UNITED STATES
NUCLEAR REGULATORY COMMISSION
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September 23, 2020

Mr. Joel P. Gebbie
Senior Vice President and Chief
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One Cook Place
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SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 – 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-017)

Dear Mr. Gebbie:

The purpose of this letter is to document the staff's evaluation of the Donald C. Cook Nuclear Plant, Units 1 and 2 (D.C. Cook, CNP), 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 D.C. Cook.

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 November 4, 2019 (ADAMS Accession No. ML19310D805), Indiana Michigan Power Company (I&M, the licensee), provided its SPRA submittal in response to Enclosure 1, Item (8) of the 50.54(f) letter, for D.C. Cook. The SPRA submittal was later supplemented by letter dated July 16, 2020 (ADAMS Accession No. ML20206K894). As applicable, 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). This report was endorsed by the NRC by letter dated February 15, 2013 (ADAMS Accession No. ML12319A074). In addition, consistent with the licensee's submittal, the NRC staff utilized a reviewer checklist that is based on ASME (American Society of Mechanical Engineers)/ANS (American Nuclear Society) RA-S Case 1 "Case for ASME/ANS Ra-Sb-2013, Standard for Level 1/Large Early Release Frequency

Probabilistic Risk Assessment for Nuclear Power Plant Applications" (herein called the "Code Case Standard"). Use of this reviewer checklist for licensees choosing to use the Code Case Standard was described in a letter to the Nuclear Energy Institute (NEI) dated July 12, 2018 (ADAMS Accession No. ML18173A017). The reviewer checklist for the NRC staff's assessment of the CNP SPRA submittal is contained in Enclosure 1 to this letter. As described below, the NRC staff has concluded that the CNP 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 27, 2014 (ADAMS Package Accession No. ML14092A327), I&M submitted the reevaluated seismic hazard information for CNP. The NRC performed a staff assessment of the submittal and issued a response letter on April 21, 2015 (ADAMS Accession No. ML15097A196). The NRC's assessment concluded that I&M 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 at CNP.

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) 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, CNP was expected to complete an SPRA with an estimated completion date of June 30, 2018, which would also assess high frequency ground motion effects. By letter dated November 15, 2017 (ADAMS Accession No. ML17321A083), the licensee requested to extend the SPRA submittal to November 6, 2019. The NRC staff responded in a letter dated February 1, 2018 (ADAMS Accession No. ML18011A217). In addition, I&M was expected to perform a limited-scope evaluation for the CNP spent fuel pool (SFP). This SFP limited-scope evaluation was submitted by letter dated October 12, 2016 (ADAMS Accession No. ML16288A843). The staff provided its assessment of the CNP SFP evaluation by letter dated November 9, 2016 (ADAMS Accession No. ML16308A086).

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

In its letter dated November 4, 2019, I&M provided the SPRA submittal that initiated the NRC's Phase 2 decisionmaking process for CNP. The NRC described this Phase 2 decision making 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 describes 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 screening recommendations 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 CNP is described below. Based on its evaluation, the staff recommended to the SMRP that CNP be classified as a Group 1 plant and therefore, no further regulatory action was warranted.

EVALUATION

Upon receipt of the licensee's SPRA submittal dated November 4, 2019, a technical team of NRC staff members 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 December 9, 2019 (ADAMS Accession No. ML19345F938), 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 submittal dated November 4, 2019, the licensee developed and documented the SPRA to respond to Enclosure 1 of the 50.54(f) letter, Item 8(b) and Section 6.8 of the SPID. The SPRA included performance of an independent peer review against the Code Case Standard which is summarized in Appendix A of the licensee's submittal. In addition, the licensee also performed a closeout independent assessment of the resolution of the finding level facts and observations (F&Os) from the full-scope peer review following the process accepted by the NRC (ADAMS Accession No. ML17079A427). The closeout independent assessment also included a concurrent focused-scope peer review for upgrades to the SPRA. Appendix A of the licensee's submittal provided a summary of the full-scope and closeout independent assessment peer reviews, including excerpts from the corresponding peer review reports. Appendix A of the licensee's submittal also included a summary of the SPRA finding level F&Os that remained open after the closeout process, along with the licensee's dispositions and the impact the open F&Os may have on the SPRA results. These elements 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 I&M as the licensee for CNP. The staff exercised the audit process by reviewing selected licensee documents via an electronic reading room (e-portal) as documented in Enclosure 3 to this letter.

During the audit process, the staff developed questions to clarify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA submittal. The staff's clarification questions dated March 25, 2020, and April 27, 2020 (ADAMS Accession Nos. ML20232B443 and ML20232B456, respectively), were sent to the licensee to support the audit. The licensee subsequently provided answers to the questions on the e-portal, which the staff reviewed. The staff determined that the answers to the questions provided in the e-portal served to confirm statements that the licensee made in its November 4, 2019, SPRA submittal.

Since the licensee's internal events PRA (IEPRA) model was used as the basis for the development of the SPRA model, the NRC staff reviewed the IEPRA F&Os and the associated dispositions during the SPRA audit process to assess any potential impact on the SPRA submittal. The NRC staff confirmed that the licensee's dispositions to these findings were appropriately incorporated into the SPRA model and did not identify any modeling issues that could impact the conclusions of the SPRA submittal.

Based on the staff's review of the licensee's submittal, including the resolution of the peer review findings as described above, 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.

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 Code Case Standard, as described in the NRC letter to the NEI dated July 12, 2018. Enclosure 1 contains the staff's application of the SPRA checklist to CNP's submittal. As documented in the checklist, the staff concluded that the CNP SPRA meets the intent of the SPID guidance, including the documentation requirements of the Code Case Standard.

Following the staff's conclusion on the SPRA's technical adequacy, the staff reviewed the risk and safety insights contained in the CNP SPRA submittal. The staff also used the screening criteria described in a staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200), 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" to guide its review and screening recommendation to the SMRP. The criteria in the staff's guidance document includes thresholds to assist in determining whether to apply the backfit screening process described in Management Directive 8.4, "Management of Facility-Specific Backfitting, Forward Fitting, Issue Finality, and Information Requests," dated September 20, 2019 (ADAMS Accession No. ML18093B087), to the SPRA submittal review. As part of this review the staff considered the planned modifications described in the CNP SPRA. By letter dated July 16, 2020 (ADAMS Accession No. ML20206K894), I&M provided a regulatory commitment to perform the plant modifications described in the SPRA submittal. These modifications are expected to reduce the SLERF values for both units at CNP. The CNP SPRA submittal demonstrated that the plant SCDF and SLERF for both units were not below the initial screening values in the staff memorandum dated August 29, 2017. Based on the SCDF and SLERF results, the NRC staff utilized the CNP SPRA submittal and other available information in conjunction with the guidance in the staff memorandum dated August 29, 2017, to complete a detailed screening evaluation. The detailed screening concluded that CNP should be considered a Group 1 plant because:

- Sufficient reductions in SCDF and SLERF cannot be achieved by potential modifications considered in this evaluation, other than those identified by the licensee and included as a regulatory commitment, 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, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," 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 the detailed screening evaluation and its review of the CNP SPRA submittal, the technical team determined that recommending CNP to be classified as a Group 1 plant was appropriate and additional review and/or analysis to pursue a plant-specific backfit was not warranted. This determination assumes that the modifications described in the licensee's submittal and supplemental regulatory commitment are implemented as planned.

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 will be providing SPRA submittals. The technical review team provided the results of the CNP review to the TRB with the Phase 2 recommendation that CNP be categorized as a Group 1 plant, meaning that no further response or regulatory actions are required. The TRB members assessed the information presented by the technical team and agreed with the team's recommendation for classification of CNP as a Group 1 plant.

Subsequently, the technical review team consulted with the SMRP and presented the results of the review including the recommendation for CNP 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 team. The SMRP approved the staff's recommendation that CNP should be classified as a Group 1 plant, meaning that no further response or regulatory action is required.

AUDIT REPORT

The generic audit plan dated July 6, 2017, 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 included a review of licensee documents through an electronic reading room. An audit summary document is included as Enclosure 3 to this letter.

REGULATORY COMMITMENT

In its supplemental letter dated July 16, 2020 (ADAMS Accession No. ML20206K894), the licensee provided a regulatory commitment to complete permanent plant modifications. The NRC staff notes that NEI 99-04, "Guidelines for Managing NRC Commitments" (ADAMS Accession No. ML003680088), as endorsed by the NRC in SECY-00-0045, "Acceptance of NEI 99-04, "Guidelines for Managing NRC Commitments"" (ADAMS Accession No. ML003679799), provides an acceptable method to manage commitments. If the licensee were to change this regulatory commitment, the staff expects to be informed in accordance with the process outlined in NEI 99-04, as endorsed by the NRC. If the commitment were to be changed, the staff may revisit its conclusion.

