the SPRA? • Is the SSI analysis capable of capturing uncertainties and realistic? • Is the probabilistic response analysis capable of providing the full distribution of the responses?	Yes Yes No
Notes from staff reviewer: Section 4.3 of the SPRA submittal describes the structural response analysis performed for different buildings. The SSI analysis was used to develop the ISRS for the RB-TB complex and SBO, while fixed-base analysis was performed for the Crib House. For SSI analysis, the licensee used a deterministic response analysis accounting for variabilities in strain-compatible soil profiles, structural characteristics, and the earthquake acceleration time histories. Soil profiles and structural characteristics were based on Best Estimate (BE), Lower Bound (LB) and Upper Bound (UB) properties and five SSI cases were selected for structural analysis. Variability in ground motion is considered using five independent sets of time histories spectrally matched to the floor input response spectra. Uncertainties from the SSI	

analysis were addressed by the five SSI cases, each paired with five-time histories generating 25 sets of ISRS. The deterministic structural response was used to calculate median (50%) and conservative (84%) ISRS.

The SSI analysis used an industry standard software for model development and analysis, where the structural systems were modeled using 3D finite element models and the soil as horizontal layers. The deterministic SSI analysis had a cut-off frequency of 50 Hz.

The RB-TB Complex consists of multiple buildings constructed on a common foundation. Also, ground motion incoherency was considered in the SSI analysis. Analysis was performed for the GMRS hazard level and three times the GRMS, which is based on the insights gained from risk quantifications. Baseline uncracked concrete was considered in the structural analysis for the GMRS level, while structural analysis with three times the GMRS considered cracked concrete where appropriate. In response to F&O 19-13 and 24-1 (SFR-C1), the licensee stated that the reference earthquake for evaluating fragility of all components in RB-TB complex is based on three times the GMRS seismic input and anchored to the corresponding peak ground acceleration (PGA).

The SBO building analysis was performed for the 10^{-5} 1/yr hazard level. Hazard consistent strain-compatible soil properties were used. The SSI analysis used an industry standard software for model development and analysis. The deterministic SSI analysis had a cut-off frequency of 50 Hz. The deterministic structural response was used to calculate median (50%) and conservative (84%) ISRS.

The Unit 2/3 Crib House, which is founded on hard rock, is a reinforced concrete structure with a steel frame on top. Three-dimensional fixed-base analysis was performed on the Crib House with input ground acceleration defined at a 10^{-5} 1/yr seismic hazard exceedance level. The deterministic structural response was used to calculate the ISRS and building displacements.

Based on staff review during an audit, the dynamic analyses of the structure included a response spectrum analysis to identify regions in the structure for potential concrete cracking and a time history analysis with the modified concrete properties at those regions. The ISRS developed from this analysis was used for the fragility evaluation of the SSCs in the Crib House. The licensee stated in the submittal that one time-history was used for developing the ISRS because the limited components housed in the Crib House (vertical and horizontal pumps) are high capacity and low risk significant.

Based on the NRC review of information in the submittal and auditing of structural response documents in the e-Portal, the staff finds the deterministic approach to evaluate structural response and floor response spectra to be appropriate. The deterministic approach, consideration of variability in soil and structural properties are consistent with ASCE/SEI 4-16 recommendation and industry practice.

There are no F&Os related to SFR-C2, C4, C5, and C6

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs SFR-C2, C4, C5, and C6 in the ASME/ANS Standard, as well as to the requirements in the SPID.	N/A
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's FIRS modeling is consistent with the prior NRC review of the GMRS and soil velocity information.	Yes
• The licensee's structural model meets the intent of the SPID guidance and the ASME/ANS Standard's requirements.	Yes
• The response analysis accounts for uncertainties in accordance with the SPID guidance and the ASME/ANS Standard's requirements.	Yes
• The NRC staff concludes that an acceptable consistency has been achieved among the various analysis pieces of the overall analysis of site response and structural response.	Yes
• The licensee's structural model does not meet the intent of the SPID guidance and the ASME/ANS Standard's requirements but is acceptable on another justified basis.	N/A

TOPIC 8: Screening by Capacity to Select SSCs for Seismic Fragility Analysis (SPID Section 6.4.3)

The selection of SSCs for seismic fragility analysis used a screening approach by capacity following Section 6.4.3 of the SPID. If no, see items D and E. If yes, see items A, B, and C.	Yes
Potential Staff Findings: A) The recommendations in Section 6.4.3 of the SPID were followed for the screening aspect of the analysis, using the screening criteria therein.	Yes
B) The approach for retaining certain SSCs in the model with a screening-level seismic capacity follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	Yes
C) The approach for screening out certain SSCs from the model based on their inherent seismic ruggedness follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	Yes
D) The ASME/ANS Standard has been followed.	Yes
E) An alternative method has been used and its use has been appropriately justified.	N/A
Notes from staff reviewer: Section 4.4.1 of the SPRA submittal states that a multi-step and iterative screening process was used to reduce the SEL to the point where only significant SSCs were included in the SPRA. Some components were screened out of the SPRA because, as stated in the submittal, these components do not contribute to plant safety or shutdown thus, they are considered non-contributors to the overall response of the SPRA. The licensee identified passive components that are inherently seismically rugged and screened those out from the SPRA. During walkdown and walk-by process components were identified that are low-likelihood and non-risk significant. A High Confidence Low Probability of Failure (HCLPF) capacity of 2g was assigned to these components. In F&O 19-5 (SFR-B1), the peer review points out insufficient basis was provided for screened-out SSCs based on judgement. In response, the licensee asserted that the following was accomplished: (i) documentation of technical justification for rugged components; (ii) development of fragility for some components that were initially judged	

rugged; (iii) development of bounding calculations for less rugged SSCs; and (iv) development of fragilities for FLEX equipment. In response to F&O 24-6 (SFR-F1), the licensee performed bounding calculations to provide justification for assigning a HCLPF capacity of 2.0g to component groups (e.g., pumps, fans, air handlers). Using the audit process, staff confirmed that the HCLPF capacity calculated for these components exceeded the assigned value. Based on the SPRA submittal, supplemented by the audit review, the staff concludes that the capacity-based screening criterion recommended in SPID Section 6.4.3 was properly followed for the SPRA; the licensee's disposition is acceptable; and the screening of high-seismic capacity of components was addressed in accordance with ASME/ANS Standard SFR-B1 requirements. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SR SFR-B1 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of a screening approach for selecting SSCs for fragility analysis meets the intent of the SPID guidance.	Yes
• The licensee's use of a screening approach for selecting SSCs for fragility analysis does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 9: Use of the CDFM/Hybrid Methodology for Fragility Analysis (SPID Section 6.4.1)

The CDFM/Hybrid method was used for seismic fragility analysis. If no, See item C) below and next issue. If yes:	Yes
Potential Staff Findings:	
A) The recommendations in Section 6.4.1 of the SPID were followed appropriately for developing the CDFM HCLPF capacities.	Yes
B) The Hybrid methodology in Section 6.4.1 and Table 6-2 of the SPID was used appropriately for developing the full seismic fragility curves.	Yes
C) An alternative method has been used appropriately for developing full seismic fragility curves.	N/A

Notes from staff reviewer:

Section 4.4.2 of the SPRA submittal states that a three-step quantification process was used for fragility evaluation of SEL components starting with site-specific representative fragility and improving the fragilities for top risk contributors to Seismic Core Damage Frequency (SCDF) and SLERF in second and third quantifications.

