20.0 REQUIREMENTS RESULTING FROM FUKUSHIMA NEAR-TERM TASK FORCE RECOMMENDATIONS

This chapter addresses the requirements resulting from the Fukushima Near-Term Task Force (NTTF) recommendations that are applicable to the Levy Nuclear Plant (LNP) Units 1 and 2 Combined License (COL). The applicable recommendations address four topics: a reevaluation of the seismic hazard (related to Recommendation 2.1), mitigation strategies for beyond-design-basis external events (related to Recommendation 4.2), spent fuel pool (SFP) instrumentation (related to Recommendation 7.1), and emergency preparedness staffing and communications (related to Recommendation 9.3).

Background

In response to the events at Fukushima resulting from the March 11, 2011, Great Tohoku earthquake and tsunami in Japan, the U.S. Nuclear Regulatory Commission (NRC) established the NTTF to conduct a systematic and methodical review of NRC processes and regulations to determine whether the agency should make additional improvements to its regulatory system and to make recommendations to the Commission for policy direction. In July 2011, the NTTF issued a 90-day report, SECY-11-0093 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11186A950), "Near Term Report and Recommendations for Agency Actions Following the Events in Japan," identifying 12 recommendations. On September 9, 2011, in SECY-11-0124, "Recommended Actions to Be Taken Without Delay From NTTF Report," (ADAMS Accession No. ML11245A144) the staff provided to the Commission for its consideration NTTF recommendations that can and, in the staff's judgment, should be initiated, in part or in whole, without delay. In SECY-11-0124 the staff identified and concluded that the following subset of actions had the greatest potential for safety improvement in the near-term:

1. Recommendation 2.1: Seismic and Flood Hazard Reevaluations

2. Recommendation 2.3: Seismic and Flood Walkdowns

3. Recommendation 4.1: Station Blackout Regulatory Actions

4. Recommendation 4.2: Equipment covered under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(hh)(2)

5. Recommendation 5.1: Reliable Hardened Vents for Mark I Containments

6. Recommendation 8: Strengthening and Integration of Emergency Operating Procedures, Severe Accidents Management Guidelines, and Extensive Damage Mitigation Guidelines

7. Recommendation 9.3: Emergency Preparedness Regulatory Actions (staffing and communications).

On October 3, 2011, in SECY-11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned," (ADAMS Accession No. ML11272A203) the staff identified two actions in addition to the actions discussed in SECY-11-0124 which had the greatest potential for safety improvement in the near-term. The additional actions are:

1. Inclusion of Mark II containments in the staff's recommendation for reliable hardened vents associated with NTTF Recommendation 5.1

2. The implementation of SFP instrumentation proposed in Recommendation 7.1

The staff also prioritized the NTTF recommendations into Tier 1, Tier 2, and Tier 3, where the recommendations in Tier 1 represent those that the staff determined should be started without unnecessary delay, while recommendations in Tier 2 are those that could not be initiated in the near term, and recommendations in Tier 3 require further study to support regulatory action.

On February 17, 2012, in SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," (ADAMS Accession No. ML12039A103) the staff provided the Commission with proposed orders and requests for information to be issued to all power reactor licensees and holders of construction permits.

On March 9, 2012, the Commission then approved issuance of the proposed orders with some modifications in the staff requirements memorandum (SRM) to SECY-12-0025. As set forth in the Orders in SRM-SECY-12-0025, additional requirements are needed to provide adequate protection to public health and safety or to significantly enhance the protection of public health and safety. In accordance with its statutory authority under Section 161 of the Atomic Energy Act of 1954, as amended (the Act), the Commission may impose these requirements.

On March 12, 2012, the NRC issued Orders EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" and EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" to the appropriate licensees and permit holders (ADAMS Accession Nos. ML12054A679 and ML12054A735).

The staff also issued the request for information pursuant to 50.54(f) regarding Recommendations 2.1, 2.3 and 9.3, as described in SECY-12-0025, to the appropriate licensees and permit holders in letters dated March 12, 2012 (ADAMS Accession No. ML12053A340).

The following Tier 1 recommendations in SECY-11-0137 as addressed in SECY-12-0025 were considered in determining those that are applicable to the LNP COL review:

1. Recommendation 2.1: Seismic and Flood Hazard Reevaluations

2. Recommendation 2.3: Seismic and Flood Walkdowns

3. Recommendation 4.1: Station Blackout Regulatory Actions

4. Recommendation 4.2: Equipment covered under 10 CFR 50.54(hh)(2)

5. Recommendation 5.1: Reliable Hardened Vents for Mark I and Mark II Containments

6. Recommendation 7.1: Spent Fuel Pool Instrumentation

7. Recommendation 8: Strengthening and Integration of Emergency Operating Procedures, Severe Accidents Management Guidelines, and Extensive Damage Mitigation Guidelines

8. Recommendation 9.3: Emergency Preparedness Regulatory Actions (staffing and communications)

Staff determined that the following four recommendations were applicable and should be addressed by the LNP COL applicant:[1]

1. Recommendation 2.1: Seismic reevaluations - Order licensees to reevaluate the seismic hazards at their sites against current NRC requirements and guidance, and if necessary, update the design basis and structures, systems, and components important to safety to protect against the updated hazards.

2. Recommendation 4.2: Equipment covered under 10 CFR 50.54(hh)(2) - Order licensees to provide reasonable protection for equipment currently provided pursuant to 10 CFR 50.54(hh)(2) from the effects of design-basis external events and to add equipment as needed to address multiunit events while other requirements are being revised and implemented.

3. Recommendation 7.1: Spent fuel pool instrumentation - Order licensees to provide reliable spent fuel pool level instrumentation.

4. Recommendation 9.3: Emergency preparedness regulatory actions (staffing and communications) - Order licensees to do the following until rulemaking is complete:

[1] The applicant, Duke Energy Florida, LLC., was formerly identified as Duke Energy Florida, Inc. and Progress Energy Florida, Inc. In a letter dated April 15, 2013, Progress Energy Florida notified the NRC that its name was changing to Duke Energy Florida, Inc. effective April 29, 2013. The name changes and a 2012 corporate merger between Duke Energy and Progress Energy are described in Chapter 1 of the SER. Because a portion of the review described in this chapter was completed prior to the name change, the NRC staff did not change references to “Progress Energy Florida,” or “PEF,” to “Duke Energy Florida,” or “DEF.”

- Determine and implement the required staff to fill all necessary positions for response to a multi-unit event.
- Provide a means to power communications equipment needed to communicate onsite (e.g., radios for response teams and between facilities) and offsite (e.g., cellular telephones and satellite telephones) during a prolonged station blackout.

The staff determined that the remaining Tier 1 recommendations did not need to be further considered in the LNP COL review. The applicant evaluated the flood hazard using the current guidance and methodologies, and staff has, therefore, determined that the flood reevaluation portion of Recommendation 2.1 has already been addressed. Therefore, there are no additional requirements to address Recommendation 2.1 for flooding reevaluation applicable for the LNP COL application. Additionally, the staff determined that Recommendation 2.3 was not applicable to the LNP COL because the plant is not yet constructed, and Recommendation 5.1 was not applicable because it applied to boiling water reactor (BWR) type plant designs with Mark I and Mark II Containments. Recommendations 4.1 and 8 did not need to be further considered because SECY-11-0137 and its associated SRM direct that regulatory action associated with them be initiated through rulemaking.

In SECY-12-0025, the staff stated that it would request all COL applicants to provide the information required by the orders and request for information letters through the review process. Accordingly, for the LNP COL application, the staff issued request for additional information (RAI) Letter No. 108 (ADAMS Accession No. ML120550146), dated March 15, 2012, related to Implementation of Fukushima Near-Term Task Force Recommendations pertaining to seismic hazard reevaluation, mitigation strategies for beyond-design-basis external events, spent fuel pool instrumentation, and emergency preparedness based on Recommendations 2.1, 4.2, 7.1, and 9.3, as modified by SRM-SECY-12-0025. The following sections of this chapter present the staff's safety evaluation related to these areas.

20.1 Recommendation 2.1, Seismic Hazard Reevaluation

20.1.1 Introduction

SECY-12-0025, Enclosure 7, Attachment 1 to Seismic Enclosure 1 (ADAMS Accession No. ML12039A103), related to seismic hazard reevaluation, specifies the use of NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," in a site probabilistic seismic hazard analysis (PSHA) and describes an updated cumulative absolute velocity (CAV) filter methodology. The NRC staff issued NUREG-2115 in January 2012 as a replacement to the Electric Power Research Institute-Seismic Owners Group (EPRI-SOG) (EPRI 1986, 1989) and the Lawrence Livermore National Laboratory (LLNL) (Bernreuter et al., 1989) seismic source models for the central and eastern United States (CEUS). NUREG-2115 describes the implementation of a Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 assessment process for developing the new regional seismic source characterization (SSC) model for the CEUS. Consistent with SECY-12-0025, as well as the need to consider the latest available information in the PSHA for the LNP site, the NRC staff requested that the applicant evaluate the seismic hazards at the LNP site against current NRC requirements and guidance.

Safety Evaluation Report (SER) Section 20.1 provides the staff's evaluation of the seismic hazards at the LNP site, performed in accordance with SECY-12-0025. The information discussed in SER Section 20.1 supports the staff's evaluation in SER Sections 2.5.2 "Vibratory Ground Motion," 2.5.4 "Stability of Subsurface Materials and Foundations," 3.7 "Seismic Design," and 19.55.6.3 "Site-Specific Seismic Margin Analysis."

20.1.1.1 Summary of CEUS SSC Model

In this section, the staff summarizes the CEUS SSC model, which the applicant used for its seismic hazard reevaluation in response to RAI Letter No. 108 (ADAMS Accession No. ML120550146). This summary focuses on the parts of the CEUS SSC model that are applicable to the LNP site seismic hazard and provides background and a framework for the staff's technical evaluation of the applicant's seismic hazard reevaluation in SER Section 20.1.4. The specific deviations taken by the applicant during model implementation from the as-is model published in NUREG-2115 are described and evaluated in SER Section 20.1.4.

On January 31, 2012, the NRC, U.S. Department of Energy (DOE), and EPRI issued a new SSC model and report for use in seismic hazard assessments for nuclear facilities in the CEUS. This cooperative project replaces seismic source models developed in the 1980s by the EPRI-SOG (EPRI 1986, 1989) and the LLNL (Bernreuter et al., 1989).

The new model addresses the need for an up-to-date regional SSC model for the CEUS that includes: (1) a full assessment and incorporation of uncertainties, (2) a range of diverse technical interpretations from the informed scientific community, (3) an up-to-date earthquake database, (4) proper and appropriate documentation, and (5) comprehensive, participatory peer review. The cooperative project for this new model was conducted using processes described in the SSHAC guidance NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts." The model was developed using a SSHAC Level 3 assessment process, with the goal of representing the center, body, and range of technically defensible interpretations of the available data, models, and methods.

The CEUS SSC model is a new seismic source model for the CEUS, the broad region of the United States east of the Rocky Mountains. The CEUS SSC study region is shown in SER Figure 20.1-1. The CEUS SSC Project resulted in products and methodological improvements that have value for future users as follows: (1) data evaluation and data summary tables that identify all the data considered by the project team and that indicate the team's views of the quality of the data and degree of reliance placed on any given data set, (2) database of geologic, geophysical, and seismological data, (3) earthquake catalog with uniform moment magnitudes (M), (4) updated paleoseismicity data and guidance, and (5) recommendations for future applications of the SSC model. For purposes of demonstrating the CEUS SSC model, the project also included sample calculations at the seven sites identified in SER Figure 20.1-1.

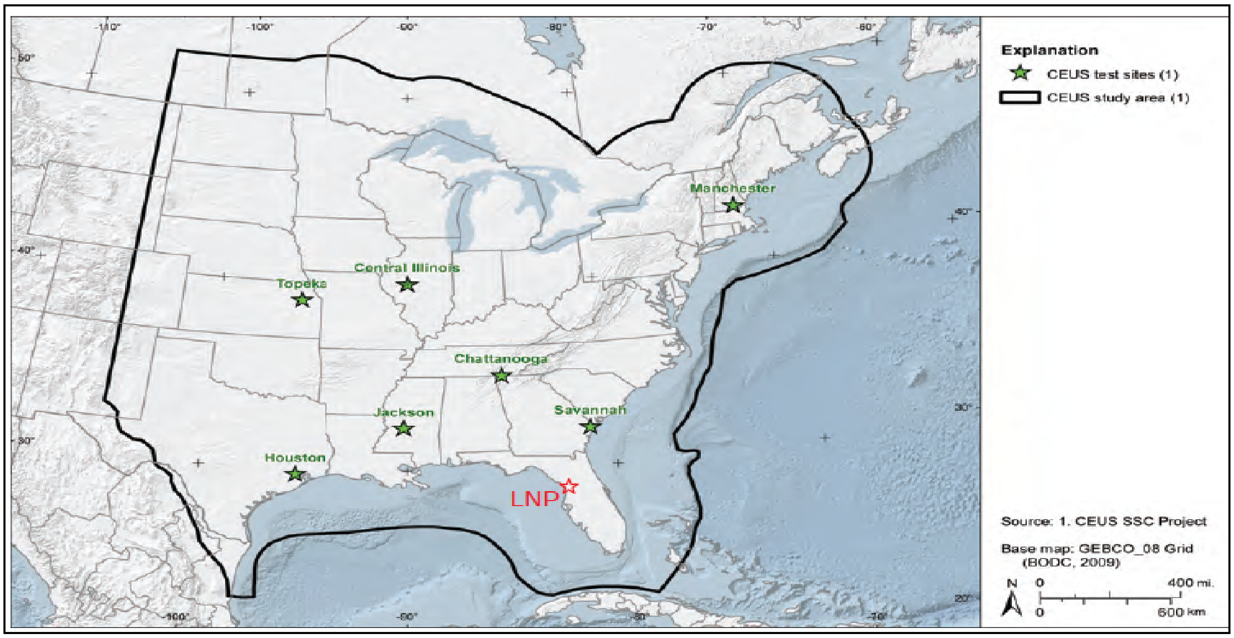

Figure 20.1-1. CEUS SSC Model Study Region (black line), the Location of the Seven Test Sites (green stars), and the LNP Site (red star)
(FSAR Figure 2.5.2-324 and NUREG-2115 Figure 8.1-1)

The EPRI-SOG model was used by the applicant in its final safety analysis report (FSAR) evaluation of vibratory ground motion in FSAR Section 2.5.2. In accordance with Regulatory Guide (RG) 1.208, "A Performance-Based Approach To Define the Site-Specific Earthquake Ground Motion," recent licensing applications for nuclear facilities submitted to the NRC—including the LNP application—used the EPRI-SOG model as a starting point and updated the model, as appropriate, on a site-specific basis for the application's PSHA. While the applicants updated the EPRI-SOG model on a site-specific basis, the NRC has not conducted a systematic update of the full model in over 20 years. The project to develop the CEUS SSC model created an up-to-date CEUS seismic hazard model that took into account data used to develop the previous two models, new data and information developed in the interim years, and other information and hazard analyses that were developed as part of licensing actions for proposed and existing nuclear power facilities. Lastly, the CEUS SSC model contains updated methods for evaluating the data and quantifying uncertainties within the PSHA model. Because the LNP applicant submitted its COL application to the NRC for review in July 2008, before the CEUS SSC model was published in NUREG-2115 in January 2012, the applicant used the EPRI-SOG model in its initial application and later updated the application to include a sensitivity evaluation of the seismic hazard at the LNP site using the newer CEUS SSC model.