CONCLUSION

Based on the staff's review of the CNP 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. The staff notes that this conclusion is dependent on the completion of the planned modifications, as described in the SPRA submittal.

Application of this review is limited to the review of the 10 CFR 50.54(f) response associated with NTTF Recommendation 2.1 "Seismic." 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 Stephen Philpott at (301) 415-2365 or via e-mail at Stephen.Philpott@nrc.gov.

Sincerely,

/RA/

David J. Wrona, Acting Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

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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SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 – 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-0017) DATED SEPTEMBER 23, 2020

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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 submittals to satisfy Near-Term Task Force (NTTF) Recommendation 2.1: Seismic. These submittals are being 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 U.S. Nuclear Regulatory Commission (NRC) staff for this purpose. The SPRA peer reviews are also expected to follow the guidance in NEI 12-13 (NEI, 2012) as supplemented by NRC staff comments in its acceptance letter dated March 7, 2018 (NRC, 2018a, 2018b).

The SPID indicates that an SPRA submitted for the purpose of satisfying NTTF Recommendation 2.1: Seismic (hereafter referred to as NTTF Recommendation 2.1) must meet the requirements in the ASME-ANS [American Society of Mechanical Engineers-American Nuclear Society] PRA Methodology Standard (the ASME-ANS Standard). According to the SPID, 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.

Recently, the ASME-ANS Joint Committee on Nuclear Risk Management (JCNRM), which develops and maintains the PRA standards at issue, has issued a new set of requirements for Seismic PRAs, ASME/ANS RA-S Case 1 (ASME/ANS, 2017), herein called the “Code Case Standard.” The Code Case Standard contains alternative requirements to Addendums A and B for Part 5 (SPRA) of the PRA Standard. The reasons for developing the Code Case Standard were to make the SPRA requirements more consistent in some areas with the rest of the standard, and also to respond to comments from users concerning the scope or the level of detail of some of the requirements.

The use of the Code Case Standard by a licensee is voluntary, but it is the NRC staff’s understanding that some nuclear power plant licensees will be developing and subsequently submitting their SPRAs in response to NTTF Recommendation 2.1 using the Code Case Standard instead of either the Addendum A or the Addendum B version.

The NRC staff wrote a letter to the JCNRM on March 12, 2018 (NRC, 2018), which states in part that, “The NRC staff finds the process for developing a PRA for seismic events proposed in the ASME/ANS RA-S Case 1 acceptable,” while also setting forth some conditions that must be met by a licensee’s submittal if the Code Case Standard is used. Specifically, an attachment to that letter contains detailed staff comments on the Code Case Standard that need to be addressed by any submittal that references the Code Case Standard. As stated in the staff’s March 2018 letter “[l]icensees may choose to retain their facility’s current SPRA approach or revise it consistent with the Code Case. Any licensee use of the Code Case is voluntary.”

The purpose of this staff guidance document (checklist) is to provide guidance and a checklist to the staff for the review of prospective licensee submittals using the Code Case Standard, similar to the earlier guidance and checklist (NRC, 2017) covering submittals using either the 2009 Addendum A version or the 2013 Addendum B version of the Standard.

This new staff guidance document (and checklist) is a stand-alone document. It does, however, rely heavily on the guidance material in the earlier staff guidance and checklist document, and uses a vast majority of the material in the earlier document directly.

The following table provides a checklist covering each of the Supporting Requirements (SRs) in the Code Case Standard. For most SRs, the SPID guidance does not differ from the requirement in the Code Case Standard. However, because the guidance in the SPID and the criteria of the Code Case Standard differ in some areas, or the SPID does not explicitly address an SR, the staff has developed the checklist to help NRC reviewers to address and evaluate the differences, as well as to determine the appropriate technical requirement (Code Case Standard or SPID) against which the SPRA for NTTF Recommendation 2.1 submittals should be reviewed.

In general, the SPID allows departures or differs 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 Code Case 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 Code Case Standard.

(iii) The SPID has some requirements that are not in the Code Case Standard.

All of the technical positions in the SPID have been endorsed by the NRC staff for NTTF Recommendation 2.1 submittals, subject to certain conditions concerning peer review outlined in the staff's letter to NEI dated March 7, 2018 (NRC, 2018a, 2018b), which supersedes the staff's November 12, 2012, letter to NEI (NRC, 2012).

The checklist in this document is comprised of the 16 "Topics" that require additional staff guidance because the SPID contains specific guidance that differs from the Code Case Standard or expands on it. The earlier checklist covering staff review of submittals using Addendum A or Addendum B of the ASME-ANS Standard was discussed during a public meeting on December 7, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16350A181). Each topic is covered below under its own heading, "Topic 1," "2," etc.

- 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 SSE [Safe Shutdown Earthquake] - to- GMRS [Ground Motion Response Spectra] - 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, ISRS [In-Structure Response Spectra], and Fragilities
- Topic 8: Screening by Capacity to Select SSCs [Structures, Systems, and Components] for Seismic Fragility Analysis (SPID Section 6.4.3)
- Topic 9: Use of the CDFM [Conservation Deterministic Failure Margin]/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 LERF [Large Early Release Frequency] (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 (PSHA) 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 peer reviewers. The relevant peer review findings are those that relate to the SHA requirements in the Code Case Standard, as well as to the requirements in the SPID.	Yes
• although some peer review findings and observations 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 Indiana Michigan Power Company (I&M, the licensee) for the Donald C. Cook Nuclear Plant (CNP), updated its site response analysis to incorporate the beach sand layer that was omitted from its Seismic Hazard and Screening Report (SHSR) submittal (ADAMS Package No. ML14092A327). The NRC staff assessment (ADAMS Accession No. ML15097A196) of the SHSR submittal included the staff's conclusion that once it is adjusted for a layer of beach sand, the CNP seismic hazard reevaluation would be suitable for other activities associated with the NRC NTTF Recommendation 2.1: Seismic. This sand layer is included in the Foundation Input Response Spectra (FIRS) calculations and is appropriately propagated through the SPRA. 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 peer reviewers. The relevant peer review findings are those that relate to all SRs under HLR-SHA-E in the Code Case Standard, as well as to the requirements in the SPID.	Yes
• although some peer review findings and observations 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 wave 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 approaches 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 and previously accepted by the staff.	Yes
If yes, the control point can be used in the SPRA and the NRC staff's earlier acceptance governs.	
If no, 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 peer reviewers. The relevant peer review findings are those that relate to the requirements in the SPID. No requirements in the Code Case Standard specifically address this topic.	Yes

• Although some peer review findings and observations 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	
Used an existing structural model	No
Used an enhancement of an existing model	Yes
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 SPRA submittal describes the seismic analysis of the structures that support the safety-related components and systems. Table 4-1 of the submittal provides a summary of the structural modeling, foundation conditions, and the analytical approaches used for the Containment, Auxiliary Building, and Turbine Building/Screenhouse. These buildings included structures, systems, and components (SSCs) in the Seismic Equipment List (SEL). The Auxiliary Building is primarily a concrete structure, while the Turbine Building, including the Screenhouse, is a steel structure with a concrete substructure. The Containment Buildings, one for each unit, consist of a concrete containment shell and dome structure, primary shield wall and crane wall.

New three-dimensional (3D) finite element models were developed for the Auxiliary Building and Turbine Building/Screenhouse. The Containment Building structure and the internals, including the East Main Steam Stop Enclosure, were represented by a combined Lumped Mass Stick Model (LMSM). Based on the NRC staff review of supporting documents, the LMSM for the Containment Building was enhanced from the previously developed model. These three buildings are constructed on separate foundations and therefore, were analyzed independently. The Auxiliary Building and the Turbine Building/Screenhouse models took advantage of the symmetry between Units 1 and 2.

Section 4.3 of the SPRA submittal explains that CNP is a soil site requiring soil-structure interaction (SSI) analysis for evaluating building responses. Deterministic SSI analyses were performed accounting for the uncertainty in the soil response. The results from the SSI analysis was used to develop median and 84th percentile Non-Exceedance Probability (NEP) in-structure response spectra (ISRS) where the SEL systems and components are located. The demand from the ISRS was used for fragility evaluation of the SEL components.