During the second and third risk quantifications, component-specific fragility for some of the risk-significant items was refined using the CDFM/Hybrid methodology. Fragilities were estimated based on the HCLPF using the CDFM approach and considering the aleatory variability and epistemic uncertainty given in SPID Table 6-2. For the third quantification, fragility estimate for risk-significant SSCs, the licensee also used a hybrid separation of variable (SOV) method, where the composite logarithmic standard deviation was estimated using HCLPF and the SSC median capacity. In addition, the licensee stated in the submittal that the first quantification of fragility was also based on estimating HCLPF with generic uncertainty values from the SPID.

During the staff audit, fragility reviews of selected components confirmed that the CDFM/Hybrid methodology followed guidance in technical reports EPRI NP-6041and other EPRI guidance and accounted for site-specific data including ISRS developed from the structural response analysis. Failure modes considered included structural failure, functional failures, seismic interaction with surrounding equipment or objects, and relay chatter. The NRC staff concludes Dresden approach is consistent with the recommendations in Section 6.4.1 and Table 6-2 of the SPID.

In response to F&Os (19-14, 22-3, 22-4, 22-5, 24-5) related to HCLPF, the licensee updated the fragility calculations for piping supports for Loss of Coolant Accidents, Motor Control Center, and Crib House; considered horizontal directional peak response in the fragility evaluation; and assessed the validity of using generic variability and uncertainty values.

The NRC staff reviewed the disposition of F&O 24-4 (SFR-D1) in supporting documents, which addressed potential failure modes considering both structural and foundation failures for the Dresden Lock and Dam complex. The licensee used a fixed-base finite element model to calculate out-of-plane bending moment demands. Dresden Lock and Dam fragility was evaluated using the CDFM/Hybrid approach at a reference earthquake three times the GMRS consistent with the hazard level in the SPRA model.

In response to F&O 24-3, the licensee evaluated the fragility for ventilation chimneys for Unit 1 and Units 2/3, which were not originally included in the SPRA model. The concern of the F&O was that failure of chimneys could impact nearby SSCs. Based on the review of supporting documents, the NRC staff finds the fragility of the chimneys was developed by estimating HCLPF capacity using the CDFM method and the failure of chimneys was included in the SPRA model. The licensee determined that failure of chimney for Unit 1 would not impact SEL components; however, the chimney for Units 2/3 could impact the Crib House, Turbine Building Heater Bay, and above ground service water pipe. The licensee calculated the probability of these SSCs being impacted by the chimney failure. Chimneys are not included in SPRA submittal Tables 5.4-2, 5.4-3, 5.5-2 and 5.5-3, which summarize Fussell-Vesely (F-V) importance, thus the F-V importance measure for chimneys is likely to be less than 0.005.

Based on the SPRA submittal, supplemented by the audit review, the staff concludes that the licensee's disposition of F&Os are acceptable.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this Topic.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis meets the intent of the SPID guidance.	Yes
• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)

The SPID requires that certain SSCs that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility using a methodology described in Section 6.4.2 of the SPID.	Yes
Potential Staff Findings:	
The NRC staff review of the SPRA's fragility analysis of SSCs sensitive to high frequency seismic motion finds that the analysis is acceptable.	Yes
The flow chart in Figure 6-7 of the SPID was followed.	Yes
The flow chart was not followed but the analysis is acceptable on another justified basis.	N/A

Notes from staff reviewer:

Sensitivity of SSCs to high frequencies is addressed in Sections 4.1.2 and 4.4.2 of the SPRA submittal. The NRC staff review of the submittal and supporting documents showed that high-frequency sensitive devices identified in EPRI 3002004396 were included in the SPRA and the high-capacity components; e.g., relays and similar devices, were screened out consistent with Figure 6-7 of the SPID. Some medium and low capacity components were also screened out based on the circuit analysis. The SSCs sensitive to high frequencies that are included in the SPRA were evaluated based on a capacity versus demand, consistent with the recommendation in Section 6.4.2 of the SPID. In the three-step fragility quantification process discussed in Section 4.4.2, the relay fragilities were refined based on risk contribution. As shown in the risk quantification Tables 5.4-2 to 3 and Tables 5.5-2 to 3, several relay chatter fragilities were evaluated using CDFM analysis.

The NRC staff reviewed the licensee's application of the CDFM approach for functional failure of selected relays located in the RB-TB complex. The relay fragility methodology, used during the second quantification, is based on EPRI 6041 and other EPRI guidance and 84th percentile Non-exceedance Probability ISRS. The staff verified that the fragility analysis incorporates resolution of F&O 21-1, 24-1, 21-7, and 22-3, which concerned the use of ISRS evaluated based on three times the GMRS seismic ground motion, documenting verification of generic equipment ruggedness spectra (GERS) caveat, and application of horizontal direction peak response.

Using the audit process, the NRC staff confirmed that recommendations in SPID Section 6.4.2 were followed in the SPRA, and the high-seismic capacity of components were addressed in accordance with ASME/ANS Standard SFR-F3 requirements.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): None.	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SR SFR-F3 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's fragility analysis of SSCs sensitive to high frequency seismic motion meets the intent of the SPID guidance.	Yes
• The licensee's fragility analysis of SSCs sensitive to high-frequency motion does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)

The SPID requires that certain relays and related devices (generically, "relays") that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility. Although following the ASME/ANS Standard is generally acceptable for the fragility analysis of these components, the SPID (Section 6.4.2) contains additional guidance when either circuit analysis or operator-action analysis is used as part of the SPRA to understand a given relay's role in plant safety. When one or both are used, the NRC reviewer should use the following elements of the checklist.	
i) Circuit analysis: The seismic relay-chatter analysis of some relays relies on circuit analysis to assure that safety is maintained. (A) If no, then (B) is moot. (B) If yes: Potential Staff Finding: The approach to circuit analysis for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
ii) Operator actions: The relay-chatter analysis of some relays relies on operator actions to assure that safety is maintained. (A) If no, then (B) is moot. (B) If yes: Potential Staff Finding: The approach to analyzing operator actions for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
Notes from staff reviewer: The licensee discussed relay chatter evaluation performed in Section 4.1.2 of the submittal. The evaluation involves relays and other potentially chatter sensitive devices including circuit breakers and motor starters. During the audit process of supporting documentation, the NRC staff confirmed that the circuit analysis was part of the chatter analysis evaluation. The licensee stated that circuit analysis was performed in	

accordance with requirements of ASME/ANS SPRA standard. The circuit analysis resulted in many relay chatter scenarios screened out from further evaluation based on no impact to component function. For the chatter analysis, relays were assumed to malfunction only during the strong motion portion of the seismic event. The relays that were not screened out, were considered in the SPRA model for further evaluation and summarized in Table 4.1.2-1 of the SPRA. The SPRA also relied on operator actions where such actions are feasible. The quantification of operator error probabilities is discussed in Section 5.1 of the SPRA. In response to F&O 25-3 (SPR-B4), the licensee stated that the fragility of relays impacting the low-pressure coolant injection (LPCI) valves during LOCA events was included in the SPRA model. These relays were previously screened out. Based on the analysis, the licensee stated the relays were non-risk significant. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs Seismic Plant Response Analysis (SPR)-B6 (Addendum A) or SPR-B4 (Addendum B) in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's analysis of seismic relay-chatter effects meets the intent of the SPID guidance.	Yes
• The licensee's analysis of seismic relay-chatter effects does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables Methodology (SPID Section 6.4.1)