The CEUS SSC model consists of three models of seismic sources – the Mmax zones model, the seismotectonic zones model, and the repeated large magnitude earthquake (RLME) sources model. First, the CEUS SSC model characterizes the CEUS study area using two conceptual source models that assess the spatial and temporal distribution of future seismicity. These are the Mmax zones model and the seismotectonic zones model, which represent the

background or distributed seismicity in the CEUS using two different approaches of characterizing future earthquakes.

The Mmax zones model is based on average or "default" characteristics that are representative of large areas of the CEUS or the entire study area, such that Mmax zones cover large areas and are based on historical seismicity and broad-scale geologic and tectonic data.

The seismotectonic zones model includes information that allows for an assessment of spatial variations of future earthquake characteristics at a finer scale than the Mmax zones model. The seismotectonic zones model uses historical seismicity and regional-scale geologic and tectonic data to characterize seismic sources zones.

Finally, the RLME sources model is the third type of seismic source. The RLME sources model is not based on distributed seismicity in an areal source like the Mmax and seismotectonic zones models, but mainly on earthquake recurrence paleoseismic data and, as its name suggests, it represents the sources on which repeated large magnitude earthquakes occur.

SER Figure 20.1-2 shows where the three types of source zones appear on the CEUS SSC model master logic tree. As described in NUREG-2115, the RLME sources are characterized by the historical and paleoseismic records and are defined as having experienced two or more earthquakes having a moment magnitude of at least M 6.5. The geographic locations of the RLME sources are shown on SER Figure 20.1-3.

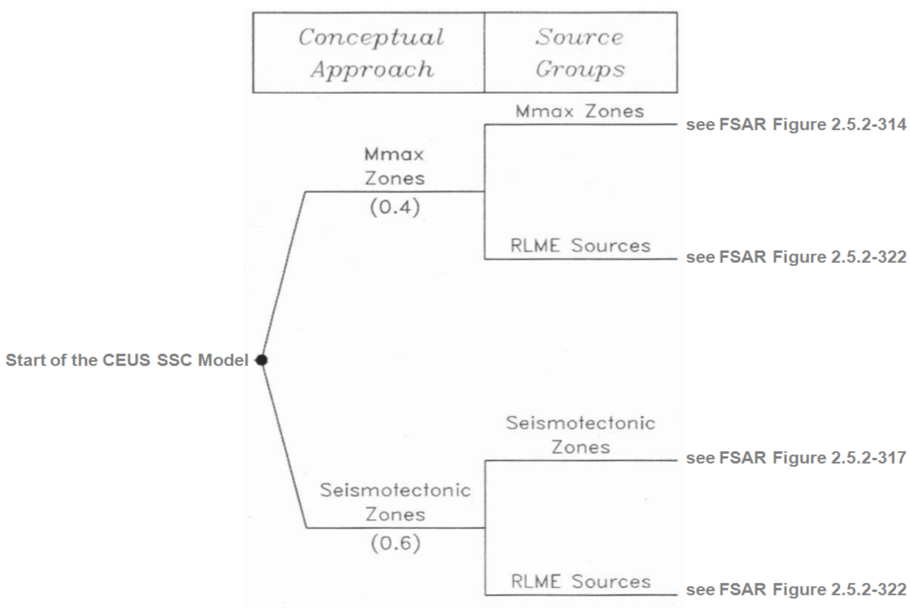

Figure 20.1-2. CEUS SSC Master Logic Tree Showing the Mmax Zones, Seismotectonic Zones, and RLME Sources Models for Assessing the Spatial and Temporal Characteristics of Future Earthquake Sources in the CEUS

The logic tree continues to the right and the continuations are shown in the listed FSAR figures. (Modified from FSAR Figure 2.5.2-312 and NUREG-2115 Figure 4.2.1-1)

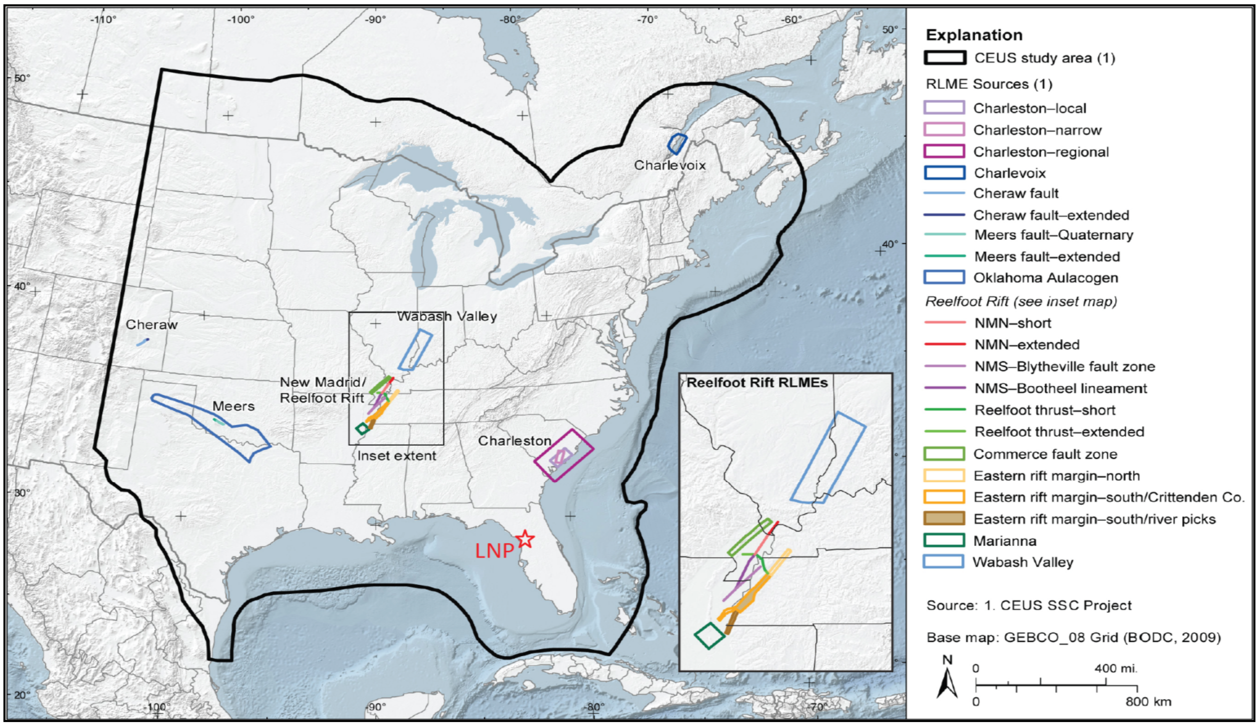

Figure 20.1-3. CEUS SSC Model Study Region (black line), RLME Sources (multicolored lines and polygons), and the Location of the LNP Site (red star)
(FSAR Figure 2.5.2-313 and NUREG-2115 Figure 4.2.2-2)

Each seismic source in the Mmax zones, seismotectonic zones, and RLME sources models is defined by a source geometry, a set of maximum magnitude (Mmax) distributions, a set of recurrence parameters (rate and *b*-values) or methods, and uncertainties. These source characteristics explain where earthquakes may occur, how large the events may be, how often they are expected, and how uncertain those characterizations are, respectively. There are five alternate sources characterized as Mmax zones, 17 sources characterized as seismotectonic zones, and 10 RLME sources. Each of the seismic source zones can have multiple alternative characterizations (geometries, Mmax distributions, recurrence parameters), so the CEUS SSC logic tree weights each source and each alternative, as determined through the SSHAC Level 3 process, and combines them to create the whole model. New to the CEUS SSC model is the use of M as the input magnitude unit, while the EPRI-SOG model used body-wave magnitude (m_b) as its input unit. Additionally, each CEUS SSC areal source has recurrence parameters specified in cells of 0.25-degree longitude by 0.25-degree latitude or 0.5-degree longitude by 0.5-degree latitude. The EPRI-SOG model used 1-degree longitude by 1-degree latitude cells. The smaller cell sized used in the CEUS SSC model achieves higher resolution, especially important for more active regions.

For the Mmax zones model, the CEUS SSC logic tree for the Mmax zones is shown in FSAR Figure 2.5.2-314 and four source geometries are shown in FSAR Figures 2.5.2-315 and 2.5.2-316, while the fifth Mmax zone covers the entire CEUS study region (SER Figures 20.1-1 and -3). The LNP site is located in the “Mesozoic and younger extended

prior" (MESE) Mmax source zones, where MESE-N and MESE-W distinguish between narrow (N) and wide (W) geometry interpretations.

For the seismotectonic zones model, the CEUS SSC logic tree for the seismotectonic zones is shown in FSAR Figure 2.5.2-317, and the sources geometries are shown in FSAR Figures 2.5.2-318 through 2.5.2-321. The LNP site is located in the "extended continental crust-Gulf Coast" (ECC-GC) seismotectonic source zone.

For the RLME model, the CEUS SSC logic tree for the Charleston RLME source is shown in FSAR Figure 2.5.2-322. The Charleston RLME source is the closest RLME source to the LNP site. Each of the 10 RLME sources (SER Figure 20.1-3) has a logic tree defining the uncertainty in its characterization. The characterization of the Charleston RLME source in the CEUS SSC model is similar to the updated Charleston seismic source (UCSS) (SNC, 2006 and 2007) used by the applicant in FSAR Section 2.5.2.4 and discussed by the staff in SER Section 2.5.2.2 and evaluated in 2.5.2.4. FSAR Figure 2.5.2-323 compares the source geometries of the UCSS with the CEUS SSC Charleston RLME source. Comparison of FSAR Figures 2.5.2-214 and 2.5.2-322 shows that the Mmax distributions are the same in the two models and the recurrence frequency of large earthquakes is also very similar for the two models, being approximately 1.8×10^{-3} earthquakes per year.

20.1.1.2 Summary of Cumulative Absolute Velocity Filter Application

In calculations of vibratory ground motion consistent with RG 1.208, applicants can implement the EPRI CAV model (EPRI, 2006) in PSHA calculations. The method is described in RG 1.208 and is based on the probability that earthquakes of a given magnitude can produce damaging ground motions, where the damaging ground motion is defined as CAV exceeding 0.16 g second. The EPRI (2006) model requires exceedance of 0.16 g second level ground motion for the application of the CAV filter, and does not limit earthquake magnitude level. Results of testing the EPRI CAV model indicate that earthquakes of moment magnitude (M) less than 5 have little probability of producing ground motions greater than 0.16 g second. The EPRI (2006) methodology is to perform the hazard integration using a minimum magnitude of M 4.0 and the earthquake recurrence parameters developed for magnitude M 4.0 and larger earthquakes. The guidance in SECY-12-0025 Enclosure 7, Attachment 1, to Seismic Enclosure 1 (ADAMS Accession No. ML12039A188) updated the use of the CAV filter. The updated CAV filter described in SECY-12-0025 for use with the CEUS SSC model limits the CAV filter application not only to 0.16 g second and higher level of ground motion, but also to only magnitudes less than M 5.5. This additional earthquake magnitude requirement affects the integral hazard calculations and may result in a non-negligible increase of the ground motion response spectra (GMRS), which makes the GMRS more conservative.

20.1.2 Summary of Application

The applicant provided information to evaluate the seismic hazard at its site against current NRC requirements and guidance. The information was provided in a response to RAI Letter No. 108 (ADAMS Accession No. ML120550146), which requested, among other things, that the applicant evaluate the seismic hazard at its site against current NRC requirements and guidance as described in SECY-12-0025 Enclosure 7, Attachment 1 to Seismic Enclosure 1 (ADAMS Accession No. ML12039A188), and, if necessary, update the design basis and

structures, systems, and components important to safety to protect against the updated hazards. The applicant responded to RAI Letter No. 108 in Progress Energy Letter NPD-NRC-2012-029 (ADAMS Accession No. ML122230155), dated August 1, 2012. The applicant's response proposed to incorporate changes into the following FSAR Sections:

- 2.5.2.7, "Sensitivity Evaluations for CEUS SSC" (LNP COL 2.5-2)
- 2.5.4.8.7, "Liquefaction Potential Evaluations for CEUS SSC" (LNP COL 2.5-9)
- 3.7.2.4.1.7, "Sensitivity Evaluations for Regulatory Guide 1.60 Spectra FIRS" (LNP SUP 3.7-3 and LNP SUP 3.7-6)
- 3.7.2.8.4, "Median Centered Adjacent Building Relative Displacements for 10^{-5} UHRS" (LNP SUP 3.7-5)
- 19.55.6.3, "Site-Specific Seismic Margin Analysis" (LNP COL 19.59.10-6)

The applicant subsequently incorporated the proposed changes into Revision 5 of the LNP COL FSAR. The applicant supplemented its response with clarifying information in two additional letters dated October 15 and October 31, 2012 (ADAMS Accession Nos. ML12291A857 and ML12313A163).

20.1.3 Regulatory Basis

The applicable regulatory requirements for seismic hazard reevaluation are established and described in the following:

- 10 CFR 100.23, "Geologic and Seismic Siting Criteria," with respect to obtaining geologic and seismic information necessary to determine site suitability and to ascertain that any new information derived from site-specific investigations does not impact the ground motion response spectra derived by a probabilistic seismic hazard analysis.
- 10 CFR 52.79 (a)(1)(iii), "Contents of Applications; Technical Information in Final Safety Analysis Report," (specifically 10 CFR 52.79 (a)(1)(iii)) as it relates to consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 2, "Design Bases for Protection against Natural Phenomena," which requires, in part, that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions.

- Public Law 112-74, “Consolidated Appropriations Act, 2012,” Section 402, states that the NRC shall require reactor licensees to reevaluate the seismic hazards at their sites against current applicable Commission requirements and guidance for such licenses as expeditiously as possible. It also requires each licensee to confirm to the Commission that the design basis for each reactor meets the requirements of its license, as well as current applicable Commission requirements and guidance for such license. The Conference Report for PL 112-74 directs the Commission to implement Fukushima recommendation 2.1 consistent with, or more expeditiously than, the “schedules and milestones” proposed by NRC staff on October 3, 2011 in SECY-11-0037, “Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned.”