Using the audit process, the NRC staff confirmed that the potential impact between the Turbine Building and Auxiliary Building was appropriately addressed. The resolution of peer review finding level facts and observations (F&O) 22-4 (part 5) concluded that the measured gap between the buildings was higher than originally anticipated, thus an impact between the buildings is not considered credible. The licensee identified collapse of the Turbine Building as a potential hazard to the adjacent buildings (including Auxiliary Building and Screenhouse), and to the equipment inside the Turbine Building, and evaluated structural fragility of the Turbine Building. The licensee addressed potential effects of soil-liquefaction, lateral spreading, and settlement at the site and screened out these hazards from further consideration in the SPRA based on site-specific evaluation.

The NRC staff used the audit process to assess the structural modeling and response analyses and confirmed that that SPID Section 6.3.1 criteria 1 through 7 were appropriately addressed. The 3D-finite element and LMSM structural models are sufficient to capture the overall structural response, torsional effects resulting from eccentricities, and in-plane floor flexibility. The NRC staff's audit review confirmed that appropriate modes of vibration of the structures were considered in the analysis and the modeling approaches applied the requirements of American Society of Civil Engineers (ASCE) Standard 4-16 (ASCE, 2017). Thus, NRC staff finds that SPID Section 6.3.1 criteria 1 through 7 were met for the Auxiliary Building and Turbine Building/ Screenhouse Building structural models. Individual LMSMs in the Containment Building do not satisfy all of criteria 1 through 7 because vertical ground motion was not used in the evaluation of the structural response. In response to an NRC audit question, the licensee stated that given the thick floor slab and proximity of the SSCs to vertical structural elements, the effect of vertical floor amplifications is expected to be negligible. The staff review found that the SPID criteria for the combined Containment Building with multiple LMSMs are satisfied.

There are seven F&Os associated with Code Case Standard SR SFR-B3, which requires use of realistic mathematical models for evaluating building responses. F&Os 28-5, 28-6, and 28-7 relate to appropriate and realistic consideration of mass in the structural models, and F&O 28-10 relates to consideration of embedment depths for the Auxiliary Building in the foundation slab model. These F&Os were closed using the NRC-accepted independent assessment process outlined in Appendix X to NEI 12-13. For F&Os 2-1, 28-2, and 28-4, the licensee concluded that these F&O are not likely to impact SPRA results. The disposition of F&Os 2-1 and 28-4 stated that cracking predicted in the Auxiliary Building model analyses was not significant enough to have an effect on the ISRS amplitude, and therefore damping for uncracked concrete was used in the analysis. The licensee stated in the supporting documents that use of results of the uncracked model and associated damping introduces a small conservative bias. In response to the staff audit question on the disposition of F&O 28-2, the licensee stated that the modeling simplifications used for Containment Building LMSM are justified based on the model results and sensitivity analysis, which considered Nuclear Steam Supply System (NSSS) and non-symmetrical elements, e.g., primary shield wall, crane wall, etc., including their rotational and torsional effects due to eccentricity. The licensee further stated that the combination of the NSSS components LMSM with the Containment Building LMSM meets the dynamic coupling criteria of ASCE 4-16 (ASCE, 2017). The staff reviewed the closeout dispositions, licensee audit responses, and supporting information provided during the audit, and concludes that the resolutions of these F&Os are acceptable.

Thus, NRC staff finds that SPID Section 6.3.1 Criterial 1 through 7 were met and that D.C. Cook used realistic mathematical models to represent the 3D dynamic

characteristics of the building structures for seismic response calculations in accordance with ASME/ANS Code Case SFR-B3 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 peer reviewers. The relevant peer review findings are those that relate to the SR requirement SFR-B3 in the Code Case Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	Yes
• 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.	No
If yes, on which structure(s)? Structure #1 name:	N/A
Structure #1: If used, is V_S > about 5,000 feet (ft.)/second (sec.)?	N/A
If 3,500 ft./sec. < V_S < 5000, was peak-broadening or peak shifting used?	N/A
Potential Staff Finding: The demonstration of the appropriateness of using this approach is adequate.	N/A
Notes from staff reviewer: The SPRA submittal states in Section 4.3.2 that the CNP site is characterized as a soil site and all safety-related structures are founded or embedded on soil. The NRC staff confirmed from Table 3-2 of the SPRA submittal that the shear wave velocity (Vs) of the soil is less than 1500 ft./sec down to a depth of approximately 170 feet. The soil layer is underlain by Paleozoic sedimentary rocks with Vs greater than 6000 ft/sec. The staff concludes that CNP's consideration of SSI effects for structural response analysis is justified. There were no F&Os associated with fixed-base analysis. 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 peer reviewers.	N/A

The relevant peer review findings are those that relate to the requirements in the SPID. No requirements in the Code Case Standard specifically address this topic.	
• Although some peer review findings and observations 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.	N/A
• 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.	Yes
If no, this issue is moot.	
If yes, on which structure(s)? Nuclear Steam Supply System (NSSS) components, including the reactor vessel, steam generators, reactor coolant pumps, pressurizer, and piping.	
Scaling based on:	
Previously developed In-Structure Response Spectra (ISRS)	Yes
Shapes of previous uniform hazard spectrum/review-level earthquake (UHS/RLE)	Yes
Shapes of new UHS/RLE	Yes
Structural natural frequencies, mode shapes, participation factors	Yes
Potential Staff Findings:	
If a new UHS or RLE is used, the shape is approximately similar to the spectral shape previously used for ISRS generation.	Yes
If the shape is not similar, the justification for seismic response scaling is adequate.	N/A
Consideration of non-linear effects is adequate.	N/A
Notes from staff reviewer: In Section 4.4.2.2 of the SPRA report submittal and in response to an NRC staff audit question, the licensee identified that scaling of demand was used to evaluate the fragility of the NSSS components defined above. The previous safe shutdown earthquake (SSE) ISRS was scaled to the Review Level earthquake (RLE) ISRS in accordance with SPID Section 6.3.2, with consideration of structural natural frequencies, mode shapes, and participation factors. The NRC staff concludes that the scaling approach was appropriate and that it meets the adequacy of structural models, foundation characteristics, and similarity of input ground motion as required in ASME/ANS Code Case requirement SFR-B2. There were no F&Os related to SFR-B2.	

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 peer reviewers. The relevant peer review findings are those that relate to the SR requirement SFR-B2 in the Code Case Standard, as well as to the requirements in the SPID.	N/A
• Although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of seismic response scaling adequately meets the intent of the SPID guidance.	Yes
• 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 standard. See all of the SR requirements under HLR-SFR-B in the Code Case Standard. 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 soil-structure-interaction 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. FIRS site response developed with appropriate building specific soil velocity profiles. Structure #1 name: Containment Building (CB) Structure #2 name: Auxiliary Building (AB) Structure #3 name: Turbine Building/Screenhouse (TB) Are all structures appropriately considered?	Yes Yes
2. Are models adequate to provide realistic structural loads and response spectra for use in the SPRA? 1. Is the SSI analysis capable of capturing uncertainties and realistic? 2. Is the probabilistic response analysis capable of providing the full distribution of the responses?	Yes Yes N/A

Notes from staff reviewer:

The SPRA submittal described in Section 4.3.2 that the structural response evaluation includes soil-structure interaction (SSI) analysis to develop the in-structure response spectra (ISRS). The submittal stated that the structures are founded on soil and the effects of SSI dominated the structural response. Three-dimensional (3D) finite element models were developed for the Auxiliary Building and Turbine Building/Screenhouse Buildings and a LMSM was used for the Containment Building. The foundation of the Containment Building is embedded, whereas the foundations of the Auxiliary Building and the Turbine Building were considered to be surface founded in the structural models. The horizontal and vertical response cutoff frequency for the Containment Building was 40 Hertz (Hz). For the Auxiliary Building and the Turbine Building/Screenhouse, the horizontal cutoff frequency was 20 Hz and the vertical cutoff frequency was 40 Hz.

To account for the uncertainty of the SSI, site-specific best estimate (BE), upper bound (UB), and lower bound (LB) soil profiles were considered in accordance with ASCE 4-16 (ASCE, 2017). The structural models were based on nominal stiffness parameters. In response to a staff audit question, the licensee stated that variation in stiffness was addressed in several ways. Sensitivity studies were performed for the Containment and Turbine Buildings where variation in stiffness and damping were considered using cracked and uncracked concrete properties. This resulted in some changes in the modal frequencies, but not significant changes in the magnitude of the response. For the Auxiliary Building, the licensee stated that the model was modified with updated demand and significant changes in the mass and stiffness; however, this resulted in only small changes in frequencies and magnitude of the response. In addition, the frequency range of interest was defined over a broad range to capture all peaks after frequency shifts. The NRC staff found that the licensee structural models address the concerns about variation in structural stiffness in F&O 28-11 (regarding SFR-B4).