The CDFM methodology has been used in the SPRA for analysis of the bulk of the SSCs requiring seismic fragility analysis.	Yes
If no, the staff review will concentrate on how the fragility analysis was performed, to support one or the other of the "potential staff findings" noted just below.	N/A
If yes, significant risk contributors for which use of SOV fragility calculations would make a significant difference in the SPRA results have been selected for SOV calculations.	Yes
Potential Staff Findings: A) The recommendations in Section 6.4.1 of the SPID were followed concerning the selection of the "dominant risk contributors" that require additional seismic fragility analysis using the SOV methodology.	Yes
B) The recommendations in Section 6.4.1 were not followed, but the analysis is acceptable on another justified basis.	N/A
Notes from staff reviewer: Section 4.4.2 of the SPRA submittal details the process used in determining which SSC fragilities would be determined using CDFM or SOV. Tables 5.4-2, 5.4-3, 5.5-2, and 5.5-3 provide a listing of each unit's SCDF and SLERF risk significant SSCs (F-V values greater than 0.005). While the fragilities for the majority of the SSCs reported in these tables were determined using the CDFM method, several of the SSCs having the highest risk significance as determined by F-Vs were calculated using the SOV method. While the percentage of the risk significant SSC fragilities determined using the SOV method was small relative to those determined using the CDFM method, given the SCDF risk values are less than 1E-05 and the SLERF risk is significantly dominated by offsite power (by F-V), the impact of not performing SOV fragilities on the remaining SSCs is determined to not change the insights of this submittal. Accordingly, the NRC staff concluded that the licensee's approach was to achieve more detailed fragility analyses for dominant risk contributors using the SOV approach or a more refined CDFM approach is reasonable. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this Topic.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the SOV methodology meets the intent of the SPID guidance.	Yes
• The licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the SOV methodology does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 13: Evaluation of SLERF (SPID Section 6.5.1)

<table>
<tr><td>The NRC staff review of the SPRA's analysis of SLERF finds an acceptable demonstration of its adequacy.

<u>Potential Staff Findings</u>:
A) The analysis follows each of the elements of guidance for SLERF analysis in Section 6.5.1 of the SPID, including in Table 6-3.

B) The SLERF analysis does not follow the guidance in Table 6-3 but the analysis is acceptable on another justified basis.</td><td>Yes

Yes

N/A</td></tr>
<tr><td colspan="2">Notes from staff reviewer:
Section 4.1.1 of the SPRA submittal discusses the SEL selection basis, which includes those associated SSCs with preventing or mitigating radioactive release if a core damage event occurs. Specifically, Table 4.1.1-1 describes systems associated with containment pressure and temperature control, vapor suppression, and containment isolation. With regards to SPRA modeling, Section 5.1 describes how SLERF accident sequences are identical to the full power internal-events (FPIE) PRA methodology, and that no additional unique LERF accident scenarios were found. During the NRC audit, it was determined that all but two FPIE F&Os were resolved and closed out using the NEI F&O Closure process. One of them, FPIE F&O 4-3 regarding SR SY-A4 to perform walkdowns to confirm the system analysis correctly reflects the as-built, as-operated plant remains OPEN. The peer review team stated that a possible resolution is to perform fresh walkdowns and interviews. The licensee's disposition for this F&O is that walkdowns performed for internal flooding, fire, and seismic should provide reasonable assurance that the SPRA model reflects the as-built, as-operated plant. Given the margin to the submittal's SCDF and SLERF risk thresholds, the NRC staff concludes that this issue would not impact the submittal results. The second one, FPIE F&O 6-19 regarding SY-B1 for not having common cause failures (CCFs) for the control rod drive (CRD) pumps, the licensee's disposition states that the CRD CCFs are now included in the SPRA model. Therefore, this F&O does not impact the results of this submittal.

Deviation(s) or deficiency(ies) and Resolution: None

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes that:
The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to SRs SFR-F4, SPR-E1, SPR-E2, and SPR-E6 (Addendum B only) in the ASME/ANS Standard, as well as to the requirements in the SPID.</td><td>Yes</td></tr>
</table>

• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's analysis of SLERF meets the intent of the SPID guidance.	Yes
• The licensee's analysis of SLERF does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 14: Peer Review of the SPRA, Accounting for NEI 12-13 (SPID Section 6.7)

The NRC staff review of the SPRA's peer review findings, observations, and their resolution finds an acceptable demonstration of the peer review's adequacy.	Yes
Potential Staff Findings: A) The analysis follows each of the elements of the peer review guidance in Section 6.7 of the SPID.	Yes
B) The composition of the peer review team meets the SPID guidance.	Yes
C) The peer reviewers focusing on seismic response and fragility analysis have successfully completed the Seismic Qualifications Utility Group training course or equivalent (see SPID Section 6.7).	Yes
In what follows, a distinction is made between an "in-process" peer review and an "end-of-process" peer review of the completed SPRA submittal. If an in-process peer review is used, go to (D) and then skip (E). If an end-of-process peer review is used, skip (D) and go to (E).	
D) The "in-process" peer review process followed the guidance in the SPID (Section 6.7), including the three "bullets" and the guidance related to NRC's additional input in the paragraph immediately following those three bullets. These three bullets are: • The SPRA findings should be based on a consensus process, and not based on a single peer review team member • A final review by the entire peer review team must occur after the completion of the SPRA project • An "in-process" peer review must assure that peer reviewers remain independent throughout the SPRA development activity.	N/A
If no, go to (F). If yes, the "in-process" peer review approach is acceptable. Go to (G).	

E) The "end-of-process" peer review process followed the peer review guidance in the SPID (Section 6.7). If no, go to (F). If yes, the "end-of-process" peer review approach is acceptable. Go to (G). F) The peer-review process does not follow the guidance in the SPID but is acceptable on another justified basis. G) The licensee peer-review findings were satisfactorily resolved or were determined not to be significant to the SPRA conclusions for this evaluation.	Yes N/A Yes

Notes from staff reviewer:

Section 5.2 and Appendix A of the SPRA submittal provides a discussion of the technical adequacy of the model. The peer review was conducted in January 2019 using the NEI 12-13 process and the ASME/ANS RA-Sb-2013 PRA standard (Addendum B). Addendum B is approved by the NRC-endorsed SPID for SPRA submittals. All elements of the SPRA were peer reviewed against the capability category II (CC-II) requirements of the standard. NEI 12-13 has been approved by the NRC for use in risk-informed licensing applications, subject to certain NRC staff comments and proposed resolutions (NRC, 2018a, 2018b).

The SPRA submittal provides the qualifications for each of the peer review team members and states that the peer reviewers were independent of the SPRA development. Concurrence on the assignment of capability categories to each SR was based on a consensus process involving all members of the review team. The team members focusing on review of the fragility analysis and plant response model, and who participated in the plant walkdown were stated to have SQUG training course or equivalent in the peer review report. The resumes in the submittal and peer review report for each of these peer reviewers, which were reviewed by the NRC staff during the audit, were shown to demonstrate sufficient SPRA experience.