In addition, the geologic and seismic characteristics should be consistent with appropriate sections from the following guidance:

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 2.5.2, “Vibratory Ground Motion,” Revision 4

- RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” Revision 1

- RG 1.132, “Site Investigations for Foundations of Nuclear Power Plants,” Revision 2

- RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition)”

- RG 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion”

- RG 1.198, “Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites”

- DC/COL ISG-017, “Interim Staff Guidance on Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses”

- DC/COL ISG-020, “Seismic Margin Analysis for New Reactors Based on Probabilistic Risk Assessment”

- SECY-12-0025 states, in part, that the staff will also request all COL applicants to provide the information required by the specified orders and request for information letters described in this paper, as applicable, through the review process. Enclosure 7 to SECY-12-0025 contains a request for information letter addressing the NTTF Recommendation 2.1 seismic reevaluation, and Enclosure 7, Attachment 1 to Seismic Enclosure 1, describes an acceptable process for developing the information requested.

20.1.4 Technical Evaluation of LNP CEUS SSC Model Sensitivity Evaluation

This SER section provides the staff's evaluation of the applicant's responses to RAI Letter No. 108 (ADAMS Accession No. ML120550146) as they relate to the applicant's evaluation of the seismic hazard at its site against current NRC requirements and guidance as described in SECY-12-0025 Enclosure 7, Attachment 1 to Seismic Enclosure 1 (ADAMS Accession No. ML12039A188). To address the guidance described in SECY-12-0025, the applicant evaluated potential seismic hazards at the LNP site using the CEUS SSC model (NUREG-2115) and applying the CAV filter as described in the SECY, and then performed a sensitivity study comparing the results with those the applicant previously produced using the EPRI-SOG model.

During the applicant's development of its RAI response, the staff conducted a site audit to review calculation packages, interact with the applicant regarding the sensitivity evaluation conducted for the LNP COL application, and to review the applicant's quality assurance documents related to the seismic hazard calculation software. The staff conducted the audit at the Progress Energy Florida offices in Raleigh, NC, on June 18, 19, and 20, 2012, and the audit concluded with a public meeting. The audit summary is available in ADAMS Accession No. ML12235A301.

20.1.4.1 Implementation of the CEUS SSC Model for the LNP Site

The applicant performed hazard calculations using the CEUS SSC model that included contributions from all distributed seismicity source zones that extend within 1,000 kilometers (km) (621 miles (mi)) of the LNP site. Specifically, the applicant included all five Mmax source zones and 12 of the 17 seismotectonic source zones in its calculations of the CEUS SSC hazard at LNP site. The seismotectonic source zones included by the applicant in the hazard calculation were AHEX, GHEX, ECC-AM, ECC-GC, MIDC (-A, -B, -C, and -D), PEZ (-N and -W), RR, and RR-RCG. These sources can be seen in FSAR Figures 2.5.2-318 through -321.

Regarding the RLME sources, the applicant used the Charleston sources and the Reelfoot Rift–New Madrid Fault System (NMFS) fault sources in its hazard calculations. The Charleston RLME source specified in NUREG-2115 contains three alternative source geometries: a local, narrow, and regional source. NUREG-2115 describes the Charleston RLME regional source as being modeled with two alternative fault rupture orientations: (1) fault ruptures are parallel to the long axis of the source (northeast) with a weight of 0.80, and (2) fault ruptures are parallel to the short axis of the source (northwest) with a weight of 0.20. In a letter dated October 15, 2012 (ADAMS Accession No. ML12291A857), the applicant described how calculations for the LNP site were performed using only the northeast orientation for the Charleston RLME regional source with a weight of 1.0. The applicant stated that it performed the calculations in this manner because in NUREG-2115 the hazard at the Savannah test site showed only small sensitivity to the orientation of ruptures in the regional source geometry and the use of only the northeast-southwest orientations is more conservative, producing a higher hazard.

NUREG-2115 includes the results of a sensitivity analysis showing that, at a 10^{-5} annual exceedance frequency when both fault rupture orientations are modeled, the percent difference between the weighted mean average hazard and the northeast orientation is less than

5 percent, indicating that mean hazard at Savannah is not significantly affected by having two alternative rupture orientations for the Charleston regional source. SER Figure 20.1-4 shows the difference in hazard between modeling of the two alternative fault rupture orientations. Because the difference between the hazard using the two orientations is not large and the northeast orientated fault ruptures are weighted 0.80 in this source's final hazard results, the northeast orientated fault ruptures dominate the weighted mean average hazard at the Savannah test site. Therefore use of only the northeast orientated fault ruptures would result in hazard calculations within 5 percent of using both fault rupture orientations.

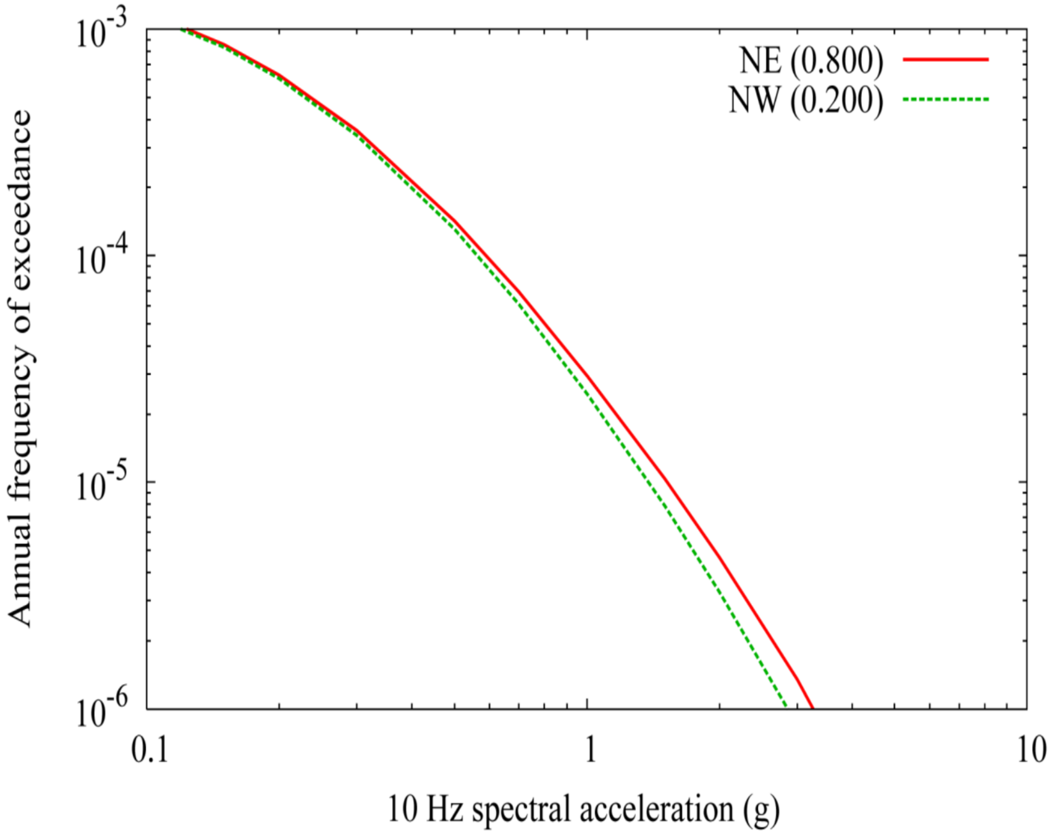

Figure 20.1-4. CEUS SSC Model Charleston RLME Sensitivity to Rupture Orientation at the Savannah Test Site at 1 Hz (top) and 10 Hz (bottom)
(NUREG-2115 Figures 9.3-1 and -2)

The applicant performed sensitivity calculations for the LNP site using a model with northeast ruptures weighted 0.8 and northwest ruptures weighted 0.2 for the Charleston regional source geometry. The applicant's sensitivity calculations showed that hazards for the 10^{-4} and 10^{-5} annual exceedance frequencies at 1 Hz spectral accelerations are approximately 0.04 percent lower than those presented in FSAR Figure 2.5.2-340. The applicant stated that it chose to run the sensitivity calculations at 1 Hz because, for the LNP site, the hazard at this spectral frequency is dominated by the contributions from the Charleston RLME source. Therefore, 1 Hz would be the best frequency at which to perform the sensitivity calculations because the effect

of the differences in Charleston RLME source geometries also would be dominant in the hazard calculation.

Regarding the Reelfoot Rift–NMFS fault sources, the applicant did not use the other Reelfoot Rift RLME sources, such as the Eastern rift margin (ERM) sources, Commerce fault zone, Marianna, or the Wabash Valley source. The applicant chose not to use these sources because of their low contribution to the hazard at the CEUS SSC Chattanooga test site, as shown in SER Figure 20.1-5, and because the LNP site is located even farther from those sources than the Chattanooga site is.

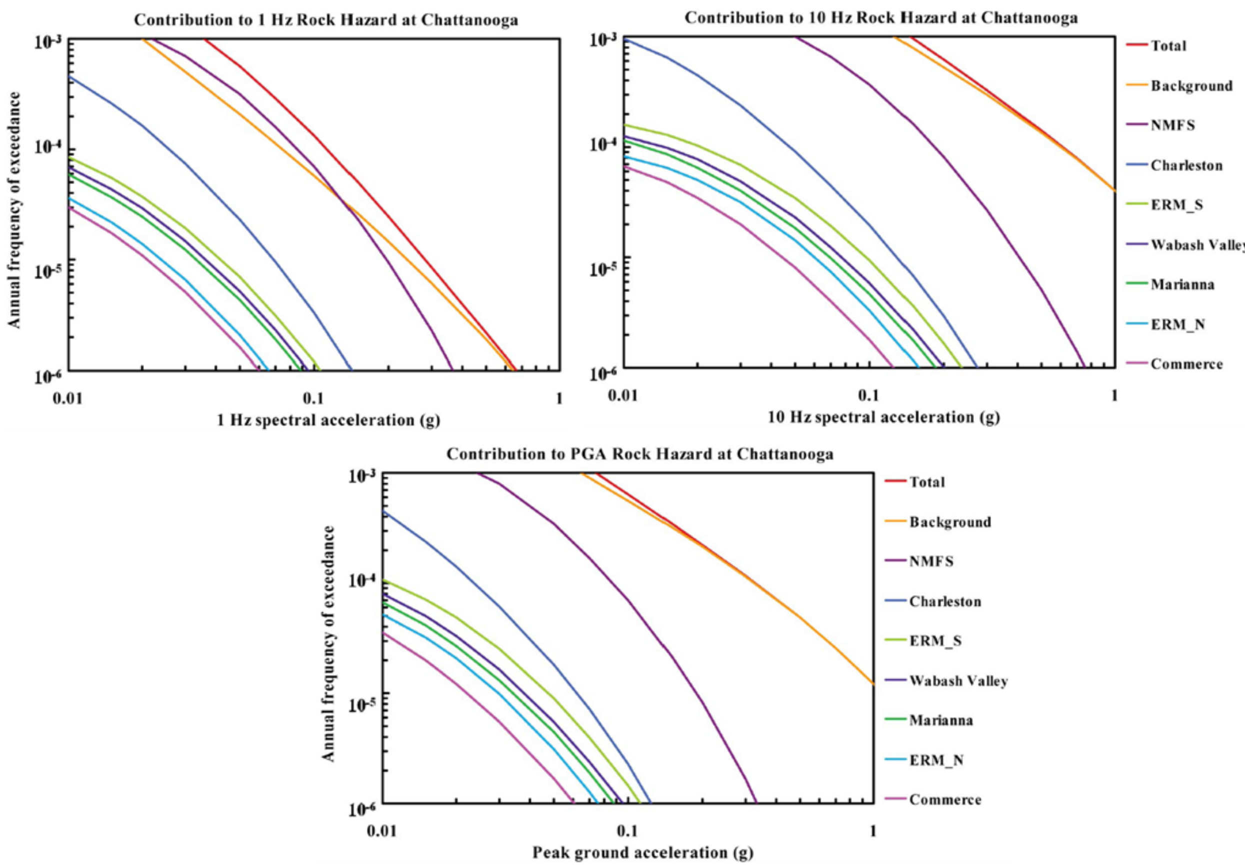

Figure 20.1-5. CEUS SSC Model Chattanooga Test Site Mean Rock Hazard at 1 Hz, 10 Hz, and Peak Ground Acceleration (PGA) (100 Hz) for the Total Hazard and the Contribution by RLME Sources and Background.
(NUREG-2115 Figure 8.2-2d, -2e, and -2f)

NUREG-2115 describes how the equation for the fault rupture area and seismic moment from Somerville et al. (2001), combined with the relationship between moment magnitude and seismic moment from Hanks and Kanamori (1979), should be used in the CEUS to estimate fault rupture area from moment magnitude consistent with the magnitude scale used in modern ground motion prediction equations for CEUS earthquakes. NUREG-2115 describes the

combination of the Somerville et al. (2001) and the Hanks and Kanamori (1979) equations as NUREG-2115 Equation H-1:

$$\log_{10}A = M - 4.366 \qquad \text{Equation 20.1-1}$$

However, this equation is slightly incorrect since combining the fault rupture area and seismic moment equations from Somerville et al. (2001) Table 4 and Hanks and Kanamori (1979) yields this result:

$$\log_{10}A = M - 4.35 \qquad \text{Equation 20.1-2}$$

In both equations, A represents the fault rupture area and M is moment magnitude. The applicant described in a supplement to its response to RAI Letter No. 108 (ADAMS Accession No. ML12313A163) that it used the equation published in NUREG-2115 (SER Equation 20.1-1). However, the applicant performed sensitivity calculations to determine the effect of using the NUREG-2115 Equation H-1 (SER Equation 20.1-1) as compared to using SER Equation 20.1-2 for calculation of spectral accelerations on hard rock. The applicant stated that using SER Equation 20.1-2 produces a fault rupture area for a given magnitude that is approximately 4 percent greater than the corresponding value calculated using NUREG-2115 Equation H-1 (SER Equation 20.1-1). The applicant's calculations for all seven structural frequencies provided in the EPRI ground motion model (0.5, 1.0, 2.5, 5, 10, 25 and 100 Hz) showed that, for annual exceedance frequencies of 10^{-4} and 10^{-5}, spectral accelerations differ by less than 0.2 percent. The applicant concluded that the difference between using NUREG-2115 Equation H-1 (SER Equation 20.1-1) versus SER Equation 20.1-2 has negligible effect on the total rock hazard calculations. For the distributed sources, such as the seismotectonic and Mmax zones, the applicant used the epicentral distance adjustments from EPRI (2004), which models the effect of earthquake ruptures using point sources rather than an extended rupture area and is also based on the Somerville et al. (2001) formula. Based on the applicant's use of the epicentral distance adjustments from EPRI (2004), the applicant used the NUREG-2115 Equation H-1 (SER Equation 20.1-1) equation only when modeling the areal RLME sources, of which Charleston is dominating at the LNP site. Therefore, the applicant's sensitivity study results, which show spectral acceleration percent differences of less than 0.2 percent, demonstrate the negligible difference between using SER Equations 20.1-1 and -2 when modeling the areal RLME sources (e.g., Charleston).