The ground response spectra, called Review Level earthquake (RLE), used for the structural response of each building is equivalent to 0.8 times the uniform hazard spectra at 1.0E-05 /yr probability of exceedance. The licensee developed a set of synthetic time histories for the response analysis of each structure spectrally matched to the RLE. Each set consists of three time histories (two horizontal and one vertical). The submittal stated that the acceptability of the single set of three time histories (vertical and two horizontal components) was verified by performing a sensitivity study where the synthetic time history was compared to five new time history sets using real earthquake records as seeds, with no significant difference. The adequacy of the synthetic time history representing the characteristic site-specific ground motion was accepted by the independent focused peer review in response to F&O 28-09 (SFR-B1). In response to F&O 28-13 (SFR-B4), the disposition stated that based on the review of power spectral density of the five time histories and the synthetic time history at the frequency range of interest for risk significant components, gaps in the energy are not significant, and therefore, the SPRA results are not expected to be impacted.

The deterministic SSI was used to estimate in-structure response spectra (ISRS) for systems and components. The ISRS for SSC locations were based on retrieved in-structure time histories from nodes at appropriate locations in the finite element models to produce spectral accelerations within the range of frequencies of interest. The bounding soil case was used for the 84 percent Non-Exceedance Probability demand. Average values of the UB, BE, and LB response were used for the median acceleration demands. For fragility estimate of equipment and systems, response spectra clipping process was applied based on the guidance of EPRI TR-103959 (EPRI, 1994).

Based on the NRC staff's review of information in the submittal and auditing of structural response information via the e-Portal, the staff finds the licensee's deterministic approach to evaluate structural response and ISRS to be appropriate. The deterministic structural analysis and approach to address variability in soil properties, are consistent with ASCE 4-16 (ASCE, 2017) and used industry standard software for structural modeling and SSI analyses.

All Peer Review findings related to all SRs under HLR-SFR-B were addressed. Resolution of some of the SFR-B F&Os are discussed in checklist topic 4.

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

Consequence(s): N/A

The NRC staff concludes:	
• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to all SRs under HLR-SFR-B in the Code Case Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings and observations 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 Standard's requirements.	Yes

• The response analysis accounts for uncertainties in accordance with the SPID guidance and the 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 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. Potential Staff Findings:	No
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.	N/A
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.	N/A
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.	N/A
D) The Standard has been followed.	N/A
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 addressed screening of SSCs. The submittal stated that capacity-based screening approach was not used to screen SSCs in the SPRA. Although inherently rugged components were identified based on the guidance of EPRI 1025287 and EPRI NP-6041-SL (EPRI, 1991), these components were not screened out and instead were included in the SPRA with conservative fragility parameters. The rugged components were not found to be risk significant based on the SPRA quantification results. The licensee evaluated the liquefaction potential at the plant site and liquefaction was screened out from the fragility analysis based on the liquefaction induced settlement calculations, which showed that the foundation integrity is not significantly impacted by	

differential settlement. There are no F&Os identified related to SFR-C1, SFR-C2 and SPR-B5. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes:	
• The peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to the SR requirements SFR-C1, SFR-C2, and SPR-B5 in the Code Case Standard, as well as to the requirements in the SPID.	N/A
• Although some peer review findings and observations 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.	N/A
• 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 Conservation Deterministic Failure Margin (CDFM)/Hybrid method was used for seismic fragility analysis. If no, See item C) below and next issue. If yes: Potential Staff Findings:	Yes
A) The recommendations in Section 6.4.1 of the SPID were followed appropriately for developing the CDFM High Confidence Low Probability of Failure 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:

In Section 4.4.2 of its submittal, the licensee stated that the seismic fragility evaluation was performed using the methods described in EPRI NP-6041-SL, EPRI 1019200 (EPRI, 2009) and EPRI TR-103959. As discussed in Section 4.4.1 of its submittal, the licensee used a screening process based on EPRI 1019200 and EPRI NP-6041-SL, where “screened-out” SSCs were assigned a generic fragility, and component-specific fragilities were developed for those that were “screened-in.” The process is typically used to minimize the number of components for which fragility calculations must be performed; however, all components were included in the SPRA, as stated in Section 4.4.1 of the submittal. Based on the seismic margin screening methodology discussed in EPRI 1019200, the licensee either screened-in or screened-out SSCs from the detailed fragility calculation by verifying during plant walkdowns whether the SSCs are qualified to EPRI NP-6041-SL, Table 2-3 for structures and Table 2-4 for the components. Several structural components, e.g., Containment Building, Auxiliary Building (concrete), and the foundations of the Screenhouse and Turbine Building, were screened out because they met the requirements in EPRI NP-6041-SL, while the Turbine Building, Control Room Celling, and masonry block walls were screened-in for detailed fragility calculations. For component fragilities, the licensee evaluated anchorage, functional, and seismic interaction failures. The licensee explained that if the fragility of a component was dominated by a single failure mode, only that failure mode was modeled in the SPRA. However, if multiple failure modes for a component had

similar fragilities, then all fragilities were considered in the SPRA modeling. Section 4.4.2 of the submittal explains that the hybrid method was used to develop the fragilities of the SSCs. The licensee used the EPRI NP-6041-SL methodology to calculate high confidence low probability of failure (HCLPF) using the conservative deterministic failure margin (CDFM) method. In accordance with the hybrid method, the median capacity was then evaluated using HCLPF and the generic variability and uncertainty values from Table 6-2 of the SPID. The submittal states that in using the initial fragility estimate, dominant contributors were identified based on quantification of the SPRA. The fragilities were further refined by more detailed CDFM or using the Separation of Variables (SOV) method on those sub-set of SSCs. The licensee's procedures for development of CDFM/Hybrid fragilities using EPRI NP-6041-SL, EPRI 1019200 and EPRI TR-103959 guidance is consistent with Section 6.4.1 of the SPID. There were no F&Os for this topic. 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 peer reviewers. The relevant peer review findings are those that relate to the requirements in the SPID. No requirements in the Code Case Standard specifically address this Topic.	N/A
• Although some peer review findings and observations 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)

<table>
<tr><td>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.

<u>Potential Staff Findings</u>:
The NRC staff review of the SPRA's fragility analysis of SSCs sensitive to high frequency seismic motion finds that the analysis is acceptable.

The flow chart in Figure 6-7 of the SPID was followed.

The flow chart was not followed 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:

In Section 4.1.2 of the submittal, the licensee stated that evaluation of high frequency relays was performed in accordance with SPID Section 6.4.2. Sections 4.4.2.2 and 5.1.5 of the D.C. Cook submittal discussed screening of chatter sensitive relays and contactors. A systematic process based on screening, SPRA quantification, and chatter analysis was used to identify risk significant relays. Using the audit review process, the NRC staff confirmed that the licensee provided adequate information on SSCs that are sensitive to high-frequency seismic motion.

There are no F&Os related to SFR-E5.

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 peer reviewers. The relevant peer review findings are those that relate to the SR requirement SFR-E5 in the Code Case Standard, as well as to the requirements in the SPID.</td><td>

N/A</td></tr>
</table>

• Although some peer review findings and observations 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 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 of these are used, the NRC reviewer should use the following elements of the checklist.	
i) <u>Circuit analysis</u>: The seismic relay-chatter analysis of some relays relies on circuit analysis to assure that safety is maintained. (A) If <u>no</u>, then (B) is moot. (B) If <u>yes:</u> <u>Potential Staff Finding</u>: The approach to circuit analysis for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
ii) <u>Operator actions</u>: The relay-chatter analysis of some relays relies on operator actions to assure that safety is maintained. (A) If <u>no</u>, then (B) is moot. (B) If <u>yes:</u> <u>Potential Staff Finding</u>: The approach to analyzing operator actions for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
Notes from staff reviewer: The D.C. Cook submittal stated in Section 4.1.2 that an extensive relay chatter evaluation was performed following the guidance of SPID Section 6.4.2, ASME/ANS RA-Sb-2013, and EPRI 3002000709 (EPRI, 2013). During the audit review of supporting	

<table>
<tr><td colspan="2">documentation, the NRC staff confirmed that the circuit analysis was part of the chatter evaluation. Circuit analysis was done assuming the plant is running at 100 percent power and normal operational conditions. For systems whose primary function is mitigation after shutdown, the circuit analysis was done for shutdown conditions. The circuit analysis resulted in many relay chatter scenarios screened out from further evaluation based on no impact to the component function. The relays that were susceptible to chatter were further considered in the SPRA model. Fragility of risk significant relay groups, listed in Tables 5.4-2 and 5.5.-2, were evaluated using SOV. Using the audit review process, the NRC staff confirmed that the licensee provided adequate information regarding the circuit analysis and chatter evaluation summaries. The staff also confirmed that operator actions were appropriately considered for relay-chatter evaluation and screening.