Appendix A of the submittal states the internal events PRA (IEPRA) model-of-record was peer reviewed in November 2016 against the CC-II requirements of the PRA standard (ASME/ANS Addendum B, 2013) and Regulatory Guide (RG) 1.200, Rev. 2. Addendum B of the PRA standard is not approved by RG 1.200, Rev. 2. However, during the NRC audit it was determined that the IEPRA Peer Review report states that the Dresden model was judged against CC-II requirements of the ASME/ANS Addendum A, 2009 report, which is endorsed by RG 1.200, Rev.2 and followed the process outlined in NEI 05-04. During the audit it was determined that the IEPRA F&Os, with the exception of two, were closed in September 2017 Closure Review using the February 21, 2017 NEI proposal guidance. However, it does not indicate if the NRC staff comments (NRC, 2017a, 2017b) were considered. During the audit the licensee confirmed that the F&O Closure process adequately addressed the relative NRC staff comments concerning the September 2017 closeout.

Deviation(s) or deficiency(ies) and Resolution: None

Consequence(s): N/A	
The NRC staff concludes that:	
• The licensee's peer-review process meets the intent of the SPID guidance.	Yes
• The licensee's peer-review process does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 15: Documentation of the SPRA (SPID Section 6.8)

The NRC staff review of the SPRA's documentation as submitted finds an acceptable demonstration of its adequacy.	Yes
The documentation should include all the items of specific information contained in the 50.54(f) letter as described in Section 6.8 of the SPID.	Yes

Notes from staff reviewer:

Tables 2-1 and 2-2 of the submittal provide a cross-reference of information required by 10 CFR 50.54(f) and specified in Section 6.8 of the SPID to the sections of the submittal where the information can be found. The level-of-detail of the information provided is generally consistent with that specified in Section 6.8 of the SPID. The SPID requires that there should be sufficient information to assess the results to all key aspects of the analysis. Sections 5.3.2, 5.6, and A.8 of the submittal identify and discuss key assumptions and sources of uncertainty for the SPRA, with sensitivity analyses on some of these parameters provided in Section 5.7. Sections 5.4 and 5.5 of the submittal provide the SPRA results.

Section 5.6 of the submittal presents the SPRA quantification uncertainty results for SCDF and SLERF (i.e., the median (50%) and the 95th percentiles). The mean from the uncertainty analysis was not provided, but rather the SCDF point estimate of 5.80E-06 per year for both Units 2 and 3 and SLERF point estimates of 2.9E-06 per year and 2.8E-06 per year for Units 2 and 3, respectively, were provided in the submittal. The licensee further clarified that the SCDF and SLERF values for the two units are very similar because Units 2 and 3 are symmetrical. According to the NRC staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200), the NRC staff utilizes the mean SCDF and SLERF to develop a recommendation on whether the plant should move forward as a Group 1, 2, or 3 plant. During the audit, the licensee provided the actual mean SCDF and SLERF from the uncertainty analysis, which are 8.65E-06 per year and 4.32E-06 per year, respectively. In addition, the 95th percentile SCDF and SLERF of 2.76E-05 per year and 1.47E-05 per year, respectively, from the uncertainty analysis were provided in the submittal. Because Units 2 and 3 are nearly identical from a seismic risk perspective, these mean and 95th percentile values were used in the NRC staff's screening evaluation reported in Enclosure 2 for both units.

Tables 5.4-2, 5.4-3, 5.5-2, and 5.5-3 of the SPRA submittal provide the dominant risk contributors to SCDF and SLERF. Most of the risk significant SSCs are not reflected in the top ten cutsets provided for SCDF and SLERF in Tables 5.4-6 and 5.5-6, respectively. During the audit, the licensee provided the top 200 SCDF and SLERF cutsets and identified the cutsets in which the risk-significant SSCs were located. The NRC staff determined that the dominant risk contributors are appropriately reflected in the quantification results.

Section 5.7 of the submittal provides several sensitivity studies that provides insights to the NRC staff when evaluating the SPRA submittal. Regarding sensitivity case 7c related to extending the %G08 interval, the licensee showed its total initiator frequency for the %G08 is different from the base case provided in Table 5.5-1. During the audit,

the licensee clarified the inconsistency was due to the conversion of reactor calendar year (e.g., 8760 hours) to reactor critical year (e.g., 8760 hours times the power availability factor). The licensee provided a supplement to their submittal dated March 6, 2020 (ADAMS Accession No. ML20066K784), to update the appropriate values for the sensitivity case 7c. The NRC staff determined the inconsistency in the initiator frequency for sensitivity case 7c has no impact on the results of the submittal. According to Section 4.1.1 of the SPRA submittal, Diverse and Flexible Coping Strategies (FLEX) is credited in the SPRA to provide emergency AC power, via credit for the FLEX portable diesel generators, and low-pressure injection, via credit for the FLEX portable diesel fire pumps. The NRC memorandum dated May 30, 2017 (ADAMS Accession No. ML17031A269), "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis," provides the NRC staff's assessment of the challenges of incorporating FLEX coping strategies and equipment into a PRA model in support of risk-informed decisionmaking in accordance with the guidance of RG 1.200, Revision 2 (ADAMS Accession No. ML090410014). The licensee provided the results of a sensitivity analysis that removed credit for FLEX. These results show that FLEX equipment and actions have minimal impact on the SPRA results. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes that: • The licensee's documentation meets the intent of the SPID guidance. The documentation requirements in the ASME/ANS Standard can be found in HLR-SHA-J, HLR-SFR-G, and HLR-SPR-F. • The licensee's documentation does not meet the intent of the SPID guidance but is acceptable on another justified basis.	Yes N/A

Topic 16: Review of Plant Modifications and Licensee Actions, If Any

The licensee: • identified modifications necessary to achieve seismic risk improvements. • provided a schedule to implement such modifications (if any), consistent with the intent of the guidance • provided Regulatory Commitment to complete modifications • provided Regulatory Commitment to report completion of modifications.	No N/A N/A N/A
Plant will: • complete modifications by: • report completion of modifications by:	N/A N/A
Notes from the Reviewer: See Enclosure 2 for discussion. Deviation(s) or Deficiency(ies), and Resolution: None Consequences: N/A	
The NRC staff concludes that: • The licensee identified plant modifications necessary to achieve the appropriate risk profile. • The licensee provided a schedule to implement the modifications (if any) with appropriate consideration of plant risk and outage scheduling.	N/A N/A

REFERENCES

ASCE, 2017. "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," ASCE/SEI 4-16, American Society of Civil Engineers, Reston, VA, 2017

ASME/ANS Addendum A, 2009: Standard ASME/ANS RA-Sa-2009, Addenda A to ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2009

ASME/ANS Addendum B, 2013: Standard ASME/ANS RA-Sb-2013, Addenda B to ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2013

EPRI-SPID, 2012: "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," Electric Power Research Institute, EPRI report 1025287, November 2012, (ADAMS Accession No. ML12333A170)

NEI, 2012: NEI 12-13 "External Hazards PRA Peer Review Process Guidelines," Nuclear Energy Institute, August 2012 (ADAMS Accession No. ML12240A027)

NRC, 2017a: Giitter, Joseph, and Ross-Lee, Mary Jane, U.S. Nuclear Regulatory Commission, letter to Krueger, Greg, Nuclear Energy Institute, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," dated May 3, 2017 (ADAMS Accession No. ML17079A427)

NRC, 2017b: Risk Informed Steering Committee, U.S. Nuclear Regulatory Commission, memorandum to Stacey L. Rosenberg, U.S. Nuclear Regulatory Commission, "U.S. Nuclear Regulatory Commission Staff Expectations for an Industry Facts and Observations Independent Assessment Process," dated May 1, 2017 (ADAMS Accession No. ML17121A271)