The staff evaluated the applicant's implementation of the CEUS SSC model for the LNP site. RG 1.208 guides applicants to investigate seismic sources within multiple areas, the largest area being described by a radius of 320 km (200 mi) around the site, defined as the site region. Recent COL and ESP applications submitted to the NRC have included seismic sources that reach within the site region in the seismic hazard calculations, in addition to large magnitude sources that lay beyond the 320 km (200 mi) radius, which is consistent with guidance in RG 1.208. Thus, the staff considers the LNP applicant's use of a 1,000 km (621 mi) inclusion zone as appropriate and conservative for use in the applicant's sensitivity evaluation of the CEUS SSC model.

The staff considers the applicant's inclusion of the RLME sources of Charleston and Reelfoot Rift–NMFS fault sources and the exclusion of other RLME sources to be appropriate for the

LNP site hazard calculations using the CEUS SSC model. As shown in SER Figure 20.1-2, the Charleston and Reelfoot Rift RLMEs are the closest RLME sources to the LNP site, and both lie beyond the LNP Site Region. The applicant's sensitivity study showed a 0.04 percent change in the CEUS SSC uniform hazard response spectra (UHRS) because of the Charleston RLME sensitivity to rupture orientation at the LNP site. The NUREG-2115 sensitivity calculations showed a less than 5 percent difference in hazard at the Savannah site because of the Charleston RLME sensitivity to rupture orientation. Given these findings, the staff concludes that the applicant's use of only the northeast orientation for the Charleston RLME regional source with a weight of 1.0 adequately characterizes the hazard from the Charleston RLME regional source at the LNP site.

Regarding the applicant's use of the Reelfoot Rift RLMEs, the applicant stated that the hazard contribution at the Chattanooga test site from the Reelfoot Rift RLMEs is minimal from any source that is not the Reelfoot Rift–NMFS fault sources. Thus, since the LNP site is even farther from the Reelfoot Rift RLME sources than the Chattanooga test site is, the effect on hazard at the LNP site would be even less. Therefore, the applicant only included the Reelfoot Rift–NMFS fault sources in its hazard evaluation. The staff agrees with this logic. SER Figure 20.1-5 shows that the Reelfoot Rift–NMFS fault sources are large contributors to the total hazard, while the other Reelfoot Rift RLME sources are not. The figure also shows that the LNP site is located approximately 708 km (440 mi) farther southeast from the Reelfoot Rift RLME sources than the Chattanooga site.

The staff evaluated the applicant's analysis of the use of NUREG-2115 Equation H-1 (SER Equation 20.1-1) instead of SER Equation 20.1-2 as a method to model seismic sources (ADAMS Accession No. ML12313A163). Based on the applicant's sensitivity calculations showing that the differences in using the two equations is less than 0.2 percent, the staff concludes that it produces negligible effect on the applicant's rock hazard calculations. Additionally, the applicant's use of the EPRI (2004) point source approximation is acceptable to the staff for the LNP site. A sensitivity study was performed in NUREG-2115 to determine the influence of modeling distributed seismicity sources using the fault rupture model versus modeling the sources using the point source approximation in EPRI (2004). In the study, the effect of the seismotectonic zone Midcontinent A (MIDC-A) using the two methodologies was modeled at the Central Illinois test site for 1 and 10 Hz. The seismic hazard calculations using the two methodologies produced less than 10 percent difference. This analysis is applicable to the LNP site because the MIDC-A source has a similar Mmax distribution to the ECC-GC source in which the LNP site is located. Because the applicant's sensitivity study showed negligible effect of the NUREG-2115 Equation H-1 and the NUREG-2115 sensitivity study showed negligible effect of using the EPRI (2004) point source approximation, the staff concludes that the applicant's modeling of the effects of the RLME, seismotectonic, and Mmax distributed seismic sources is acceptable.

The applicant performed the hazard calculations using ground motion prediction equations as described in EPRI (2004, 2006). The applicant used these equations consistent with the use for the EPRI-SOG hazard model results as described in FSAR Section 2.5.2.4.4. Since the applicant used the equations in the same manner it previously did for the EPRI-SOG hazard model, the staff considers it appropriate for use in the applicant's sensitivity evaluation of the CEUS SSC model.

20.1.4.2 Verification of CEUS SSC Model Implementation

As described in SER Section 20.1.1.1 and shown in SER Figure 20.1-1, NUREG-2115 documented the use of the CEUS SSC model at seven test sites. In FSAR Section 2.5.2.7.2, the applicant described its independent calculation of hazard at these seven sites. The applicant performed this calculation to demonstrate its adequate implementation of the CEUS SSC model. The applicant stated that it closely matched the test sites' mean and fractile hazard curves, as shown in FSAR Figures 2.5.2-325 through -331. The applicant then used the mean hazard curves to calculate the ground motion levels with annual frequencies of exceedance of 10^{-4}, 10^{-5}, and 10^{-6}, as listed in FSAR Table 2.5.2-232. The applicant stated that the differences in ground motion values are generally less than 5 percent and that these differences are not considered significant. The applicant then concluded that its implementation of the CEUS SSC model is adequate for use in computing hazard at the LNP site.

As shown in SER Figure 20.1-1, the Savannah site is the closest CEUS SSC test site to the LNP site. Second, the Chattanooga site is comparable to the LNP site because the Chattanooga site is located at a similar distance from the Charleston RLME source as the LNP site. The Chattanooga site is approximately 547 km (340 mi) from the Charleston RLME source, and the LNP site is located at a distance of approximately 482 km (300 mi). The staff considers the Savannah and Chattanooga sites to be the test sites applicable to the LNP site. Consistent with the applicant's analysis, the percent difference between the applicant's calculation and that of NUREG-2115 spectral acceleration values calculated at the Chattanooga site show less than a 3 percent difference at annual frequencies of exceedance of 10^{-4}, 10^{-5}, and 10^{-6} at 1 Hz, 10 Hz, and peak ground acceleration (PGA). The staff does not consider this to be a significant difference in ground motion values. SER Figure 20.1-6 shows the applicant's closely matched calculation to the Chattanooga test site mean and fractile hazard curves.

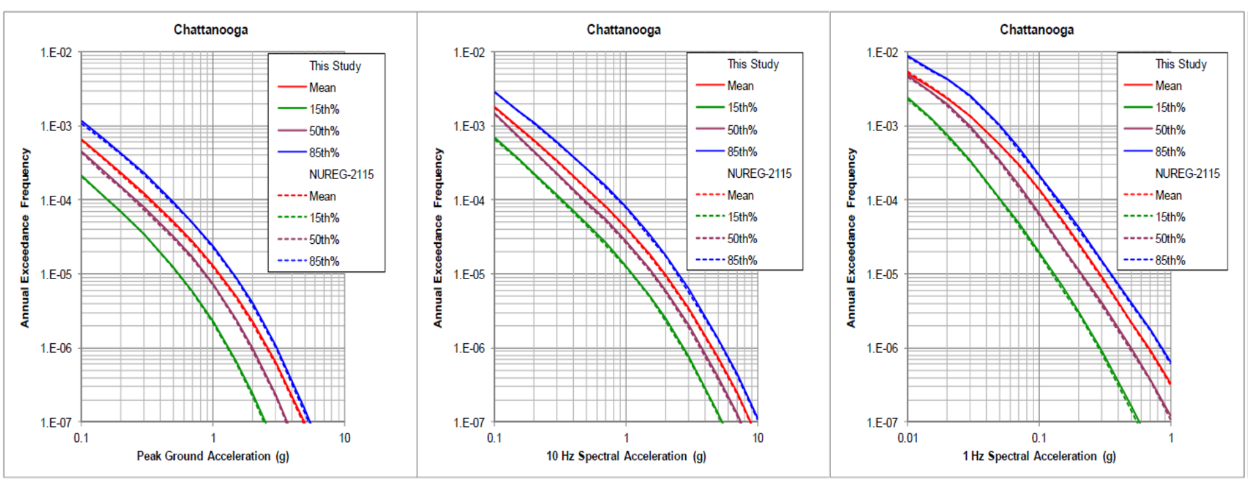

Figure 20.1-6. Comparison of Mean and Fractile Hazard at the Chattanooga Site Computed by the Applicant ("This Study") and Reported in NUREG-2115
(FSAR Figure 2.5.2-326)

The percent differences at the Savannah site, however, are larger and range between 5.3 and 13.1 percent. SER Figure 20.1-7 shows the results graphically. These percent differences are the only values that rise above the applicant's generalized assessment of "values are generally less than 5 percent". The applicant attributes the differences to implementation details of modeling the large-magnitude rupture locations for sites near the Charleston RLME sources, where the Savannah site is located 128 km (80 mi) southwest of the Charleston RLME sources. As described in SER Section 2.5.2.7.2.6, the applicant tested two implementation methods for modeling the Charleston RLME at the Savannah test site: (1) A series of closely spaced pseudo faults parallel to the northeast orientation of the zone and earthquake ruptures were modeled as occurring uniformly along these faults, and (2) the source zone was filled with a grid of uniformly spaced points and at each point magnitude-dependent ruptures were placed with the specified northeast orientation with a random location on the grid point.

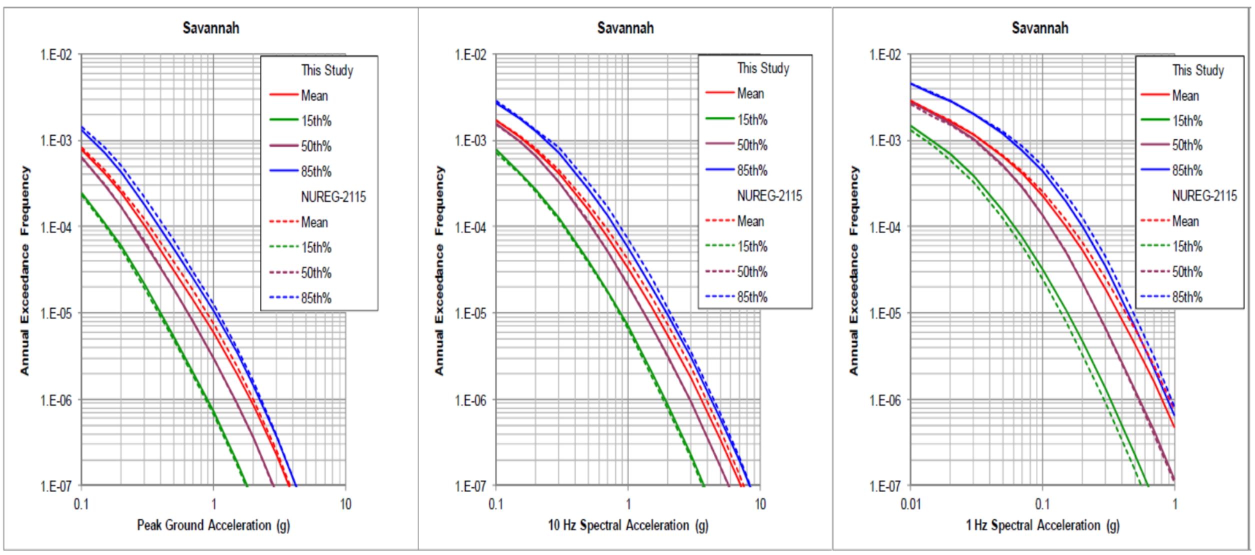

Figure 20.1-7. Comparison of Mean and Fractile Hazard at the Savannah Site Computed by the Applicant ("This Study") and Reported in NUREG-2115
(FSAR Figure 2.5.2-330)

According to the applicant, when method (1) was compared with method (2), the effect of using method (2) was that the model increased the probability of rupture locations near the boundary. This alternative process, method (2), produced acceptable results in comparison with those presented in NUREG-2115 using method (1). In a letter dated October 15, 2012 (ADAMS Accession No. ML12291A857), the applicant stated it used implementation method (1) in its hazard calculations for the LNP site. From testing the two implementation methods, the applicant concluded that the differences in modeling the Charleston RLME source have minimal impact on the computation of the mean hazard at larger distances, such as at the LNP site, as evidenced by the applicant's ability to reproduce the hazard at the Chattanooga test site

The staff reviewed the applicant's implementation of the NUREG-2115's CEUS SSC model. The staff considers the less than 5 percent differences in spectral acceleration values for the Chattanooga test site to be well within the precision of any PSHA calculations. The staff

considers the applicant's tests of multiple implementation methods for the Charleston RLME source as a thorough investigation of the realization of that source model at the Savannah test site. The percent differences in spectral acceleration values ranging between 5.3 and 13.1 percent calculated at the Savannah test site are still within the accuracy of PSHA modeling. Additionally, since the Chattanooga test site lies at a similar distance from the Charleston source as the LNP site, the staff concludes that at those distances the applicant's calculations are consistent with the calculations detailed in NUREG-2115. Therefore, since the applicant used the same input parameters as described and used in NUREG-2115 and the applicant's model results are within the precision expected for the model, the staff considers the applicant's comparison results at the Chattanooga and Savannah test sites to be adequate. Finally, since both the comparison results at the Chattanooga and Savannah test sites are found to be adequate to the staff, and the other five test site comparisons show similar low percent differences, the staff considers the applicant's modeling and comparison of hazard at all seven test sites to be an acceptable demonstration that the applicant has adequately implemented the calculation of seismic hazard using the CEUS SSC model. Lastly, during the staff's audit of the applicant's quality assurance documents related to the seismic hazard calculation software, the staff verified the findings through documentation of the software development process.

20.1.4.3 Uniform Hazard Response Spectra and Deaggregation Results

To calculate the CEUS SSC hard rock UHRS, the applicant compiled the mean seismic hazard curves calculated for the LNP site. Consistent with the applicant's EPRI-SOG hazard calculations, the applicant used the CEUS SSC model to calculate hazard at seven spectral frequencies: 0.5, 1, 2.5, 5, 10, 25 Hz, and PGA. SER Figure 20.1-8 shows the applicant's calculation of mean total hazard from the EPRI-SOG and CEUS SSC models at 10 Hz and 1 Hz and the contribution to hazard at those frequencies of the three main source contributors: distributed seismicity sources, the Charleston sources, and the Reelfoot Rift–NMFS fault sources. This data allowed the applicant to isolate the cause of the differences in mean total hazard and to analyze which source type could be attributed to that difference.