There are no F&Os related to SPR-B6.

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

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes that:</td><td></td></tr>
<tr><td>• the peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The relevant peer review findings are those that relate to SR requirement SPR-B6 in the Code Case Standard, as well as to the requirements in the SPID.</td><td>N/A</td></tr>
<tr><td>• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.</td><td>N/A</td></tr>
<tr><td>• the licensee’s analysis of seismic relay-chatter effects meets the intent of the SPID guidance.</td><td>Yes</td></tr>
<tr><td>• 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.</td><td>N/A</td></tr>
</table>

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 separation of variables (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 separation-of-variables 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 6.4.3 of the SPID provides the guidance for selecting SSCs for refined fragility analysis. This section refers to NP-6041-SL for the threshold to exclude SSCs from the SPRA and only if the SSC does not affect the SCDF risk value or insights. Section 4.4.2 of the SPRA submittal explains that representative fragility values were developed for all SEL equipment and structures using the CDFM approach. Section 4.4.2 states that the SOV method was used for risk significant SSCs. The licensee states in Section 4.4.2.1 that the screening criteria of EPRI NP-6041-SL, Revision 1 was used to screen structures from further fragility analysis. The NRC staff's review confirmed that the SOV method was used to evaluate the fragilities for a limited number of SSCs (relay groups) and the licensee's use of the SOV fragility methodology followed the guidance in EPRI TR-103959, which is recommended in the SPID. Tables 5.4-2 and 5.5-2 of the SPRA submittal provide a listing of the risk-significant SSCs for Unit 1 SCDF and SLERF, respectively, and the method used to develop the fragility for each SSC. These tables show that the fragilities for just two of eight risk significant SSCs for SCDF and 4 of 20 risk significant SSCs for SLERF were developed using the SOV method. Regarding risk-significant component SSCs, the licensee's resolution to SPRA F&Os 22-2 and 22-5 (regarding supporting requirements SFR-E2 and SFR-E3) resulted in updated fragilities for many of the risk-significant SSCs.	

<table>
<tr><td colspan="2">
However, the F&Os remain open pending the results of sensitivity studies related to determining the impact for the other risk significant SSCs that were not refined. One sensitivity study, described in the SPRA submittal as Case 14c, increased the median capacity for these SSCs by 50 percent. The results of this sensitivity study show a reduction in SCDF and SLERF, as expected. Further, the sensitivity results showed that the number of risk-significant SSCs decreased and the F-V importance measure for some SSCs increased. However, the resultant risk values did not impact the results of this submittal.

The SPRA submittal did not provide importance analysis results for Unit 2. The licensee states in Sections 5.4 and 5.5 of the submittal that, while there are small numerical differences between the two units, the risk insights from Unit 1 are applicable to Unit 2. The licensee described the modeling differences between Units 1 and 2 and explained that Unit 2 SCDF and SLERF F-V results are within 2 and 4 percent, respectively, of the Unit 1 values. However, during the audit the licensee identified the Unit 2 SCDF and SLERF risk significant SSCs. The NRC staff determined that the Unit 2 results did not change the results of the submittal.

Accordingly, the NRC staff concluded that the licensee’s approach to achieve more detailed fragility analyses for dominant risk contributors using the SOV approach or a more refined CDFM approach is reasonable for the staff’s decision on this submittal.

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

Consequence(s): N/A
</td></tr>
<tr><td>The NRC staff concludes:</td><td></td></tr>
<tr><td>• the peer review findings have been addressed and the analysis approach has been accepted by the peer reviewers. The relevant peer review findings are those that relate to SFR-E3 and the requirements in the SPID. No requirements in the Standard specifically address this Topic.</td><td>Yes</td></tr>
<tr><td>• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.</td><td>N/A</td></tr>
<tr><td>• the licensee’s method for selecting the “dominant risk contributors” for further seismic fragilities analysis using the separation-of-variables methodology meets the intent of the SPID guidance.</td><td>Yes</td></tr>
<tr><td>• the licensee’s method for selecting the “dominant risk contributors” for further seismic fragilities analysis using the separation-of-variables methodology does not meet the intent of the SPID guidance, but is acceptable on another justified basis.</td><td>N/A</td></tr>
</table>

TOPIC 13: Evaluation of LERF (SPID Section 6.5.1)

The NRC staff review of the SPRA's analysis of LERF finds an acceptable demonstration of its adequacy.	Yes
<u>Potential Staff Findings</u>: A) The analysis follows each of the elements of guidance for LERF analysis in Section 6.5.1 of the SPID, including in Table 6-3.	Yes
B) The LERF analysis does not follow the guidance in Table 6-3 but the analysis is acceptable on another justified basis.	N/A
Notes from staff reviewer: Section 4.1.1 of the submittal describes the development of a SEL, including identifying SSCs associated with containment isolation and integrity, as well as seismic-induced failures that lead to a large early release. Section 5.1 further states that the SPRA large early release sequences are based on those developed for the internal events PRA. Section 5.1.7 discusses the addition of containment isolation pathways applicable to seismic events and seismic-induced structure failures that contribute to LERF. Lastly, Appendix A of the submittal explains that both the SPRA and the internal events PRA were peer reviewed and most F&Os, including those against LERF supporting requirements, were closed using an NRC-accepted process. The staff confirmed F&O's that remained open or partially resolved have acceptable dispositions and do not impact the SPRA results for purposes of this submittal. 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 relevant peer review findings are those that relate to the SR requirements SPR-E1, E5, and E6 in the Code Case Standard, as well as to the requirements in the SPID.	Yes
• although some peer review findings and observations have not been resolved, the analysis is acceptable on another justified basis.	Yes

• the licensee's analysis of LERF meets the intent of the SPID guidance.	Yes
• the licensee's analysis of LERF 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. Potential Staff Findings: A) The analysis follows each of the elements of the peer review guidance in Section 6.7 of the SPID as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b). B) The composition of the peer review team meets the SPID guidance as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b). C) The peer reviewers focusing on seismic response and fragility analysis have successfully completed the Seismic Qualifications Utility Group (SQUG) training course or equivalent (see SPID Section 6.7). In what follows, a distinction is made between an "in-process" peer review and an "end-of-process" peer review of the completed SPRA report. 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 "in process" peer review 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. If no, go to (F). If yes, the "in process" peer review approach is acceptable. Go to (G).	Yes Yes Yes Yes N/A
E) The "end-of-process" peer review process followed the peer review guidance in the SPID (Section 6.7) as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b).	Yes

<table>
<tr><td>If <u>no</u>, go to (F).

If <u>yes</u>, 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 as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b) but is acceptable on another justified basis.

G) The licensee peer-review F&Os were satisfactorily resolved or were determined not to be significant to the SPRA conclusions for this review application.</td><td>

N/A

Yes</td></tr>
<tr><td colspan="2">Notes from staff reviewer:

Section 5.2 and Appendix A of the submittal describe the peer review process used to establish the technical adequacy of the SPRA. The SPRA peer review was conducted in November 2018 against the CC-II SRs of PRA Standard ASME/ANS RA-S Case 1 (ASME/ANS RA-S Case 1, 2017) and in accordance with the peer review characteristics and attributes described in NRC Regulatory Guide (RG) 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities.” The ASME/ANS RA-S Case 1 has been accepted by the NRC for use in regulatory applications, subject to certain NRC staff comments and proposed resolutions (NRC, 2018a). During the audit, the licensee confirmed that these NRC staff comments and proposed resolutions were considered during the SPRA peer review. The submittal explains that the peer review team utilized the peer review process for external events defined in NEI 12-13 (NEI, 2012). The use of NEI 12-13 has been accepted by the NRC, subject to certain NRC staff comments and proposed resolutions (NRC, 2018b, 2018c). During the audit, the licensee confirmed that these NRC staff comments and proposed resolutions were considered during the SPRA peer review.