NRC, 2018a: "U.S. Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13, "External Hazards PRA Peer Review Process Guidelines" (August 2012),"" NRC letter to Nuclear Energy Institute, March 7, 2018 (ADAMS Accession No. ML18025C025)

NRC, 2018b: "U.S. Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13, "External Hazards PRA Peer Review Process Guidelines" (August 2012),"" tabular compilation of NRC staff comments, appended to (NRC, 2018a), (ADAMS Accession No. ML18025C022)

NRC Staff SPRA Submittal Detailed Screening Evaluation

Introduction

The Dresden Nuclear Power Station, Units 2 & 3 (Dresden) Seismic Probabilistic Risk Assessment (SPRA) submittal (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19304B567) indicates that the point estimate seismic core damage frequency (SCDF) is 5.8E-06/reactor-year (/rx-yr) for both Units 2 and 3 and seismic large early release frequency (SLERF) is 2.9E-06/rx-yr for Unit 2 and 2.8E-06/rx-yr for Unit 3. The mean SCDF and SLERF values are not provided in the SPRA report but the 50 percent and 95 percent values were provided. During the audit the licensee provided the mean SCDF of 8.65E-06/rx-yr and SLERF of 4.32E-06/rx-yr, which are used in this evaluation. Because Units 2 and 3 are nearly identical from a seismic risk perspective, these results were used in the NRC staff's screening evaluation for both units. The NRC staff compared these values against the guidance in NRC staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200; hereafter SPRA Screening Guidance), titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic," which establishes a process the NRC staff uses to determine whether substantial safety enhancements are needed, which would support recommending the plant to move forward as a Group 1, 2, 3 plant.[2]

The SPRA Screening Guidance is based on NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," (ADAMS Accession No. ML042820192), NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," (ADAMS Accession No. ML050190193), and NUREG-1409, "Backfitting Guidelines," (ADAMS Accession No. ML032230247), as informed by Nuclear Energy Institute (NEI) 05-01, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document" (ADAMS Accession No. ML060530203). In order to determine the significance of proposed modifications in terms of safety improvement, NUREG/BR-0058 uses screening criteria based on the estimated reduction in core damage frequency, as well as the conditional probability of early containment failure or bypass. Per NUREG/BR-0058, the conditional probability of early containment failure or bypass is a measure of containment performance and the purpose of its inclusion in the screening criteria is to achieve a measure of balance between accident prevention and mitigation. The NUREG/BR-0058 uses a screening criterion of 0.1 or greater for conditional probability of early containment failure or bypass. In the context of the SPRA reviews, the staff guidance uses SCDF and SLERF as the screening criteria where SLERF is directly related to the conditional probability of early containment failure or bypass. Following NUREG/BR-0058, the threshold for the screening criterion in the staff guidance for SLERF is (1.0E-6/rx-yr), or 0.1 times the threshold for the screening criterion for SCDF (1.0E-5/rx-yr).

Since the Dresden SCDF value was below the 1E-05/rx-yr threshold no detailed screening was performed for this hazard risk. The NRC staff found that because the SLERF for Dresden was above the initial screening value of 1.0E-6/rx-yr, a detailed screening following the SPRA Screening Guidance was performed. The detailed screening shows that Dresden should be considered a Group 1 plant because:

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[2] The groups are defined as follows: regulatory action not warranted (termed Group 1), regulatory action should be considered (termed Group 2), and more thorough analysis is needed to determine if regulatory action should be considered (termed Group 3).

- Sufficient reductions in SLERF cannot be achieved by potential modifications considered in this evaluation, to constitute substantial safety improvements based upon importance measures, available information, and engineering judgement;

- Additional consideration of containment performance, as described in NUREG/BR-0058, does not identify a modification that would result in a substantial safety improvement; and

- The staff did not identify any potential modifications that would be appropriate to consider necessary for adequate protection or compliance with existing requirements.

As such, additional refined screening, or further evaluation, was not required.

Detailed Screening

Exelon Generation, LLC (Exelon, the licensee) in performing its seismic analysis in response to the Near-Term Task Force Recommendation 2.1, and the NRC in conducting its review, did not identify concerns that would require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety. In addition, there were no issues identified as non-compliances with the Dresden licenses, or the rules and orders of the Commission. For these reasons, the licensee and the staff did not identify a potential modification necessary for adequate protection or compliance with existing requirements.

The detailed screening uses information provided in the Dresden SPRA report, particularly the importance measures, SLERF, as well as other information described below, to establish threshold and target values that are used to identify areas where potential cost-justified substantial safety improvements might be identified. The detailed screening process makes several simplifying assumptions, similar to a Phase 1 SAMA analysis (NEI 05-01, ADAMS Accession No. ML060530203) used for license renewal applications. The detailed screening process uses risk importance values as defined in NUREG/CR-3385, "Measures of Risk Importance and Their Applications" (ADAMS Accession No. ML071690031). The NUREG/CR-3385 states that the risk reduction worth (RRW) importance value is useful for prioritizing feature improvements that can most reduce the risk. The Dresden SPRA report provides Fussell-Vesely (F-V) importance values, which were converted to RRW values by the NRC staff for this screening evaluation using a standard relationship formulation. Data used to develop the maximum averted cost-risk (MACR) for the severe accident mitigation alternative (SAMA) analysis provided in the *License Renewal Application, Dresden Nuclear Power Station*, dated January 2003 (ADAMS Accession No. ML030090217), was used to calculate the RRW threshold. For this analysis, the NRC staff determined the RRW threshold from the SCDF-based MACR to be 1.16. The MACR calculation includes estimation of offsite exposures and offsite property damage, which captures the impact of SLERF. Therefore, separate SLERF-based MACR calculations were not performed. The target RRWs based on the mean and 95th percentile SLERF were also calculated by the NRC staff to be 1.30 and 1.07, respectively.

Section 5.5 of the Dresden SPRA report included tables listing and describing the structures, systems, and components (SSCs) that are the most significant seismic failure contributors to SLERF. Similar tables were also provided for the most significant contributors due to failure of operator actions. The descriptions of the significant contributors included the F-V for each.

The NRC staff utilized the F-V values to calculate the RRW, the maximum monetary value of completely eliminating the failure, and the contribution to SLERF of each contributor. The results are provided in Tables 3 and 4 for the SLERF contributors for Units 2 and 3, respectively. The listed seismic-induced failures that contribute to SLERF have an RRW greater than about 1.005. These tables provide the following information by column: (1) Description of the component, (2) Failure Mode, (3) RRW, and (4) maximum SLERF reduction (MLR) from completely eliminating the failure. Two SPRA model elements or contributors exceeded the mean target RRW for SLERF. Similar results are provided for SCDF contributors in Tables 1 and 2 for informational purposes only.