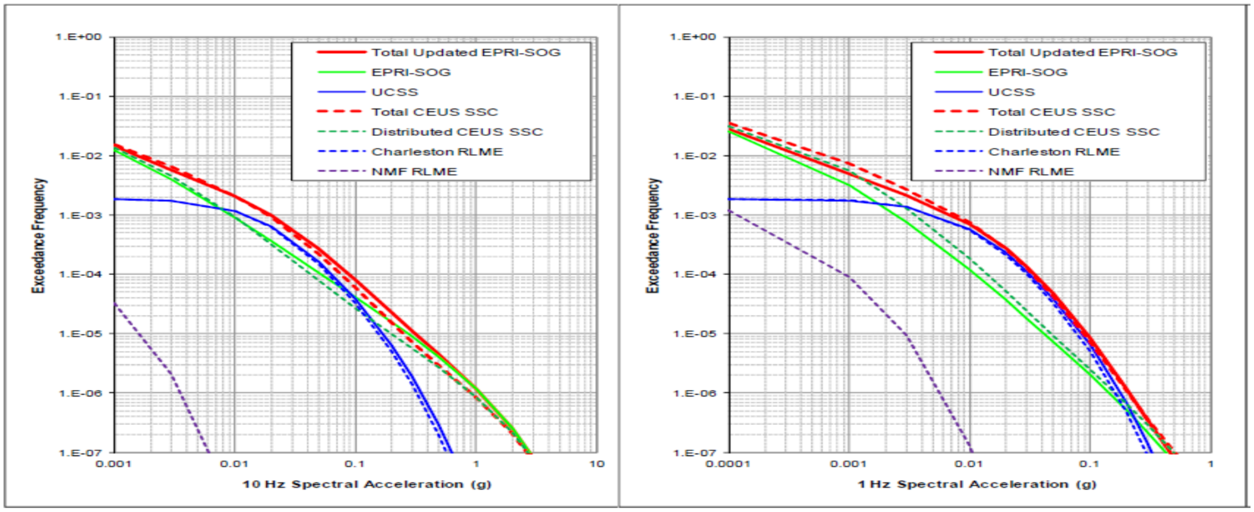

Figure 20.1-8. Contribution of the Different Source Types to the Total Mean Hazard (red) at the LNP Site for 10 Hz (left) and 1 Hz (right) – Distributed Sources (green), Charleston Sources (blue), and Reelfoot Rift–NMFS Fault Sources (purple)
(FSAR Figure 2.5.2-339)

The applicant found that the largest differences between hazards were caused by the distributed seismicity sources, the green lines in SER Figure 20.1-8. For 10 Hz, the hazard determined using the CEUS SSC model is slightly lower than that from the updated EPRI-SOG sources, and for 1 Hz, the hazard determined using the CEUS SSC model is slightly higher than that from the updated EPRI-SOG sources. The applicant attributed the differences at 10 Hz to the CEUS SSC model's lower prediction of seismicity rates in the region around the LNP site. The applicant attributed the differences at 1 Hz to the larger Mmax values for distributed seismicity sources in the CEUS SSC model compared to those for the updated EPRI-SOG model.

SER Figure 20.1-9 shows the applicant's hard rock UHRS. The UHRS based on the CEUS SSC model are lower than those based on the updated EPRI-SOG model at spectral frequencies greater than 2.5 Hz and higher at low frequencies for the 10^{-3} mean annual exceedance frequency. However, the EPRI-SOG UHRS for the 10^{-4} and 10^{-5} mean annual exceedance frequencies, which are used to calculate the GMRS, are higher than those from the CEUS SSC at all frequencies shown.

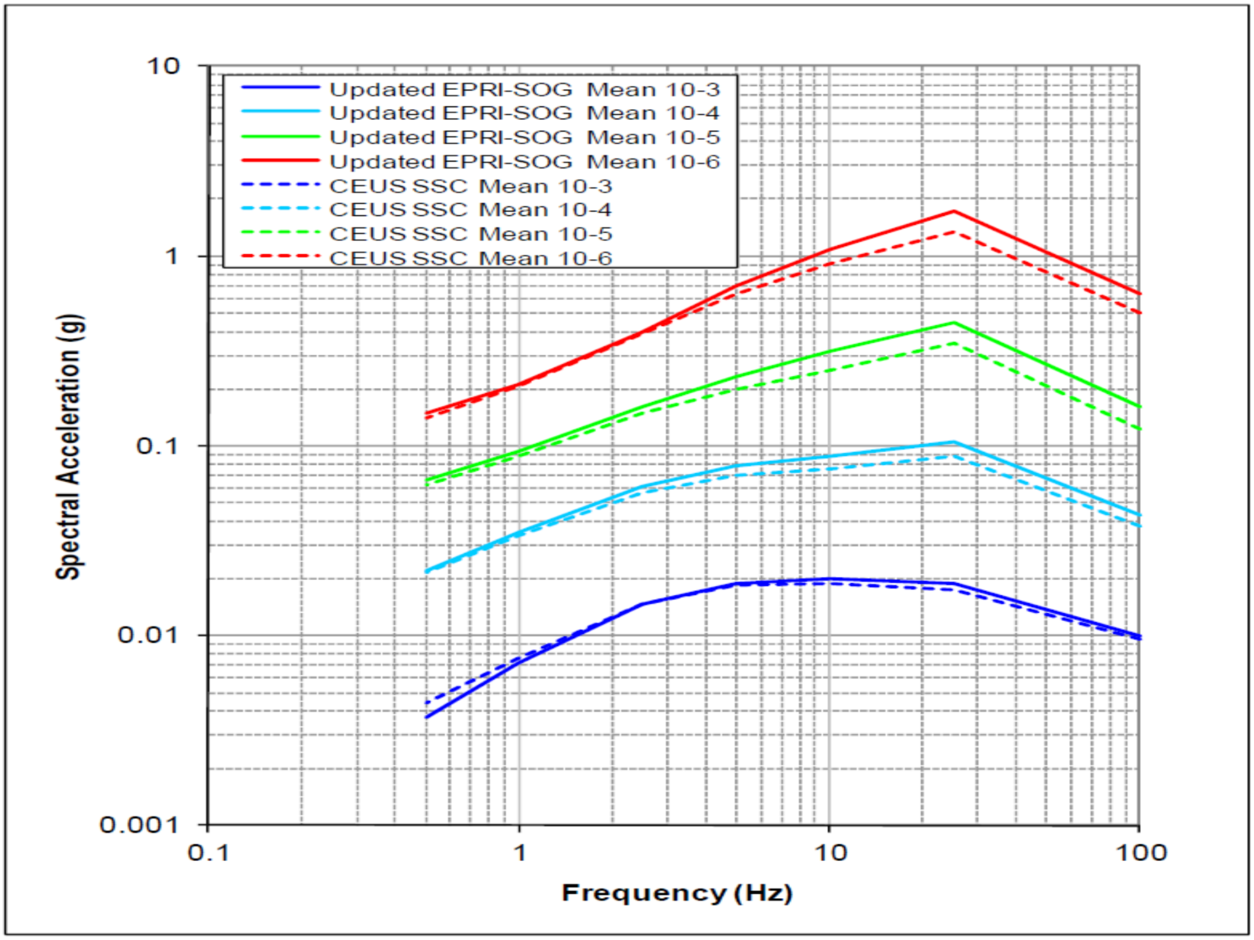

Figure 20.1-9. Comparison of Hard Rock UHRS Based on Updated EPRI-SOG and CEUS SSC models
(FSAR Figure 2.5.2-340)

Following calculation of the UHRS, the applicant deaggregated the spectra consistent with RG 1.208 to determine the controlling earthquakes. FSAR Figures 2.5.2-341, -343, and -344 show the deaggregation results for mean annual exceedance frequencies of 10^{-3}, 10^{-5}, and 10^{-6}, respectively. SER Figure 20.1-10 shows the deaggregation results for the 10^{-4} mean annual exceedance frequency using the CEUS SSC model compared to those using the EPRI-SOG model. The applicant demonstrated that the deaggregation results are similar to those for the updated EPRI-SOG hazard results, as seen in SER Figure 20.1-10.

The staff also conducted an independent confirmatory analysis of the hard rock hazard and UHRS at the LNP site. The staff calculated the hazard using the CEUS SSC model (NUREG-2115) for distances up to 750 km (469 mi) for distributed seismicity sources and 1,000 km (621 mi) for RLME sources. In its confirmatory analysis, the staff used the EPRI 2004-2006 ground motion attenuation model (same as the applicant). The staff's calculations of the 10^{-4} and 10^{-5} UHRS are enveloped by the applicant's calculations. Because the UHRS

results and the total hard rock hazard curves developed by the staff are in good agreement with those developed by the applicant, the staff finds the applicant's PSHA acceptable.

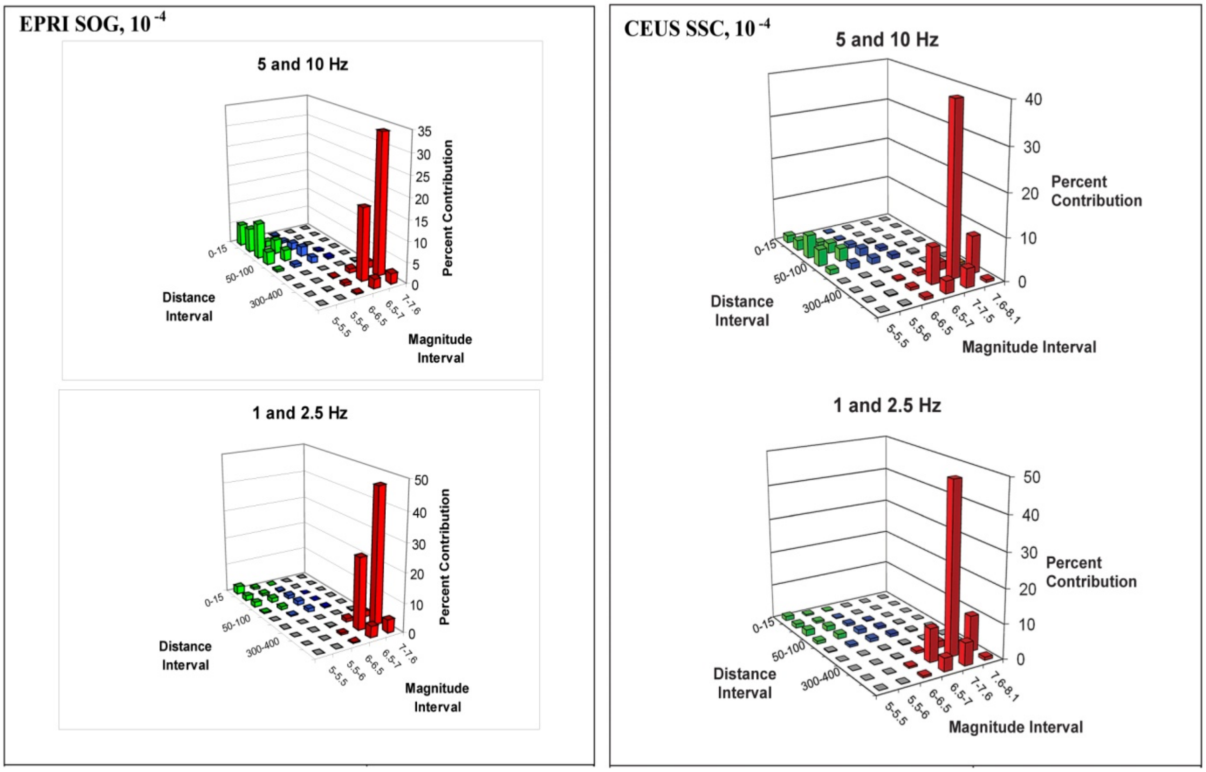

Figure 20.1-10. Comparison of 10^{-4} Deaggregation Results from EPRI SOG (left) and CEUS SSC (right) Models
(FSAR Figures 2.5.2-240 and 2.5.2-342)

The staff evaluated the applicant's determination of the hard rock UHRS and deaggregation results. Since the applicant used a method to determine the UHRS that was consistent with the calculation in FSAR Section 2.5.2.4 and guidance in RG 1.208, the staff concludes that the applicant properly calculated its hard rock UHRS. Finally, the comparison of the deaggregation results (SER Figure 20.1-10) shows that, as expected, the earthquakes controlling the spectra are similar when using either seismic source model.

20.1.4.4 Ground Motion Response Spectra and Updated Cumulative Absolute Velocity Filter

Following the calculation of the hard rock UHRS, the applicant calculated the GMRS using a CAV filter and the CEUS SSC model. The applicant then compared its CEUS SSC GMRS to its previous GMRS that were determined using the updated EPRI-SOG model, as described in FSAR Section 2.5.2.6. For calculation of the CEUS SSC GMRS, the applicant used the same seismic source inputs it used for calculation of the UHRS. Additionally, to calculate the CEUS SSC GMRS, the applicant used the same amplification functions it developed in FSAR Section 2.5.2.5 for use with the EPRI-SOG hazard results. The applicant did not recalculate the

amplification functions based on the similarity of the UHRS deaggregation results of the CEUS SSC and EPRI-SOG models (SER Figure 20.1-10).

Following the guidance in SECY-12-0025 Enclosure 7, Attachment 1, to Seismic Enclosure 1 (ADAMS Accession No. ML12039A188), the applicant updated its use of the CAV filter from what it used with the updated EPRI-SOG model. For use with the EPRI-SOG model, the applicant implemented the CAV filter described in EPRI (2006). Using the EPRI (2006) methodology, the applicant performed the hazard integration using a minimum magnitude of M 4.0 and the earthquake recurrence parameters developed for magnitude M 4.0 and larger earthquakes. Using the updated CAV filter with the CEUS SSC model, as described in SECY-12-0025, the applicant limited the CAV filter application to magnitudes less than M 5.5. The applicant's calculation of ground motions at the 10^{-4} annual exceedance frequency are zero when it used the EPRI-SOG model and EPRI (2006) CAV methodology. Using the CEUS SSC model and updated CAV methodology from SECY-12-0025, the ground motions at the 10^{-4} annual exceedance frequency are not equal to zero. The applicant also saw an effect at the 10^{-5} annual exceedance frequency, but the difference is not as large. The FSAR Figure 2.5.2-352 shows the results at the 10^{-4}, 10^{-5}, and 10^{-6} annual exceedance frequencies at the GMRS elevation. The 10^{-4} and 10^{-5} UHRS based on the CEUS SSC model using modified CAV are higher than those using the updated EPRI-SOG model with full CAV. The applicant stated that the higher motions are primarily caused by the modification to the CAV methodology. For the 10^{-6} UHRS, the results based on the CEUS SSC model using modified CAV and those using the updated EPRI-SOG model with full CAV are similar for frequencies of 5 Hz and less and lower at higher spectral frequencies. The lower UHRS amplitudes at spectral frequencies above 5 Hz are due to the difference in the rock hazard between the two models.