The SPRA submittal provides the qualifications for each of the peer review team members and states that the peer reviewers were independent of the CNP PRA development. Concurrence on the assignment of capability categories to each SR was based on a consensus process involving all members of the review team. Two members focusing on review of the fragility analysis and who participated in the plant walkdown were stated to have SQUG training course or equivalent. The submittal does not indicate if the peer review team members focusing on review of the plant response model and who participated in the plant walkdown had SQUG training. However, the resumes for each of these peer reviewers, which were reviewed by the NRC staff during the audit, were shown to demonstrate significant PRA experience, which is judged by the NRC staff to satisfy the SPID SQUG training guidance for these members.

All elements of the SPRA were peer reviewed against the capability category II (CC-II) requirements SPRA standard. The submittal states that all but ten F&Os were closed during an F&O closure review using an NRC-accepted process, and that all SRs have been determined to meet the CC-II requirements. During the audit, the licensee</td></tr>
</table>

confirmed that the NRC's accepted process for closure of F&Os (NRC, 2017a, 2017b) was used, which included a self-assessment by the licensee as to whether each F&O disposition was a PRA maintenance or upgrade, and an assessment by the F&O closure team of concurrence or disagreement of this determination. The closure team concluded that all but three of the F&O dispositions were PRA maintenance and that these three dispositions incorporated use of a new methodology necessitating a focused-scope peer review of SRs SHA-I1, SHA-I2, and SFR-B3. Three F&Os were developed by the focused-scope peer review team, of which one was closed by the closure team. The 12 open F&Os and associated dispositions are provided in Table A.8-1 of the SPRA submittal. The NRC staff reviewed these dispositions and determined all but two (F&Os 22-2 and 22-5) were adequately dispositioned for this SPRA submittal. The impact of the dispositions to F&Os 22-2 and 22-5 on this SPRA submittal is addressed under Topic 12. Section 5.1 of the submittal states the internal events PRA (IEPRA) model-of-record as of April 6, 2018 was used as the basis for the development of the SPRA model. During the audit, the staff determined that the IEPRA (including internal flooding) was peer reviewed in September 2015 against the CC-II requirements of the PRA standard (ASME/ANS Addendum A, 2009) and RG 1.200, Revision 2. The licensee explained that this peer review utilized the NEI 05-04 peer review process and that focused-scope peer reviews of the IEPRA (Pre-initiator HRA and LERF) were conducted in June 2016 and August 2017, respectively, using Addendum A of the PRA Standard and RG 1.200, Revision 2. During the audit, the licensee provided the internal events F&Os and associated dispositions. The NRC staff's review of the dispositions of F&Os 2-10 and HR-B2-01, regarding pre-initiator Human Reliability Analysis (HRA), determined that several updates were made to the IEPRA HRA. During the audit, the licensee confirmed that all appropriate IEPRA pre-initiators were incorporated into the SPRA model used for the submittal. Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes: • the licensee's peer-review process meets the intent of the SPID guidance as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b). • the licensee's peer-review process does not meet the intent of the SPID guidance as supplemented by NRC staff comments in the NRC letter dated March 7, 2018 (NRC 2018a, 2018b), but is acceptable on another justified basis.	Yes N/A

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

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

The documentation should include all of the items of specific information contained in the 50.54(f) letter as described in Section 6.8 of the SPID.</td><td>Yes

Yes</td></tr>
<tr><td colspan="2">Notes from staff reviewer:

Tables 2-1 and 2-2 of the submittal provide a cross-reference of information required by the 10 CFR 50.54(f) letter 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.9 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 percent), mean, and the 95th percentiles). This information was used in the NRC staff's screening evaluation reported in Enclosure 2.

According to Section 5.1 and Appendix A.7 of the SPRA submittal, Diverse and Flexible Coping Strategies (FLEX) were not credited in the SPRA. Section 5.7.2 of the submittal lists the results of a sensitivity study crediting FLEX, which shows a negligible impact on the SCDF and SLERF.

Submittal Tables 5.4-1 and 5.5-1 summarize the top ten cutsets as documented in the CNP SPRA Quantification notebook. During the review it was noted that the multiplication of the basic event (BE) probabilities in the cutset did not match the cutset probability. During the audit, the licensee explained that certain BEs were complement events and represented success probabilities. However, the submittal shows failure probabilities and an additional calculation is required (i.e., success probability = 1 – failure probability) to determine the complement success probability. The NRC staff confirmed the cutset probabilities when using the success probabilities. This manual approach is similar to the one used in the ACUBE software to address minimum cut upper bound estimation and therefore does not impact the results of the submittal.

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

Consequence(s): N/A</td></tr>
<tr><td>The NRC staff concludes:

• The licensee's documentation meets the intent of the SPID guidance. The documentation requirements in the Code Case</td><td>Yes</td></tr>
</table>

Standard can be found in HLR-SHA-J, HLR-SFR-F, and HLR-SPR-F. • The licensee’s documentation does not meet the intent of the SPID guidance but is acceptable on another justified basis.	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.	 Yes Yes Yes N/A
Plant will: • complete modifications at CNP by September 9, 2022	
Notes from the Reviewer: Section 6 of the submittal states that the licensee plans to develop and implement plant modifications that will provide supplemental power to the containment distributed ignition system (DIS, or hydrogen igniters) for each unit to mitigate the loss of offsite power (LOOP), and that considerable SLERF reduction could be gained by this modification. During the audit, the licensee clarified that the modification will consist of a single commercial grade diesel generator for each unit that will provide alternating current (ac) power to both trains of the DIS for up to 12 hours without refueling. The operation of this system would be performed solely from the control room. The licensee stated that once the modification is completed, the CNP SPRA model will be updated to reflect this plant addition. The licensee submitted a regulatory commitment to implement the plant modifications to provide backup power to the containment DIS to mitigate LOOP (ADAMS Accession No. ML20206K894). According to the supplement, the modifications will provide a reduction in SLERF. The staff credited the proposed modifications being implemented in accordance with the licensee's regulatory commitment during its review and decisionmaking on this SPRA submittal. Refer to Enclosure 2 for additional details in the detailed screening evaluation. Deviation(s) or Deficiency(ies), and Resolution: None Consequence(s): N/A	
The NRC staff concludes that the licensee: • identified plant modifications necessary to achieve the appropriate risk profile • provided a schedule to implement the modifications (if any) with appropriate consideration of plant risk and outage scheduling	 Yes Yes

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.

ASME/ANS, 2017: Case 1 for Standard ASME/ANS RA-Sb-2013 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, 2017.

EPRI, 1991: "A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)," EPRI NP-6041-SL, Revision 1, Electric Power Research Institute, Palo Alto, CA, August 1991.

EPRI, 1994: "Methodology for Developing Seismic Fragilities", EPRI TR-103959, Electric Power Research Institute, Palo Alto, CA, June 1994.

EPRI, 2009: "Seismic Fragility Applications Guide Update," EPRI 1019200, Electric Power Research Institute, Palo Alto, CA, December 2009.

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) as endorsed by the NRC in a February 15, 2013, letter (ADAMS Accession No. ML12319A074).

EPRI, 2013: "Seismic Probabilistic Risk Assessment Implementation Guide," EPRI 3002000709, Electric Power Research Institute, Palo Alto, CA, December 2009.

NEI, 2012: NEI 12-13 "External Hazards PRA Peer Review Process Guidelines," Nuclear Energy Institute, August 2012.

NEI, 2017: "Final Revision of Appendix X to NEI 05-04/07-12/12-16, *Close-Out of Facts and Observations (F&Os)*," Nuclear Energy Institute, February 21, 2017 (ADAMS Accession No. ML17086A431).

NRC, 2012: "U.S. Nuclear Regulatory Commission Comments on NEI 12-13, 'External Hazards PRA Peer Review Process Guidelines' Dated August 2012," NRC letter to Nuclear Energy Institute, November 16, 2012 (ADAMS Accession No. ML12321A280).

NRC, 2017: "NRC Staff Review Guidance for Seismic PRA Submittals and Technical Review Checklist," February 10, 2017 (ADAMS Accession No. ML17041A342).

NRC, 2018: "US Nuclear Regulatory Commission Acceptance of ASME/ANS RA-S Case 1," NRC letter from Brian Thomas (NRC Standards Executive) to C.R. Grantom and R.J. Budnitz, March 12, 2018 (ADAMS Accession No. ML18017A963).