These two individual contributors to SLERF individually contribute greater than 1.0E-06 per year. The highest contributor was seismically-induced loss of offsite power (OSP-C) and, according to Table 5.5-6 of the submittal, is a contributor to nine of the top 10 SLERF cutsets. Since the F-V value for OSP-C is 99.9%, based on the standard relationship formula, the corresponding RRW value is 1,000. The NRC staff observes that the formula provides exponential results when F-V values are greater than 90% and therefore the insight of an SSC with an RRW value greater than 10 are determined to be equivalent. During the audit, the licensee explained that OSP-C represents seismic-induced loss of offsite power from both the plant switchyard and from offsite power lines and that the fragility used in the SPRA is a single representative fragility representing both. Because this event involves seismic-induced failures outside of the plant boundary, the NRC staff did not pursue potential improvements to OSP-C. The second highest contributor to SLERF was seismically-induced failure of the reactor pressure vessel internals (SCRAM) such that the control rods cannot be successfully fully inserted, which is also a contributor to the same 9 top 10 SLERF cutsets as OSP-C. In sensitivity Case 4d evaluated in the SPRA submittal, the licensee postulated a plant modification to increase the fragility of the upper and lower clamps on the core shroud tie rods from 0.75g to 1.25g. This postulated modification would reduce SLERF by 37.5 percent, which is greater than 1E-06/rx-yr. However, the licensee determined this plant modification is not cost-justified for the following reasons: (1) the SPRA model conservatively assumes that the failure of the clamps leads to an anticipatory transient without a scram (ATWS), (2) modification to the core shroud tie rods would require significant amount of design planning and approval including a large amount of work within the RPV, and (3) there are several other reactor pressure vessel (RPV) internal components with fragilities less than 1.0g that would also require modification. After evaluating this information, and based on engineering judgement, the staff finds these reasons to be valid. In addition, the staff determined the proposed modification would not affect containment performance. Consequently, the NRC staff did not pursue potential improvements to SCRAM.

The NRC staff also considered combinations of basic events in accordance with the SPRA Screening Guidance. It is not the intent of that aspect of the guidance to aggregate several disparate basic events that individually have RRW values close to the mean target RRW. The total SLERF of the SPRA model seismically-failed elements identified in Tables 3 and 4 is over 8E-06 per reactor-year for both units. A review of these model elements reveals that any modifications or sets of modifications to achieve a SLERF reduction of at least 1.0E-06, other than those discussed above, will have to mitigate or prevent multiple failure types (e.g., seismically-induced failures, random failures[3], and failure of operator actions) and failure modes (e.g., seismically-induced structural failures of multiple SSCs and seismically-induced functional

[3] The licensee provided information on operator actions that are not due to the seismic event in its submittal. The staff included this information as an aid to help identify potential modifications that could reduce the overall SLERF.

failures of multiple SSCs). The NRC staff's assessment of the potential SLERF basic event combinations concluded that there is no apparent synergies or implementation cost efficiencies to be gained from the combinations.

Based on the analysis described above, the NRC staff concludes that no modifications are warranted in accordance with Title 10 of the *Code of Federal Regulations* Section 50.109 (10 CFR 50.109) to reduce SCDF and SLERF because a potential cost-justified substantial safety improvement was not identified.

In accordance with Section 3.3.2 of NUREG/BR-0058, Revision 4, the NRC staff further evaluated Dresden accident sequences impacting the conditional probability of early containment failure or bypass (CPCFB) for seismic events to determine if any substantial safety improvements would reduce the SCDF and related SLERF of those sequences. All the dominant failures are already evaluated, as described above. Also, FLEX strategies and their potential mitigation capabilities were considered in this evaluation.

Based on the available information and engineering judgement, the NRC staff concluded that there were no further potential improvements to containment performance that would rise to the level of a substantial safety improvement or would warrant further regulatory analysis.

Additionally, the NRC staff considered insights from the individual plant examination of external events (IPEEE) and SAMA analyses previously completed for Dresden to understand previous work done to identify substantial safety improvements and to further inform this review. Based on previous evaluations and based on the detailed screening completed as part of this review, no potential improvements were found.

Conclusion

Based on the analysis of the submittal and supplemental information, the NRC staff concludes that no modifications are warranted under 10 CFR Section 50.109 because:

- The staff did not identify a potential modification necessary for adequate protection or compliance with existing requirements;

- no other potential cost-justified substantial safety improvement was identified based on the estimated achievable reduction in SCDF and/or SLERF; and

- additional consideration of containment performance, as described in NUREG/BR-0058 and assessed via SLERF, did not identify a modification that would result in a substantial safety improvement.

Table 1. Importance Analysis Results of Top Contributors to Unit 2 SCDF

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
Loss of offsite power	Functional	2.415	5.07E-06
Control Panel Group C20-4 (Panels 902-15, 902-17, 902-18, 902-19, 902-16, 902-20)	Anchorage	1.084	6.71E-07
Unit 2 125 VDC Battery (549 TB)	Anchorage	1.079	6.30E-07
RPV Internals (SCRAM)	Anchorage	1.073	5.88E-07
Instrument Rack Group 18-9-1-1 (2202-7)	Block Wall	1.049	4.05E-07
Unit 2 125 VDC TRAIN B BUSSES (2-83125) - 549 TB	Functional	1.016	1.38E-07
SBODG2 Battery 6A and SBODG3 Battery 7A	Functional	1.016	1.36E-07
Unit 3 125 VDC Battery (551 TB)	Anchorage	1.015	1.31E-07
Control Panel Group C20-3 (Panels 903-8, 902-8)	Anchorage	1.015	1.24E-07
4160V Buses 23, 24	Functional	1.012	1.03E-07
Crib House	Structure	1.012	9.95E-08
Unit 3 125 VDC Battery Charger #3 - 538 TB	Anchorage	1.010	8.36E-08
Relay Chatter ID 101 (HPCI-Recov.)	Functional	1.008	7.02E-08
Dresden Lock and Dam	Structure	1.008	6.76E-08
Relay Chatter ID 613 (CS B-Unrecov.)	Functional	1.007	6.33E-08
480V MCC 35-2, 38-2, 38-3	Functional	1.007	5.97E-08
Unit 2 125 VDC TRAIN A BUSSES (2-83125) - 549 TB	Functional	1.007	5.92E-08
Control Panel Group C20-1 (Panels 902-3, 903-3, 902-4, 903-4)	Anchorage	1.007	5.92E-08
Control Panel Group C20-7 (Panels 902-33, 903-33, 902-32)	Functional	1.007	5.86E-08
Relay Chatter ID 361A (EDG2, EDG3-Recov.)	Functional	1.006	5.15E-08
4160V AC/ Switchgear 40	Functional	1.005	4.34E-08
Human Failure Events			
Failure To Control Containment Venting	Not Applicable	1.055	4.48E-07
Failure To Inject Through Loop A Given Failure Of Loop B	Not Applicable	1.019	1.60E-07
Operator Fails To Recover From Relay Chatter Impacting EDG 2, 3, And/Or 2/3 (SEISMIC)	Not Applicable	1.017	1.44E-07
Crew Fails To Align RWCU For Letdown	Not Applicable	1.016	1.35E-07
Failure To Close HPCI Steam Line Isolation Valve To Prevent Water Into HPCI Turbine Or Auxiliaries	Not Applicable	1.016	1.35E-07
Failure To Inhibit Automatic Depressurization System (Ads) (No Hp Injection) (ATWS)	Not Applicable	1.009	7.80E-08
Failure To Emergency Vent Containment Using Hard Pipe Vent	Not Applicable	1.009	7.42E-08
Operator Fails To Recover From Relay Chatter Impacting HPCI (SEISMIC)	Not Applicable	1.008	7.00E-08
Failure To Control RPV Level Low (ATWS)	Not Applicable	1.007	5.90E-08