SER Figure 20.1-11 shows the applicant's GMRS, which was calculated using the CEUS SSC model, the updated CAV methodology, and the previously determined site amplification functions. The applicant determined the GMRS from the UHRS using relationships described in RG 1.208. RG 1.208 guides applicants to calculate the GMRS using the following relationship:

$$\text{GMRS} = \text{UHRS} * \text{DF} \qquad \text{Equation 20.1-3}$$

where

$$\text{UHRS} = \text{Mean } 10^{-4} \text{ UHRS}$$
$$\text{DF} = \max\{1.0,\ 0.6\,(A_R)^{0.8}\}$$
$$A_R = 10^{-5}\text{UHRS} / 10^{-4}\text{UHRS}$$

The resulting horizontal GMRS is a combination of the site-specific spectra at the GMRS elevation at 10^{-4} and 10^{-5} annual exceedance frequencies. RG 1.208 alternatively states that if A_R is greater than 4.2, then the applicant should determine the GMRS using 45 percent of the site-specific spectra for the 10^{-5} annual exceedance frequency. For the LNP application, the applicant used both GMRS calculation methods and took the horizontal GMRS to be equal to an envelope of the two spectra. The applicant's determination of both horizontal GMRS spectra is shown in FSAR Figure 2.5.2-354. FSAR Table 2.5.2-234 lists the resulting GMRS. To determine the vertical GMRS, the applicant used the vertical to horizontal (V/H) spectral ratios described in FSAR Section 2.5.2.6.4, which were used for the EPRI-SOG calculations as well.

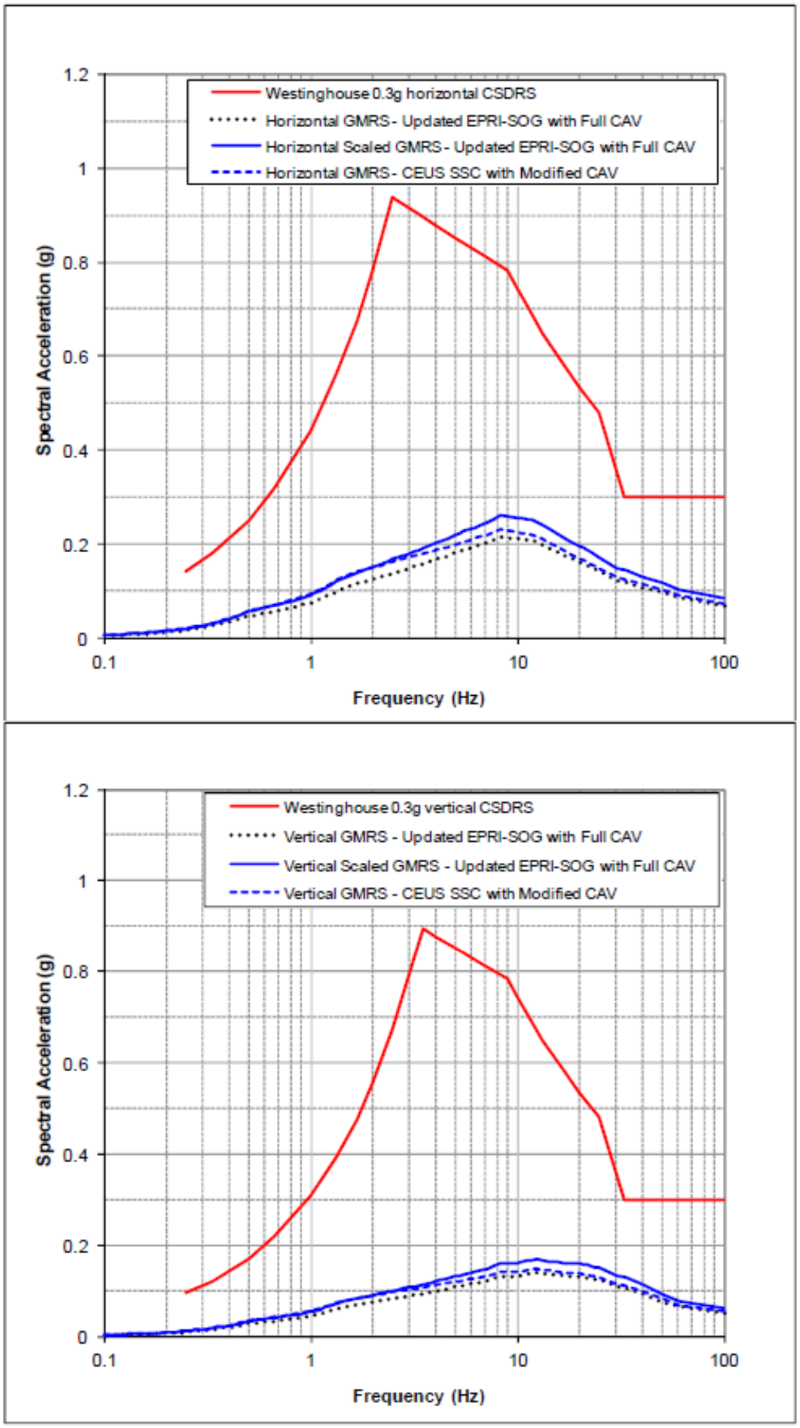

Figure 20.1-11. Comparison of GMRS Based on Updated EPRI-SOG and CEUS SSC Models.
(FSAR Figure 2.5.2-355)

In SER Figure 20.1-11, the horizontal and vertical GMRS using the CEUS SSC model is compared to the GMRS using the updated EPRI-SOG model and the AP1000 certified seismic design response spectra (CSDRS). The applicant presented percent differences between the GMRS in FSAR Table 2.5.2-234. The GMRS based on the CEUS SSC model is enveloped by the GMRS based on the updated EPRI-SOG model, except for frequencies between 0.2 and 2 Hz where the CEUS SSC-based GMRS is up to 4 percent higher. However, the GMRS calculated using the CEUS SSC approach combined with the updated CAV filter methodology resulted in higher amplitudes of response spectra than calculations based on the EPRI-SOG approach combined with the original CAV filter application presented in the LNP COL revisions 1 through 4. To address this issue, the applicant revised the original GMRS calculations presented in LNP COL revisions 1 through 4 by applying a 1.212 scaling factor to the original GMRS. This scaled GMRS is presented in Section 2.5.2.6 of FSAR Revision 5 in the LNP COL application and is described in SER Section 2.5.2.6. The 1.212 scaling factor is consistent with the scaling factor applied to the foundation input response spectra (FIRS) in compliance with the 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," requirement that the horizontal component of the FIRS in the free-field at the foundation level of the structure be an appropriate response spectrum with a minimum PGA of 0.1g.

The staff reviewed the applicant's calculation of the GMRS. Based on the similarity of the spectral shapes of the UHRS calculations using the CEUS SSC model and the updated EPRI-SOG model (SER Figure 20.1-9) and the similarity of the UHRS deaggregation results (SER Figure 20.1-10), the staff concludes that it is not necessary for the applicant to recalculate the LNP site amplification functions. The staff also concludes that the previously developed site amplification functions are appropriate for use with the CEUS SSC calculations.

Because the applicant followed staff guidance on updating the implementation of the CAV filter, and the results are consistent with the staff's expectation of an increase in ground motion caused by the changes in CAV methodology, the staff concludes that the applicant adequately calculated the GMRS regarding the implementation of CAV. Since the applicant calculated the horizontal GMRS consistent with RG 1.208 and took the maximum result from the two methods described in RG 1.208, the staff concludes that the applicant calculated the horizontal GMRS using the CEUS SSC model correctly and conservatively. The staff concludes that the applicant correctly calculated the vertical GMRS using the methods consistent with RG 1.208. The staff does not consider the approximately 4 percent difference between 0.2 to 2 Hz of the CEUS SSC GMRS to the updated EPRI-SOG GMRS to be a significant difference. The staff considers a 4 percent difference to be well within the accuracy of the PSHA and site response analyses. The applicant chose to scale its GMRS based on the updated EPRI-SOG model by a factor of 1.212. As a result, the CEUS SSC GMRS is mainly below the updated EPRI-SOG GMRS described in FSAR Section 2.5.2.6. Finally, since the applicant scaled up the GMRS, the staff concludes that a further update to the LNP site-specific GMRS is not needed and that the GMRS calculated using the updated EPRI-SOG model in FSAR Section 2.5.2.6 adequately characterizes the ground motion at the LNP site.

20.1.4.5 CEUS SSC Liquefaction Potential Evaluation

To evaluate the seismic hazard at the LNP site against the new hazard calculation requested by NRC RAI Letter 108, the applicant provided a liquefaction potential assessment using CEUS SSC in its FSAR Section 2.5.4.8.7. Because the soil under the nuclear island will be excavated and backfilled with roller-compacted concrete (RCC), the applicant only performed the LNP site-specific liquefaction analysis for soil beyond the nuclear island perimeter. Regarding the liquefaction potential of soils under the adjacent annex, turbine, and radwaste buildings, the applicant stated that for the ground motion PGA at finished grade elevation (+51 ft. NAVD88) for the performance based surface response spectra (PBSRS) soil profile computed without CAV and using the CEUS SSC model is 0.091g. This value is less than the corresponding 0.118g PGA computed without CAV and using the updated EPRI-SOG model. Therefore, the applicant concluded that the liquefaction evaluations based on the updated EPRI-SOG LNP ground motions bound those from the CEUS SSC ground motions.

The staff reviewed FSAR Section 2.5.4.8.7, associated FSAR Sections 2.5.2 and 2.5.4, and the applicant's response to NRC RAI Letter No. 108 (ADAMS Accession No. ML122230155). The staff noted that the soil profiles used to develop the PBSRS were based on the statistics of the iterated soil properties for the randomized site profiles. Earthquake-induced cyclic stresses in the soil column were based on ground motions computed for the PBSRS profile using the updated EPRI-SOG model. The staff also checked the corresponding PGAs at the GMRS elevation (elevation +36 ft. NAVD88) and at the base of the excavation (elevation -24 ft. NAVD88) to confirm that the PGA values from the updated EPRI-SOG source model at these elevations also envelop the PGA values from the CEUS SSC model. The staff verified that PGA values computed from the updated EPRI-SOG source model are 0.092g and 0.071g at the GMRS elevation and the base of the excavation, respectively, which are greater than the PGA values of 0.070g and 0.054g calculated by the CEUS SSC model at the same elevations.

RG 1.198, "Procedures and Criteria for Accessing Seismic Soil Liquefaction at Nuclear Power Sites," provides guidance on assessing soil liquefaction potential under seismic loading at nuclear power plant sites. Soil liquefaction potential can be expressed in terms of a factor of safety (FS) against the occurrence of liquefaction as:

$$FS = CRR/CSR \qquad \text{Equation 20.1-4}$$

where CRR (cyclic resistance ratio) is the available soil resistance to liquefaction, expressed in terms of the cyclic stress required to cause liquefaction, and CSR (cyclic stress ratio) is the cyclic stress generated by the design earthquake.

The staff notes that RG 1.198 endorses the Seed & Idriss/Yond procedure to evaluate soil liquefaction potential. From this method, CSR is proportional to the horizontal PGA at the ground surface that is generated by the earthquake, the ratio of total stress to effective vertical overburden stress, and a stress reduction coefficient depending on its depth below the ground surface. From this calculation it can be deduced that FS is inversely proportional to the horizontal PGA generated by the earthquake. The staff concludes that a higher PGA will result in a lower FS, which is also in agreement with general engineering principles.

Since the PGA values computed from the updated EPRI-SOG source model are greater than the PGA values calculated by the CEUS SSC model at the finished grade, the staff reasonably

concludes that the evaluation of liquefaction susceptibility of soils under the adjacent annex, turbine, and radwaste buildings using the updated EPRI-SOG source model is more conservative in comparison with the evaluation using the CEUS SSC model for this LNP site-specific case. Therefore, detailed reanalysis of the soil liquefaction potential is not necessary for the ground motions using the CEUS SSC model. The detailed technical evaluation of the soil liquefaction potential of soils under the adjacent annex, turbine, and radwaste buildings based on the ground motions using the updated EPRI-SOG model is documented in SER Section 2.5.4.8.

Based upon its review of LNP FSAR Section 2.5.4.8, the staff concludes that the applicant analyzed the liquefaction potential following the guidance of RG 1.198. The staff reviewed the applicant's analysis of PGA values from ground motions estimated by both the updated EPRI-SOG model and the CEUS SSC model, and it confirmed that the horizontal PGA values at the finished grade, ground surface, and excavation elevation computed using the updated EPRI-SOG model are higher than that by the CEUS SSC model, which leads the staff to conclude that the applicant has correctly and conservatively evaluated earthquake-induced cyclic stresses within soils in its liquefaction potential analysis. The staff concludes that the liquefaction evaluations based on the updated EPRI-SOG LNP ground motions bound those from the CEUS SSC ground motions.

20.1.4.6 Structural Seismic Evaluation

In Letter NPD-NRC-2012-029 (ADAMS Accession No. ML122230155), dated August 1, 2012, the applicant provided a response to NRC RAI Letter No. 108. The staff reviewed the applicant's response to evaluate the impact on the safety conclusions described in SER Sections 3.7 and 3.8. For determining the adequacy of the RAI response, the staff considered the applicant's ground motion sensitivity evaluations and their effect on: (1) the nuclear island floor response spectra (FRS), (2) the RCC bridging mat design, and (3) the seismic interaction between the seismic Category I and the adjacent seismic Category II structures.

During the review, the staff applied the guidance of Standard Review Plan (SRP) Sections 3.7 and 3.8, as well as relevant regulatory guides, with references to related industry standards. The staff's technical evaluation is summarized below.

20.1.4.6.1 EPRI-SOG FIRS

The FIRS for the AP1000 standard plant satisfy the applicable regulatory requirements. Appendix S to 10 CFR Part 50 requires that the horizontal component of the FIRS in the free-field at the foundation level of the structure be an appropriate response spectrum with a minimum PGA of 0.1g. SRP Section 3.7 and Interim Staff Guidance (ISG) DC/COL ISG-017, "Interim Staff Guidance on Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses," provide implementation guidance for satisfying the minimum FIRS requirement.