NRC, 2018a: "US Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13, "External Hazards PRA Peer Review Process Guidelines" (August 2102" NRC letter to Nuclear Energy Institute, March 7, 2018 (ADAMS Accession No. ML18025C025).

NRC, 2018b: "US Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13, "External Hazards PRA Peer Review Process Guidelines" (August 2102," tabular compilation of NRC staff comments, appended to (NRC, 2018a), (ADAMS Accession No. ML18025C022).

NRC Staff SPRA Submittal Detailed Screening Evaluation

<u>Introduction</u>

The Donald C. Cook Nuclear Plant Units 1 and 2 (CNP) Seismic Probabilistic Risk Assessment (SPRA) report (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19310D805) indicates that the Unit 1 mean seismic core damage frequency (SCDF) is 5.46E-05/reactor-year (/rx-yr) and seismic large early release frequency (SLERF) is 9.72E-06/rx-yr, while Unit 2 results are similar. The NRC staff assessed the risk significant results for both units. The NRC staff compared these values against the guidance in NRC staff memorandum dated August 29, 2017 (ADAMS Accession No. ML17146A200), 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" (hereafter referred to as the SPRA Screening Guidance), which establishes a process the NRC staff uses to develop a recommendation on whether the plant should move forward as a Group 1, 2, or 3 plant.[1]

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).

The NRC staff found that because the SCDF and SLERF for CNP were above the initial screening values of 1.0E-5/rx-yr and 1.0E-6/rx-yr, respectively, a detailed screening following the SPRA Screening Guidance was performed. The detailed screening shows that CNP should be considered a Group 1 plant because:

- Sufficient reductions in SCDF and SLERF cannot be achieved by potential modifications considered in this evaluation, other than the one identified by the licensee and included as a regulatory commitment, to constitute substantial safety improvements based upon importance measures, available information, and engineering judgement;

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[1] 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).

- 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

Indiana Michigan Power Company (I&M, the licensee), in performing its seismic analysis in response to the Near-Term Task Force Recommendation 2.1, and the NRC staff 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 CNP 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 licensee identified a plant modification at each unit to provide supplemental power to the containment distributed ignition system (DIS, or hydrogen igniters) to mitigate loss of offsite power (LOOP). This modification consists of a single commercial grade diesel generator for each unit to provide alternating current (ac) power to both trains of the DIS for up to 12 hours without refueling, with system operation performed solely from the control room. In its submittal, the licensee stated that this modification could reduce SLERF by 50 percent. The licensee also provided a regulatory commitment in its supplemental letter dated July 16, 2020 (ADAMS Accession No. ML20206K894), to implement these plant modifications by September 9, 2022. The NRC staff included the licensee's self-identified modifications as well as the regulatory commitment in its evaluation.

The detailed screening uses information provided in the CNP SPRA report, particularly the importance measures, SCDF, and 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 severe accident mitigation alternative (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 CNP 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 SAMA analysis provided in the *Application for Renewed Operating Licenses, Donald C. Cook Nuclear Plant Units 1 and 2*, dated October 31, 2003 (ADAMS Accession No. ML033070179), and associated supplements was used to calculate the RRW threshold, in addition to population dose-risk data provided in the *License Amendment Request to Revise Technical Specification Section 5.5.14, "Containment Leakage Rate Testing Program,"* dated March 7, 2014 (ADAMS Accession Nos. ML14071A435 and ML14071A436). For this analysis, the NRC staff determined the RRW threshold from the SCDF-based MACR to be 1.022. 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 SCDF and SLERF were also calculated by the NRC staff and ranged between 1.22 and 1.07.

Sections 5.4 and 5.5 of the CNP SPRA report included tables listing and describing the structures, systems, and components (SSCs) that are the most significant seismic failure contributors to Unit 1 SCDF and SLERF, respectively. Similar tables for Unit 2 SCDF and SLERF were made available for staff review during the audit. The licensee also provided similar tables during the audit 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 and the contribution to SCDF or SLERF of each contributor. The results are provided in Tables 1 and 2 for the SCDF contributors and Tables 3 and 4 for the SLERF contributors for Units 1 and 2, respectively. The listed seismic-induced failures that contribute to SCDF and SLERF have an RRW greater than about 1.020 for SCDF and 1.009 for SLERF. These tables provide the following information by column: (1) Description of the component, (2) Failure Mode, (3) RRW, and (4) maximum SCDF or SLERF reduction (MCR or MLR, respectively) from completely eliminating the failure.

No single SPRA model element or contributor exceeded the mean target RRW for SCDF, while one element or contributor exceeded the mean target RRW for SLERF. This element was seismically-induced structural failure of the Auxiliary Building (SC-SCIB-AB). The NRC staff experience from SAMA analyses is that the implementation cost of modifications to the Auxiliary Building sufficient to eliminate or substantially reduce the seismic risk from a seismically-induced structural failure is likely to substantially exceed the calculated MACR for this detailed screening.

In addition to failure of individual elements or contributors, the NRC staff 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. A review of the model elements in Tables 1 and 2 reveals that most modifications or sets of modifications to achieve a SCDF reduction of at least 1.0E-05/rx-yr or a SLERF reduction of at least 1.0E-06/rx-yr will have to mitigate or prevent multiple failure types (e.g., seismically-induced failures, random failures, and failure of operator actions) and/or failure modes (e.g., seismically-induced structural failures of multiple SSCs and seismically-induced functional failures of multiple SSCs).

The NRC staff's review does indicate that there is at least one potential combination of similar failures that, if eliminated, could achieve a risk reduction greater than a screening threshold (specifically, the five relay fragility groups D_1, D_2, B_4_U2, B_5_U2 and B_8_U2 in aggregate contribute about 1.56E-06/rx-yr to SLERF risk). However, given that the licensee has identified a plant modification to provide station blackout power to the containment hydrogen igniter system (DIS) that reduces SLERF (Sensitivity 1 discussed in Section 5.7 of the submittal), and because there could be multiple potential modifications required for achieving sufficient risk reduction from the five relay groups that may not be cost-justified, the staff concludes that it is unlikely to identify other cost-justified plant improvements that could substantially reduce or eliminate the risk of failure of the SSC combinations.

Given that 1) no cost-justified plant improvements were identified to address single failures to achieve a SCDF or SLERF risk reduction of at least 1.0E-05 per year and 1.0E-06 per year, respectively, 2) multiple potential modifications unlikely to be cost-justified would be necessary to achieve substantial safety improvement, and 3) the licensee plans to implement plant modifications that would reduce SLERF, the NRC staff did not pursue additional potential improvements.

Based on the analysis described above, the NRC staff concludes that modifications, other than those identified by the licensee and submitted as a regulatory commitment (ADAMS Accession No. ML20206K894), are not 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 CNP 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.

Based on the available information described above and engineering judgement, the NRC staff concluded that there were no further potential improvements to containment performance that would result in 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 CNP to understand previous work done to identify substantial safety improvements and to further inform this review. No other potential improvements were found based on this review.

<u>Conclusion</u>

Based on the analysis of the submittal and supplemental information, the NRC staff concludes that no modifications, except for the plant modifications to provide station blackout power to the containment hydrogen igniter system (DIS) identified by the licensee and included in a regulatory commitment, 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 an additional modification that would result in a substantial safety improvement.