Table 2. Importance Analysis Results of Top Contributors to Unit 3 SCDF

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
Loss of offsite power	Functional	3.012	5.78E-06
RPV Internals (SCRAM)	Anchorage	1.076	6.08E-07
Unit 3 125 VDC Battery (551 TB)	Anchorage	1.056	4.58E-07
Control Panel Group C20-4 (Panels 902-15, 902-17, 902-18, 902-19, 902-16, 902-20)	Anchorage	1.045	3.69E-07
Unit 2 125 VDC Battery (549 TB)	Anchorage	1.039	3.24E-07
Control Panel Group C20-4-1 (Panels 903-15, 903-17, 903-18, 903-19, 903-16, 903-20)	Anchorage	1.037	3.10E-07
4160V Buses 33, 34	Functional	1.021	1.76E-07
Unit 3 125 VDC Battery Charger #3 - 538 TB	Anchorage	1.019	1.59E-07
SBODG2 Battery 6A and SBODG3 Battery 7A	Functional	1.015	1.31E-07
Control Panel Group C20-3 (Panels 903-8, 902-8)	Anchorage	1.014	1.18E-07
Crib House	Structure	1.012	1.04E-07
Relay Chatter ID 521 (CS B-Recov.)	Functional	1.011	9.00E-08
Relay Chatter ID 451 (Bus 29 feed to 480 VAC MCC 29-7)	Functional	1.010	8.57E-08
Relay Chatter ID 462 (Bus 29 feed to 480 VAC MCC 29-7)	Functional	1.010	8.57E-08
Control Panel Group C20-8 (Panels 903-32)	Anchorage	1.008	7.05E-08
4160V AC/ Switchgear 40	Functional	1.008	6.59E-08
Dresden Lock and Dam	Structure	1.007	6.38E-08
Control Panel Group C20-1 (Panels 902-3, 903-3, 902-4, 903-4)	Anchorage	1.007	6.31E-08
Instrument Rack Group 18-9-1-1 (2202-7)	Block Wall	1.007	5.91E-08
Control Panel Group C20-7 (Panels 902-33, 903-33, 902-32)	Functional	1.007	5.61E-08
Relay Chatter ID 520A (CS A and B-Recov.)	Functional	1.006	4.78E-08
EDG #3 Excitation Cabinet (3-2253-21)	Anchorage	1.006	4.76E-08
Unit 2 125 VDC TRAIN B BUSSES (2-83125) - 549 TB	Functional	1.005	4.66E-08
Relay Chatter ID 386 (EDG2-Recov.)	Functional	1.005	4.43E-08
Human Failure Events			
Failure To Control Containment Venting	Not Applicable	1.063	5.10E-07
Failure To Close HPCI Steam Line Isolation Valve To Prevent Water Into HPCI Turbine Or Auxiliaries	Not Applicable	1.035	2.93E-07
Crew Fails To Align RWCU For Letdown	Not Applicable	1.035	2.93E-07
Failure To Align Portable Battery Chargers	Not Applicable	1.015	1.28E-07
Operator Fails To Recover From Relay Chatter Impacting EDG 2, 3, And/Or 2/3 (SEISMIC)	Not Applicable	1.015	1.24E-07
Failure To Emergency Vent Containment Using Hard Pipe Vent	Not Applicable	1.011	9.34E-08
Failure To Inhibit Ads (No Hp Injection) (ATWS)	Not Applicable	1.008	7.04E-08
Failure To Align Alternate Battery Given Dual Unit Loop	Not Applicable	1.006	5.39E-08
Failure To Control RPV Level Low (ATWS)	Not Applicable	1.006	5.32E-08

Table 3. Importance Analysis Results of Top Contributors to Unit 2 SLERF

Description	Failure Mode	RRW	MLR (/yr)
Seismically-failed SSCs			
Loss of offsite power	Functional	>>10	4.32E-06
RPV Internals (SCRAM)	Anchorage	2.857	2.81E-06
Unit 2 125 VDC Battery (549 TB)	Anchorage	1.136	5.18E-07
Instrument Rack Group 18-9-1-1 (2202-7)	Block Wall	1.038	1.60E-07
Control Panel Group C20-4 (Panels 902-15, 902-17, 902-18, 902-19, 902-16, 902-20)	Anchorage	1.025	1.07E-07
Unit 3 125 VDC Battery (551 TB)	Anchorage	1.019	7.86E-08
480V MCC 35-2, 38-2, 38-3	Functional	1.016	6.83E-08
Unit 2 125 VDC TRAIN A BUSSES (2-83125) - 549 TB	Functional	1.015	6.18E-08
Unit 2 125 VDC TRAIN B BUSSES (2-83125) - 549 TB	Functional	1.014	5.79E-08
Unit 3 125 VDC Battery Charger #3 - 538 TB	Anchorage	1.012	5.23E-08
Control Panel Group C20-7 (Panels 902-33, 903-33, 902-32)	Functional	1.012	5.10E-08
Control Panel Group C20-1 (Panels 902-3, 903-3, 902-4, 903-4)	Anchorage	1.010	4.13E-08
Relay Chatter ID 483 (CS A-Recov.)	Functional	1.007	2.92E-08
Relay Chatter ID 101 (HPCI-Recov.)	Functional	1.006	2.74E-08
4160V AC/ Switchgear 40	Functional	1.006	2.44E-08
Control Panel Group C20-3 (Panels 903-8, 902-8)	Anchorage	1.005	2.35E-08
Crib House	Structure	1.005	2.34E-08
SBODG2 Battery 6A and SBODG3 Battery 7A	Functional	1.005	2.28E-08
480V MCC 28-2 and 28-3	Functional	1.005	2.19E-08
Human Failure Events			
Failure to align portable battery chargers	Not Applicable	1.023	9.85E-08
Failure to inhibit ADS (no HP injection) (ATWS)	Not Applicable	1.022	9.46E-08
Failure to inject through Loop A given failure of Loop B	Not Applicable	1.015	6.35E-08
Operator Fails to Recover from Relay Chatter Impacting EDG 2, 3, and/or 2/3 (SEISMIC)	Not Applicable	1.012	5.31E-08
Operator fails to depressurize the RPV begore vessel failure	Not Applicable	1.009	3.65E-08
Failure to initiate SLC early	Not Applicable	1.008	3.29E-08
Failure to shed 25V DC loads (under SBO conditions)	Not Applicable	1.007	3.06E-08
Failure to depressurize the RPV (ADS) (ATWS)	Not Applicable	1.007	2.83E-08
Operator Fails to Recover from Relay Chatter Impacting HPCI (SEISMIC)	Not Applicable	1.006	2.74E-08