LNP FSAR Section 3.7.1.1.2, "Foundation Input Response Spectra," describes the development of the LNP site-specific FIRS. This section states that the nuclear island is supported on 10.7 m (35 ft) of RCC over rock formations as described in LNP FSAR Section 2.5.4.5. This FSAR section also states that the FIRS, developed using the updated EPRI-SOG model, were

developed at elevation -7.3 m (-24 ft) and 3.4 m (11 ft) corresponding to the bottom of the bridging mat and basemat foundation elevations, respectively. The basemat foundation FIRS at elevation 3.4 m (11 ft) were amplified (or scaled) to 0.1g PGA for the purpose of meeting the minimum spectrum regulatory requirement. The scaled FIRS at elevations -7.3 m (-24 ft) and 3.4 m (11 ft) are shown on LNP FSAR Figures 3.7-201 and 3.7-205, respectively.

The applicant used the LNP PBSRS to compute the maximum relative displacements of the annex, turbine, and radwaste buildings' drilled shaft foundations and to evaluate the site-specific seismic interaction of these buildings with respect to the nuclear island. LNP FSAR Section 2.5.2.6 describes the development of the PBSRS at the design-grade elevation of 15.5 m (51 ft).

The staff's review found the LNP site-specific FIRS and PBSRS, based on the updated EPRI-SOG model, to be acceptable on the basis that they were performance-based, broad-banded, and anchored to 0.1g PGA. The staff's evaluation of the site-specific LNP FIRS and PBSRS is described in Section 3.7.1 of this SER.

20.1.4.6.2 EPRI-SOG versus CEUS SSC FIRS

In response to NRC RAI Letter No. 108, the applicant compared the FIRS developed using the EPRI-SOG and the CEUS SSC models. The applicant concluded that the site-specific FIRS developed from the updated EPRI-SOG model and scaled to 0.1g PGA envelop the CEUS SSC FIRS. Based on these results, the applicant concluded that the results of the soil-structure interaction analysis presented in Subsections 3.7.2.4 of the LNP FSAR are valid for the LNP site ground motion based on the CEUS SSC model.

The staff reviewed the applicant's FIRS comparisons, shown in LNP FSAR Figure 2.5.2-358 and finds that the CEUS SSC horizontal and vertical FIRS are enveloped by the EPRI-SOG FIRS scaled to meet 10 CFR Part 50, Appendix S, requirements.

The staff also performed a review of the applicant's PBSRS comparisons of surface motions developed using the EPRI-SOG and CEUS SSC models, shown in LNP FSAR Figure 2.5.2-357. The comparison indicated that the PBSRS developed using the scaled EPRI-SOG model envelop the PBSRS developed using the CEUS SSC model. The staff also compared the LNP site-specific FIRS and PBSRS to the AP1000 certified seismic design response spectra (CSDRS) and notes that a significant margin exists to the standard plant CSDRS. On this basis, the staff concludes that the applicant's site-specific soil-structure interaction analysis, reviewed in Section 3.7.2 of this SER, remains valid.

20.1.4.6.3 Consideration of RG 1.60 Minimum FIRS

For the purpose of addressing the latest NRC regulatory guidance (i.e., DC/COL ISG-017, "Interim Staff Guidance on Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses"), the applicant considered a minimum FIRS at the plant foundation level consistent with the RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," spectral shape with a peak ground acceleration of 0.1 g. The staff notes that a response spectrum having the characteristics of a broad-banded RG 1.60 spectrum shape has increased energy in the low-frequency range (<10 Hz), which is of importance in

structural design. The applicant performed a sensitivity study to assess the differences in the FRS at the six key locations using the RG 1.60 FIRS and the scaled site-specific FIRS and compared them to the AP1000 CSDRS FRS. In LNP FSAR Section 3.7.2.4.1.7, the applicant described the approach for assessing the differences between the FRS based on the site-specific FIRS and the RG 1.60 FIRS.

The applicant did not perform a separate soil-structure interaction analysis using the RG 1.60 FIRS as input, but instead scaled the FRS results by appropriate scale factors derived by comparing the ratio of the RG 1.60 FIRS to the site-specific FIRS. LNP FSAR Tables 3.7-203 and 3.7-204 provide a comparison of the horizontal and vertical ratios of the RG 1.60 FIRS to the site-specific FIRS as a function of frequency (1-100 Hz). The applicant scaled the FRS at the six key locations and presented the results in LNP FSAR Table 3.7-205. The scaling was performed for the FRS dominant structural frequencies and considered horizontal and vertical responses (i.e., X, Y, and Z directions). The applicant concluded that despite increases in amplitude for the RG 1.60 FRS, there is additional margin with respect to the AP1000 CSDRS FRS.

The staff reviewed the comparison of the LNP horizontal and vertical site-specific FIRS and the RG 1.60 FIRS presented in LNP FSAR Tables 3.7-203 and 3.7-204, respectively, and finds that the RG 1.60 FIRS exceeds the site-specific FIRS in the frequency range below approximately 6 Hz in the horizontal direction. The maximum exceedance is a factor of approximately 1.60 in the 2.5 Hz range.

The staff reviewed the FRS comparisons in LNP FSAR Table 3.7-205 for predominate frequencies and noted that the maximum exceedance was a factor of 1.43 at three key locations (nodes 1761-X, 2675-Y, and 3329-Y). The predominant frequency for all three locations (in the respective directions) is 3 Hz. The staff compared the LNP FRS at these locations and found the margin to the AP1000 CSDRS FRS to be greater than the factor of 1.43. Based on the information above, the staff concludes that although the RG 1.60 FIRS exceed the site-specific FIRS in the low-frequency range, the corresponding LNP FRS remain enveloped by the AP1000 standard design FRS. On this basis, the staff finds the conclusions regarding the applicant's site-specific analysis, described in Section 3.7.2 of this SER, remain valid. The applicant's consideration of the minimum RG 1.60 FIRS on the RCC bridging mat design and seismic interaction effects are discussed below.

20.1.4.6.4 RCC Bridging Mat Design

The RCC bridging mat is a site-specific seismic Category I structure. The purposes of the RCC bridging mat are to replace the weakly cemented, undifferentiated Tertiary sediments that are present above elevation -7.3 m (-24 ft.) NAVD88, thereby creating a uniform subsurface with increased bearing capacity, and to bridge conservatively postulated karst features. In LNP FSAR Section 3.7.2.4, the applicant stated that the RCC bridging mat is designed for the soft rock site condition considered in the AP1000 standard design. The seismic demands are based on the AP1000 CSDRS, with a PGA of 0.3 g, not on the LNP site-specific demands. Staff evaluation of the RCC bridging mat design is described in Section 3.8 of this SER.

In LNP FSAR Section 3.7.2.4.1.7, the applicant described the approach for assessing the impact of the RG 1.60 minimum spectrum on the RCC bridge mat. The applicant stated that the

conceptual design of the RCC bridging mat is based on a bearing pressure of 8.9 kips per square foot (ksf) for static loading and 24.0 ksf for dynamic loading. In addition, a base shear load of 136,000 kips based on the AP1000 generic analyses was applied at the top of the RCC bridging mat. The static bearing pressure is based on design control document (DCD) Revision 19 Tier 1 Table 5.0.1. The dynamic bearing pressure is the maximum subgrade pressure at the AP1000 basemat that results from the generic AP1000 analysis for soft rock sites. The applicant concluded that because the AP1000 generic site analyses are based on the CSDRS with a 0.3 g PGA, which impose greater seismic demands than the RG 1.60 FIRS with a 0.1 g PGA, the design of the RCC bridging mat is conservative.

The staff compared the LNP FIRS to the AP1000 CSDRS and finds significant margin (more than a factor of 2) between the LNP FIRS and the CSDRS. Accordingly, the staff finds the seismic demands used for the RCC bridging mat design to be conservative for the LNP site and to satisfy the requirements of Appendix S to 10 CFR Part 50. On this basis, the staff finds that the conclusions regarding the RCC bridge mat design, described in Section 3.8 of this SER, remain valid.

20.1.4.6.5 Seismic Category I and Category II Interactions

For the LNP site, the seismic Category II and nonseismic structures adjacent to the nuclear island are supported on drilled shaft foundations. LNP FSAR Section 3.7.2.8 describes the applicant’s evaluation of seismic interaction between the nuclear island and adjacent buildings. The LNP PBSRS were used to compute the maximum relative displacements of the annex, turbine, and radwaste buildings’ drilled shaft foundations and to evaluate the seismic interaction. The maximum relative displacement calculation included the drilled shaft supported foundation mat displacements, drilled shaft interaction effects, additional displacements caused by soil column displacements, and the nuclear island (NI) displacement at design grade. The staff’s evaluation of the seismic interaction between the Category II structures, the nonseismic structures, and the nuclear island is described in Section 3.7 of this SER.

In LNP FSAR Section 3.7.2.8, the applicant assessed the effect of the RG 1.60 FIRS (applied at the surface) on the relative displacements between the seismic Category II structures and the nuclear island. The applicant’s analysis showed that the computed maximum displacements between the nuclear island and the adjacent structures were all greater for the site-specific PBSRS. LNP FSAR Table 3.7-206 indicates that the maximum relative displacement is 1.8 cm (0.7 in) and occurs between the seismic Category II portion of the annex building and the nuclear island. The applicant concluded that the maximum relative displacement of 1.8 cm (0.7 in) is less than the design gap of 5.0 cm (2 in) provided in the AP1000 standard plant design.

The staff reviewed the applicant’s analysis results provided in LNP FSAR Table 3.7-206 and finds the maximum relative displacements to be less than the AP1000 design gap described in DCD Section 3.8. The staff finds that the information provided by the applicant is sufficient to demonstrate that the seismic gaps, provided in the standard design, are adequate to prevent interaction between the NI and the adjacent structures at the LNP site. On this basis, the staff finds that the conclusions regarding seismic interaction of seismic Category I and non-Category I structures, described in Section 3.8 of this SER, remain valid.

20.1.4.6.6 Conclusions of Structural Seismic Evaluation

The NRC staff has reviewed the applicant's response to NRC RAI Letter No. 108, dated August 1, 2012. Based on the staff's technical evaluation, the staff concludes that:

1. The LNP site-specific soil-structure interaction analysis results are conservatively bounded by the standard plant analysis results and are not affected by the ground motion developed using the CEUS SSC model.

2. The sensitivity study performed for the LNP FRS, which considers a RG 1.60 minimum spectrum, demonstrates that the LNP FRS remain bounded by the CSDRS FRS.

3. The sensitivity study performed by the applicant demonstrates that there is no effect on the analysis results for the site-specific structural features of the LNP plant, including the RCC bridging mat under the nuclear island, the drilled shaft foundation supporting the buildings adjacent to the nuclear island, and the potential seismic interaction between the nuclear island and the adjacent structures.

4. For the seismic Category 1 buildings, the site-specific features such as the RCC bridging mat are designed to support seismic demands consistent with the AP1000 certified design demands, which exceed the site-specific demands at the LNP site with a substantial margin. For the non-seismic Category 1 structures, the site-specific features such as the drilled shaft foundations are designed to support seismic demands consistent with the site-specific demands at the LNP site.

20.1.4.7 Site-Specific Seismic Margins Analysis

20.1.4.7.1 AP1000 Design Seismic Margin

The applicant's evaluation of seismic margin for the LNP site is described in FSAR Section 19.55, "Seismic Margin Analysis." The NRC staff reviewed LNP COL FSAR Section 19.55, which incorporated Section 19.55 of the DCD with no departures or supplements.

The staff review found that the GMRS for the LNP site (presented in LNP COL FSAR Figure 2.5.2-296) are bounded by the CSDRS evaluated in the AP1000 DCD. The PBSRS were developed and are also bounded by those of the certified design. The applicant performed other analyses, including analysis of soil-structure interaction, to confirm that site-specific features did not cause the high confidence in low probability of failure (HCLPF) values reported in the DCD (seismic capacity) to fall below the values developed for the certified design. The staff finds that using the seismic margins analysis (SMA) provided in the DCD is conservative and acceptable for all structures, systems, and components within the scope of the DCD.

The applicant also provided supplemental information on the HCLPF value of the seismic Category I, RCC bridging mat and its effect upon the SMA. The staff found the applicant's evaluation to be consistent with the guidance in DC/COL-ISG-20, and therefore acceptable. The staff's evaluation of the structures, systems, and components within the scope of the DCD and seismic Category I RCC bridge mat is described in Section 19.55 of this SER.

20.1.4.7.2 Fukushima RAI

On March 15, 2012, the staff issued NRC Letter No. 108 requesting additional information concerning the implementation of the Fukushima Near-Term Task Force recommendations, including Recommendation 2.1 regarding reevaluation of seismic hazards. The RAI requested the applicant to evaluate its plant-specific seismic hazards against the current NRC requirements and guidance and, if necessary, update the design basis and structures, systems, and components important to safety to protect against the updated hazards.

In response to RAI Letter No. 108, the applicant compared the FIRS developed using the EPRI-SOG and CEUS SSC methods. The applicant concluded that the site-specific FIRS developed using the EPRI-SOG method and scaled to 0.1 g PGA envelop the FIRS developed using the CEUS SSC method. Based on these results, the applicant concluded that the findings pertaining to seismic margin analysis for the standard plant components, site liquefaction potential, adjacent buildings seismic interaction with the nuclear island, and the RCC bridging mat capacity for the CEUS SCC methodology ground motions are bounded by that for the EPRI-SOG methodology.

The staff performed a review of the applicant's RAI response to assess the impact on the conclusions pertaining to the seismic margin of seismic Category I structures described in Section 19.55 of this SER. The staff also addressed two seismic margin areas that are not addressed in Section 19.55 of this SER; namely site-specific differential displacement of seismic Category II and nonseismic structures and liquefaction potential.

20.1.4.8 Seismic Category I Structures

For the LNP site, the seismic Category I structures are the AP1000 nuclear island and the RCC bridging mat. The purposes of the RCC bridging mat are to replace the weakly cemented, undifferentiated Tertiary sediments that are present above elevation -7.3 m (-24 ft.) NAVD88, thereby creating a uniform subsurface with increased bearing capacity, and to bridge conservatively postulated karst features. In LNP FSAR Section 3.7.2.4, the applicant describes that the RCC bridging mat is designed for the soft rock site considered in the standard design and that the seismic demands are based on the AP1000 CSDRS. Section 3.8 of this SER describes the staff's evaluation of the RCC bridging mat design.

The staff performed a review of the applicant's FIRS comparisons, shown in LNP FSAR Figure 2.5.2-358, and finds that the horizontal and vertical FIRS developed from the CEUS SSC method are enveloped by the FIRS developed from the EPRI-SOG method and scaled to 0.1 g PGA for the full range of frequencies (0.1 to 100 Hz).