Table 1. Importance Analysis Results of Unit 1 Top Contributors to Seismic CDF

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
Seismic Loss of Offsite Power	Functional	1.166	7.75E-06
Seismic Induced Very Small LOCA	Functional	1.057	2.95E-06
Seismic Failure of Relay Group D_1	Relay Chatter	1.044	2.29E-06
Seismic Failure of Relay Group D_2	Relay Chatter	1.036	1.91E-06
Failure of Panel A11 due to Control Room Ceiling Failure	Anchorage	1.028	1.47E-06
Seismic Failure of Train CD Battery Rack	Anchorage	1.024	1.26E-06
Seismic Failure of Supplemental Diesel Generator Components	Functional	1.021	1.15E-06
Turbine Building Collapse	Structural Collapse	1.020	1.09E-06

Table 2. Importance Analysis Results of Unit 2 Top Contributors to Seismic CDF

Description	Failure Mode	RRW	MCR (/yr)
Seismically-failed SSCs			
Seismic Loss of Offsite Power	Functional	1.139	6.66E-06
Seismic Induced Very Small LOCA	Functional	1.076	3.88E-06
Seismic Failure of Relay Group D_1	Relay Chatter	1.042	2.18E-06
Seismic Failure of Relay Group D_2	Relay Chatter	1.041	2.13E-06
Failure of Panel A11 due to Control Room Ceiling Failure	Anchorage	1.029	1.53E-06
Fragility Group Relay_B_8_U2	Relay Chatter	1.029	1.26E-06

Description	Failure Mode	RRW	MCR (/yr)
Seismic Failure of Train CD Battery Rack	Anchorage	1.024	1.53E-06
Turbine Building Collapse	Structural Collapse	1.021	1.15E-06

Table 3. Importance Analysis Results of Unit 1 Top Contributors to Seismic LERF

Description	Failure Mode	RRW	MLR (/yr)
Seismically-failed SSCs			
SC-I Building Auxiliary Building	Structural	1.186	1.53E-06
Loss of Offsite Power	Structural	1.106	9.33E-07
Relay Fragility Group Relay_D_1	Chatter	1.054	4.96E-07
Relay Fragility Group Relay_D_2	Chatter	1.048	4.47E-07
Supplemental Diesel Generator System Components	Structural	1.040	3.69E-07
Seismic-Induced Fire Originating from Boiler	Structural	1.037	3.50E-07
Medium LOCA	Functional	1.037	3.50E-07
Relay Panels in Control Room	Anchorage	1.030	2.82E-07
Seismic-Induced Fire Originating from Main Turbine Oil System - Limited	Structural	1.029	2.72E-07
Fragility Group Relay_B_5_U1	Chatter	1.027	2.53E-07
Turbine Building Collapse	Structural	1.025	2.33E-07
Seismic Induced Very Small LOCA	Functional	1.015	1.46E-07
Fragility Group Relay_B_4_U1	Chatter	1.014	1.36E-07
SC-I Building - Containment Building	Structural	1.014	1.36E-07
Seismic-Induced Flood from Other Non-Safety-Related Systems	Structural	1.012	1.17E-07
Seismic-Induced Fire Originating from Main Feed Pump - 10% oil	Structural	1.011	1.07E-07
Control Room Ceiling Section 3	Structural	1.010	9.72E-08
Control Room Ceiling Section 2	Structural	1.009	8.75E-08
Plant Battery CD (Room 201)	Anchorage	1.009	8.75E-08

Fragility Group Relay_B_2_U1	Chatter	1.009	8.75E-08

Table 4. Importance Analysis Results of Unit 2 Top Contributors to Seismic LERF

Description	**Failure Mode**	**RRW**	**MLR (/yr)**
Seismically-failed SSCs			
SC-I Building Auxiliary Building	Structural	1.199	1.61E-06
Loss of Offsite Power	Structural	1.100	8.85E-07
Relay Fragility Group Relay_D_2	Chatter	1.055	5.05E-07
Relay Fragility Group Relay_D_1	Chatter	1.054	4.96E-07
Medium LOCA	Functional	1.040	3.69E-07
Supplemental Diesel Generator System Components	Structural	1.035	3.30E-07
Relay Panels in Control Room	Anchorage	1.034	3.21E-07
Fragility Group Relay_B_4_U2	Chatter	1.027	2.53E-07
Turbine Building Collapse	Structural	1.025	2.33E-07
Seismic Induced Very Small LOCA	Functional	1.024	2.24E-07
Fragility Group Relay_B_8_U2	Chatter	1.022	2.14E-07
SC-I Building - Containment Building	Structural	1.015	1.46E-07
Seismic-Induced Fire Originating from Main Feed Pump - 10% oil	Structural	1.015	1.46E-07
250 VDC Distribution Panel (Unit 2, Room 123)	Functional	1.013	1.26E-07
Control Room Ceiling Section 5	Structural	1.012	1.17E-07
Control Room Ceiling Section 2	Structural	1.011	1.07E-07
Fragility Group Relay_B_5_U2	Chatter	1.010	9.72E-08

AUDIT SUMMARY BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

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-0017)

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 10 CFR Part 50, Section 50.54(f). The list of applicable licensees in Enclosure 1 to the July 6, 2017, letter included Indian Michigan Power Company as the licensee for Donald C. Cook Nuclear Plant, Units 1 and 2 (CNP).

REGULATORY AUDIT SCOPE AND METHODOLOGY

The areas of focus for the regulatory audit are the information contained in the SPRA submittal 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 staff's clarification questions dated March 25, 2020, and April 27, 2020 (ADAMS Accession

Nos. ML20232B443 and ML20232B456, respectively), were sent to the licensee to support the audit.

The licensee provided clarifying information in the following areas:

- Information regarding fragility sensitivity analyses in response to Facts and Observations (F&Os) 22-2 and 22-5.
- Discussion of risk significant structures, systems, and components (SSCs) for Unit 2, as compared to those for Unit 1 documented in the submittal.
- Discussion describing the relationship between the CNP Internal Events PRA and the Seismic PRA – specifically the use of internal events pre-initiator Human Failure Events in the SPRA.
- Discussion of the details and implementation of the planned plant modifications to mitigate the loss of offsite power event.
- Clarification of the documentation of various details of the SPRA model logic, quantification, and particular cutsets.
- Discussion of the use of scaling or screening capacity approach used to evaluate the fragility of NSSS components.
- Discussion of the basis for the review level earthquake hazard level used in the SPRA.
- Clarification of certain details regarding the structural models used in the fragility assessment and the basis for meeting the relevant SPID criteria.
- Clarification of the licensee's disposition of some F&Os that remained open and the justification for their impact on the SPRA results.

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

<u>DOCUMENTS AUDITED</u>

- PWROG-18062-P, Revision 0, "Peer Review of the D.C. Cook Nuclear Power Plant, Units 1&2, Seismic Probabilistic Risk Assessment," dated January 2019.
- Report AEPDCC~00058-REPT-001, "SPRA Fact and Observation Independent Assessment and Focused-Scope Peer Review, Donald C. Cook Nuclear Plant Units 1 and 2," Revision 0.
- CNP Document 15C4313-RPT-002, "Fragility Analysis Plan for Cook Nuclear Plant Unit 1 & Unit 2 Seismic PRA," Revision 3.
- CNP document 15C4313-RPT-003, "Summary of Building Response Analysis for Cook

Nuclear Plant (CNP) Unit 1 & Unit 2 SPRA," Revision 4.

- CNP document 15C4313-CAL-010 "Response Analysis of Auxiliary Building," Revision 3.
- CNP document PRA-NB-SPRA-QU, Revision 0, "SPRA Model Quantification Notebook".
- CNP document 15C4313-CAL-020 "Turbine Building Seismic Interaction Evaluation," Revision 1.
- CNP document 15C4313-CAL-021 "Detailed Relay HCLPF Seismic Capacities," Revision 3.
- CNP document 15C4313-CAL-026 "HCLPF Seismic Capacities of Battery Racks for SPRA," Revision 1.
- CNP Document 15C4313-RPT-001, "Civil Structures Screening Evaluation for Cook Nuclear Plant Unit 1 & Unit 2 Seismic PRA," Revision 2.
- CNP Document 15C4313-RPT-007, "Seismic Fragility Analysis for Cook Nuclear Plant (CNP) Unit 1 & Unit 2 Structures, Systems, and Components," Revision 3.
- CNP Document PRA-SPRA-RELAY-EVAL "Seismic Probability Risk Assessment Relay Chatter Impact Evaluation," Revision 0. October 10, 2018
- CNP Document PRA-NB-SPRA-HRA, "Donald C. Cook Nuclear Plant (DC Cook) Seismic Human Reliability Analysis," Revision 0.
- CNP Document PRA-NB-SPRA-MDL, "Seismic PRA Modeling Notebook," Revision 1.
- CNP Document PRA-2015-PEER-REVIEW, "2015 Internal Events Peer Review Disposition Report," Revision 0.
- PWROG-15076-P, Revision 0, "Peer Review of the D.C. Cook Nuclear Plant Internal Events Probabilistic Risk Assessment," dated September 2015.
- Report 1BTI1V001-RPT-01, "D.C. Cook Focused Scope Peer Review – Pre-Initiator HRA," Revision 0.
- Report AEPDCC~00051-REPT-001, "Cook Nuclear Plant Seismic PRA Hydrogen Findings Closure Review," Revision 0.
- Report AEPDCC~00036-REPT-001, "Cook Nuclear Plant Evaluation of Detailed Hydrogen Analyses (01V015-RPT-01) Against the LERF Supporting Requirements of the ASME PRA Standard (2013)," Revision 0.

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 generic audit plan dated July 6, 2017.

AUDIT CONCLUSION

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