Table 4. Importance Analysis Results of Top Contributors to Unit 3 SLERF

Description	Failure Mode	RRW	MLR (/yr)
Seismically-failed SSCs			
Loss of offsite power	Functional	>>10	4.32E-06
RPV Internals (SCRAM)	Anchorage	2.967	2.86E-06
Unit 3 125 VDC Battery (551 TB)	Anchorage	1.116	4.49E-07
Unit 2 125 VDC Battery (549 TB)	Anchorage	1.032	1.35E-07
Unit 3 125 VDC Battery Charger #3 - 538 TB	Anchorage	1.023	9.68E-08
Control Panel Group C20-4 (Panels 902-15, 902-17, 902-18, 902-19, 902-16, 902-20)	Anchorage	1.017	7.04E-08
Control Panel Group C20-8 (Panels 903-32)	Anchorage	1.015	6.48E-08
Control Panel Group C20-1 (Panels 902-3, 903-3, 902-4, 903-4)	Anchorage	1.008	3.56E-08
Control Panel Group C20-4-1 (Panels 903-15, 903-17, 903-18, 903-19, 903-16, 903-20)	Anchorage	1.008	3.27E-08
Relay Chatter ID 521 (CS B-Recov.)	Functional	1.007	3.14E-08
480V MCC 28-2 and 28-3	Functional	1.007	3.01E-08
4160V AC/ Switchgear 40	Functional	1.007	2.93E-08
Relay Chatter ID 451 (Bus 29 feed to 480 VAC MCC 29-7)(1)	Functional	1.005	2.25E-08
Relay Chatter ID 462 (Bus 29 feed to 480 VAC MCC 29-7)(2)	Functional	1.005	2.25E-08
Control Panel Group C20-7 (Panels 902-33, 903-33, 902-32)	Functional	1.005	2.20E-08
Relay Chatter ID 364 (EDG2/3-Recov.)	Functional	1.005	2.19E-08
Human Failure Events			
FAILURE TO ALIGN PORTABLE BATTERY CHARGERS	Not Applicable	1.044	1.82E-07
FAILURE TO INHIBIT ADS (NO HP INJECTION) (ATWS)	Not Applicable	1.021	8.68E-08
OPERATOR FAILS TO DEPRESSURIZE THE RPV BEFORE VESSEL FAILURE	Not Applicable	1.011	4.84E-08
Operator Fails to Recover From Relay Chatter Impacting EDG 2, 3, and/or 2/3 (SEISMIC)	Not Applicable	1.010	4.19E-08
FAILURE TO DEPRESSURIZE THE RPV (ADS) (ATWS)	Not Applicable	1.008	3.23E-08
FAILURE TO INITIATE SLC EARLY	Not Applicable	1.007	3.02E-08
FAILURE TO SHED 125V DC LOAD (UNDER SBO CONDITIONS)	Not Applicable	1.006	2.52E-08
FAILURE TO ALIGN ALTERNATE BATTERY GIVEN DUAL UNIT LOOP	Not Applicable	1.005	2.36E-08

AUDIT SUMMARY BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO

DRESDEN NUCLEAR POWER STATION, UNITS 2 & 3

SUBMITTAL OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH

REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE

NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC

(EPID NO. L-2019-JLD-0016)

BACKGROUND AND AUDIT BASIS

By letter dated March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12053A340), the U.S. Nuclear Regulatory Commission (NRC) issued a request for information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(f) (hereafter referred to as the 50.54(f) letter). Enclosure 1 to the 50.54(f) letter requested that licensees reevaluate the seismic hazards for their sites using present-day methods and regulatory guidance used by the NRC staff when reviewing applications for early site permits and combined licenses.

By letter dated October 27, 2015 (ADAMS Accession No. ML15194A015), the NRC made a determination of which licensees were to perform: (1) a Seismic Probabilistic Risk Assessment (SPRA), (2) limited scope evaluations, or (3) no further actions based on a comparison of the reevaluated seismic hazard and the site's design-basis earthquake. (Note: Some plant-specific changes regarding whether an SPRA was needed or limited scope evaluations were needed at certain sites have occurred since the issuance of the October 27, 2015, letter).

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the letter issued pursuant to Title 10 CFR Part 50, Section 50.54(f). The list of applicable licensees in Enclosure 1 to the July 6, 2017, letter included Exelon Generation Company, LLC (Exelon, the licensee), for Dresden Nuclear Power Station, Units 2 & 3 (Dresden).

REGULATORY AUDIT SCOPE AND METHODOLOGY

The areas of focus for the regulatory audit are the information contained in the SPRA submittal (ADAMS Accession No. ML19304B567), its supplement (ADAMS Accession No. ML20066K784), and all associated and relevant supporting documentation used in the development of the SPRA submittal including, but not limited to, methodology, process information, calculations, computer models, etc.

AUDIT ACTIVITIES

The NRC staff developed questions to verify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA submittal. The non-docketed information was available for audit via an electronic portal. Following LIC-111, access to that portal was requested for the audit team (ADAMS Accession No. ML19309F953). After that, the staff's clarification questions (ADAMS Accession Nos. ML19333B896 and ML20035F145, respectively) were sent to the licensee. The licensee provided clarifying information in the following areas:

- Supporting information associated with structural modeling, structural response analysis, fragility analysis, capacities of relays sensitive to high frequencies, and selection of significant risk contributors.

- Resolution of internal events probabilistic risk assessment finding level facts and observations (F&Os) and associated peer review reports and clarification if those resulted in updates or maintenance for the PRA.

- Use of insights from the Peach Bottom State-of-the-Art Reactor Consequence Analysis (SOARCA) in the Dresden SPRA.

- The independent assessment report completed following the guidance in Nuclear Energy Institute (NEI) Standard 12-13, Appendix X, "Close-out of Facts and Observations," (ADAMS Accession No. ML17086A431).

The licensee's response to the questions aided in the staff's understanding of the Dresden SPRA docketed submittal. Following the review of the licensee's response and the supporting documents provided by the licensee on the ePortal, the staff determined that no additional documentation or information was needed to supplement Dresden docketed SPRA submittal.

DOCUMENTS AUDITED

- Exelon Document DR-PRA-020.005, Revision 1, Volume 1, "Dresden Seismic Probabilistic Risk Assessment Fragility Modeling Notebook," October 16, 2019.

- Exelon Document DR-PRA-020.005, Revision 1, Volume 2, "Dresden Seismic Probabilistic Risk Assessment Seismic Equipment List Notebook," October 16, 2019.

- Exelon Document DR-PRA-020.007, Revision 1, "Dresden Seismic Probabilistic Risk Assessment Seismic PRA Walkdown Notebook," October 16, 2019.

- Dresden Generating Station Units 2 & 3 PRA Peer Review Report Using ASME PRA Standard Requirements, January 2017.

- Dresden Generating Station 2009 Internal Flood PRA Peer Review Report Using ASME PRA Standard Requirements, August 2009.

- Jensen Hughes Report 032362-RPT-001, Revision 0, "2017 Risk Management (Dresden 2 and 3) Finding-Level F&O Independent Technical Review," June 2018.

- Exelon Document DR-PRA-20.006, Revision 0, "Dresden Seismic Probabilistic Risk Assessment Seismic Quantification Notebook," October 18, 2019.

- Dresden Generating Station Seismic PRA Peer Review Report, March 2019.

- Exelon Document DR-PRA-20.004, Revision 1, “Dresden Seismic Probabilistic Risk Assessment Seismic Human Reliability Analysis (HRA) Notebook,, October 18, 2019.

- Exelon Document DR-PRA-20.001, Revision 1, “Dresden Seismic Probabilistic Risk Assessment Seismic Methods Notebook,” October 16, 2019.

- Exelon Document DR-PRA-020.002, Revision 1, “Dresden Seismic Probabilistic Risk Assessment Seismic Initiating Event Notebook,” October 16, 2019.

- Exelon Document DR-PRA-020.003, Revision 1, “Dresden Seismic Probabilistic Risk Assessment Event Tree Notebook,” October 16, 2019.

OPEN ITEMS AND REQUEST FOR INFORMATION

There were no open items identified by the NRC staff that required proposed closure paths and there were no requests for information discussed or planned to be issued based on the audit.

DEVIATIONS FROM AUDIT PLAN

There were no deviations from the July 6, 2017, generic audit plan.

AUDIT CONCLUSION

The issuance of this document, containing the staff’s review of the SPRA submittal, concludes the SPRA audit process for Dresden.

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 & 3 – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2019-JLD-0016) DATE: JUNE 5, 2020

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