The staff also reviewed the applicant's comparisons of GMRS and PBSRS to the AP1000 CSDRS as depicted in LNP FSAR Figures 2.5.2-355 and 2.5.2-357. The staff concludes that the CSDRS demands envelop both the LNP GMRS and PBSRS at all frequencies. On this basis, the staff finds that the conclusions remain valid regarding the LNP seismic margin analysis of seismic Category I structures (as described in Section 19.55 of this SER).

20.1.4.8.1 Seismic Category II and Nonseismic Structures

For the LNP site, the seismic Category II and nonseismic structures adjacent to the nuclear island are supported on drilled-shaft-supported mat foundations. The PBSRS are used to compute the maximum relative displacements of the annex, turbine, and radwaste building foundations and to evaluate the site-specific seismic interaction of these buildings with respect to the nuclear island. LNP FSAR Section 3.7.2.8.4 describes the applicant's analysis of relative building displacements. The applicant's approach involved using input ground motion based on UHRS corresponding to a return period of 1×10^{5} years (10^{-5} UHRS).

Based on this input ground motion, the applicant concluded that the maximum relative displacement between the nuclear island and the annex, turbine, and radwaste building foundations was less than 2.54 cm (1 in). The applicant concluded this difference to be less than the design gap of 5.08 cm (2.0 in) specified in DCD Section 3.8.5.

The staff based its review of the applicant's seismic margin assessment of the LNP seismic Category II and nonseismic structures on guidance in DC/COL-ISG-20. DC/COL-ISG-20 states that the plant-specific plant-level HCLPF should be demonstrated to be equal to or greater than 1.67 times the site-specific GMRS (or 1.67 times the site-specific PBSRS in the case of a surface founded structure).

The staff reviewed LNP FSAR Figure 3.7-229, which compares the horizontal 10^{-5} UHRS based on the EPRI-SOG model and LNP PBSRS (multiplied by a factor of 1.67), and found the 10^{-5} UHRS ground motion to envelop the LNP PBSRS in accordance with ISG/COL-ISG-20.

The staff finds the maximum displacement (less than 2.54 cm or 1 in) of the annex, turbine, and radwaste buildings relative to the nuclear island to be less than the AP1000 DCD design gap (5.08 cm or 2.0 in) described in DCD Section 3.8.5. Accordingly, the staff finds that the information provided by the applicant is sufficient to demonstrate that the seismic gaps are adequate to prevent interaction between the nuclear island and the adjacent structures under beyond-design-basis loading. Based on the above, the staff finds the applicant's evaluation of seismic margin of seismic Category II structures to be consistent with the guidance in DC/COL-ISG-20, and therefore acceptable.

20.1.4.8.2 Liquefaction

LNP FSAR Sections 2.5.4.8.4 and 2.5.4.8.5, respectively, describe the LNP site-specific analysis of earthquake-induced cyclic stress within soils considered for liquefaction evaluation computed by the SHAKE program based on the 60 randomized soil profiles used to develop the PBSRS. The staff found the evaluation of liquefaction potential to be consistent with RG 1.198, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites," and SRP Section 2.5.4, "Stability of Subsurface Materials and Foundations," and therefore acceptable. The staff's evaluation of these FSAR Sections is described in Section 2.5.4.4.8 of this SER.

DC/COL-ISG-20, "Seismic Margin Analysis for New Reactors Based on Probabilistic Risk Assessment," states that the seismic margin analysis should consider site-specific effects such as soil liquefaction. Soil liquefaction is defined as a fluid-induced loss of soil strength with two

typical failure modes: (1) flow failure, in which the shear strength of the soil drops below the level needed to maintain stability, and (2) cyclic mobility failure (lateral spread). Either failure mode can lead to excessive strains and displacement that could result in unacceptable performance of supported structures, systems, and components.

For the purpose of seismic margins analysis, the applicant also assessed liquefaction potential for ground motions in excess of the site responses corresponding to the GMRS and PBSRS. In LNP FSAR Section 2.5.4.8.7, the applicant stated that the analysis of liquefaction potential of soils under the annex, turbine, and radwaste buildings is based on ground motions consistent with EPRI-SOG 10^{-5} UHRS. The updated EPRI-SOG plant finished grade 10^{-5} UHRS envelops both 1.67 x GMRS and 1.67 x PBSRS developed using the CEUS SSC methodology and modified CAV filter.

The staff reviewed LNP FSAR Figures 3.7-228 and 3.7-229, which compare the LNP horizontal 10^{-5} UHRS with the LNP GMRS multiplied by a factor of 1.67 and the LNP PBSRS multiplied by a factor of 1.67, respectively. This review found the 10^{-5} UHRS ground motion to envelop the LNP 1.67 x GMRS and 1.67 x PBSRS with margin. Based on this finding, the NRC staff concludes that the applicant's assumed ground motion for seismic margin considerations is conservative.

In LNP FSAR Section 2.5.4.8.6, the applicant described sensitivity analysis of the median centered liquefaction potential for 10^{-5} UHRS performed to assess whether the liquefiable zones under LNP 1 and 2 footprints are confined to the northwest corner of the LNP Unit 2 turbine building and in isolated pockets under the remaining LNP Units 1 and 2 footprints. The applicant's method and design parameters were the same as those used for design-basis liquefaction analysis. LNP FSAR Tables 2.5.4.8-203A and 2.5.4.8-203B present the results of the assessment and indicate where liquefaction is postulated. The applicant concluded that the analysis results based on median centered liquefaction potential for updated EPRI-SOG 10^{-5} UHRS are the same as those for the design-basis liquefaction analysis.

The staff based its review of the applicant's median centered liquefaction evaluation for 10^{-5} UHRS on guidance in DC/COL-ISG-20 and RG 1.198. The staff compared the liquefaction analysis results for 10^{-5} UHRS for LNP Units 1 and 2 presented in LNP FSAR Tables 2.5.4.8-203A and 2.5.4.8-203B with the results for design basis for LNP Units 1 and 2 presented in LNP FSAR Tables 2.5.4.8-202A and 2.5.4.8-202B. The staff confirmed that the locations and elevations of hypothesized liquefaction (computed factors of safety against liquefaction, FS ≤1.0 for 10^{-5} UHRS sensitivity analysis and FS ≤1.1 for design basis) are almost identical. The NRC staff, therefore, concurs that liquefiable zones under the LNP Units 1 and 2 footprints are confined to the northwest corner of the LNP Unit 2 turbine building and in isolated random pockets under the remaining LNP Units 1 and 2 footprints. The staff notes that LNP FSAR Section 2.5.4.8.5 describes design features intended to mitigate the effects of liquefaction below the turbine building. The applicant stated that for the area under the Annex, Turbine, and Radwaste building footprint, in situ soil will be replaced or improved to a depth of approximately 2.1 m (7 ft.) below existing grade (elevation 12.8 m [42 ft.] NAVD88). The plant design grade will be established at elevation 15.5 m (51 ft.) NAVD88 by placing engineered fill above the improved / replaced in situ material. In addition, the earthwork design incorporates horizontal and vertical drains to relieve pore pressure. The staff also notes that the northwest corner of

LNP Unit 2 turbine building is opposite the end of the building with the seismic Category II bay. As such, the effects of localized liquefaction do not affect the analysis of seismic interaction.

Based on the above review, the staff finds the applicant's liquefaction analysis methodology and design parameters to be consistent with ISG-20 and RG 1.198, and therefore acceptable.

20.1.4.8.3. Conclusions for Site-Specific Seismic Margin Evaluation

The NRC staff has reviewed the applicant's response to NRC Letter No. 108 dated August 1, 2012. Based on the staff's technical evaluation of the response, the staff concludes that:

1. The findings regarding the LNP seismic margin analysis of the seismic Category I AP1000 nuclear island structures and the RCC bridging mat, as described in Section 19.55 of this SER, remain valid.

2. The applicant's seismic margin analysis of LNP seismic Category II and nonseismic structures is consistent with the guidance in DC/COL-ISG-20, and therefore is acceptable.

3. The applicant's evaluation of beyond-design-basis liquefaction potential is consistent with the guidance in DC/COL-ISG-20 and RG 1.198, and therefore is acceptable.

20.1.4.8.4 Conclusions on CEUS SSC Sensitivity Evaluation

The NRC staff has reviewed the applicant's response to RAI Letter No. 108 (ADAMS Accession No. ML120550146). Based on the staff's technical evaluation, the staff concludes that:

1. The applicant demonstrated the ability to perform accurate hard rock seismic hazard calculations using the CEUS SSC model by comparing and matching the results of hazard analysis at the seven test site locations described in NUREG-2115.

2. The applicant accurately calculated the LNP site-specific UHRS, GMRS, FIRS, and PBSRS using the CEUS SSC model and, where applicable, implemented the updated CAV filter methodology, as recommended in SECY-12-0025 Enclosure 7, Attachment 1, to Seismic Enclosure 1 (ADAMS Accession No. ML12039A188).

3. The LNP site-specific UHRS, GMRS, FIRS, and PBSRS based on the use of CEUS SSC model are either bounded by those respective spectra calculated by the applicant using the updated EPRI-SOG model, or are within a range of percentage error expected for those calculations. Therefore, it is not necessary for the applicant to update the UHRS, GMRS, FIRS, and PBSRS calculated using the updated EPRI-SOG model.

4. The applicant performed liquefaction potential analysis based on the CEUS SSC ground motion estimates and demonstrated that they are bounded by the EPRI-SOG ground motion estimates. Therefore, the NRC staff concludes that the liquefaction evaluations

in FSAR Section 2.5.4.8 correctly and conservatively estimate earthquake-induced liquefaction potential.

5. The updated site-specific FIRS have no impact on the results and conclusions of the structural seismic evaluations performed by the applicant to demonstrate the adequacy of the AP1000 standard plant at the LNP site. Consequently, the NRC staff concludes that there is reasonable assurance that the requirements of 10 CFR Part 50, Appendix A, GDC 2; 10 CFR Part 50, Appendix S; and 10 CFR Part 52, Appendix D, Section VIII B6, continue to be satisfied.

6. The updated site-specific FIRS have no effect on the results and conclusions of seismic margins evaluations performed by the applicant to demonstrate the adequacy of the AP1000 standard plant at the LNP site. Consequently, the NRC staff concludes that there is reasonable assurance that the requirements, as described in Section 19.55.3 of this SER, continue to be satisfied.

20.1.5 Post Combined License Activities

There are no post COL activities related to this section.

20.1.6 Conclusion

The NRC staff reviewed the information submitted by the applicant in response to SECY-12-0025 regarding seismic hazard reevaluation. The staff confirmed that the applicant has addressed the required information and has adequately evaluated the seismic hazards at the LNP COL site against current NRC requirements and guidance – 10 CFR 100.23; 10 CFR 52.79 (a)(1)(iii); 10 CFR Part 50, Appendix A, GDC 2; Public Law 112-74, Section 402; 10 CFR Part 50, Appendix S; 10 CFR Part 52, Appendix D, Section VIII B.6; NUREG-0800, RGs 1.60, 1.132, 1.198, 1.206, 1.208; DC/COL ISG-017; and DC/COL ISG-020.

20.2 Mitigation Strategies for Beyond-Design-Basis External Events (Based on Recommendation 4.2)

20.2.1 Introduction

NRC Commission Paper SECY-12-0025 states that the NRC staff will request all COL applicants to provide the information required by the orders and request for information letters described in SECY-12-0025, as applicable, through the review process. For mitigation strategies for beyond-design-basis external events, SECY-12-0025 outlined a three-phase approach for mitigating beyond-design-basis external events. The initial phase involves the use of installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling without alternating current (ac) power. The transition phase involves providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from offsite. The final phase involves obtaining sufficient offsite resources to sustain those functions indefinitely.

SECY-12-0025 notes that the AP1000 standard design (which is incorporated by reference in the LNP COL application) includes passive design features that provide core cooling, containment, and SFP cooling capabilities for 72 hours, without reliance on ac power. The AP1000 design also includes equipment to maintain required safety functions in the long term (beyond 72 hours to 7 days). As such, provisions related to the final phase must be addressed.

NRC Interim Staff Guidance (ISG) JLD-ISG-2012-01, Revision 0 (ADAMS Accession No. ML12229A174), “Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events,” endorses with clarifications, the methodologies described in the industry guidance document, Nuclear Energy Institute (NEI) 12-06 (ADAMS Accession No. ML12242A378), “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,” Revision 0. JLD-ISG-2012-01 describes an acceptable approach for developing mitigation strategies for beyond-design-basis external events at nuclear power plants based on the guidance in NEI 12-06.

20.2.2 Summary of Application

The LNP Final Safety Analysis Report (FSAR) provides information on systems used to establish and sustain core cooling, containment, and SFP cooling capabilities for the LNP. For example, Section 6.3, “Passive Core Cooling System,” of the FSAR discusses the passive core cooling system (PXS), which provides emergency core cooling following postulated design-basis events, and incorporates by reference Section 6.3 of the AP1000 DCD Tier 2 with identified departures and supplements. FSAR Section 6.2, “Containment Systems,” and Section 9.1, “Fuel Storage and Handling,” address containment systems and fuel storage and handling systems, respectively, and incorporate by reference Section 6.2.2, “Passive Containment Cooling System,” and Section 9.1.3, “Spent Fuel Pool Cooling System,” of the AP1000 DCD Tier 2.

In SECY-12-0025, the NRC staff indicated its intent to review information provided by COL applicants to describe their mitigation strategies for beyond-design-basis external events. In light of SECY-12-0025, the staff issued RAI Letter No. 108, dated March 15, 2012, to request information regarding the LNP mitigation strategies to sustain core cooling, containment, and SFP cooling capabilities functions indefinitely.

The applicant provided an initial response to the RAI in a letter dated September 27, 2012 (ADAMS Accession No. ML12272A318). In its initial response, the LNP COL applicant proposed a license condition related to mitigation strategies for beyond-design-basis conditions resulting from an extended loss of ac power and loss of access to the normal heat sink (referred to below as an ELAP event). Subsequent to that response, the applicant provided the NRC staff with the general mitigation strategy that will be used by LNP, including the strategies for initial (0 to 72 hours) mitigation, in a letter dated April 22, 2015 (ADAMS Accession No. ML15114A359). The letter, which was Supplement 9 to the LNP response to RAI Letter No. 108, provided the staff with a Westinghouse report (designated as APP-GW-GLR-171,

"AP1000 Flex Integrated Plan," for the publicly available version) that included a description of the mitigating strategies for beyond-design-basis external events that will be applied at LNP.

In Item 12, "Fukushima Response Actions," of Part 10, "Proposed License Conditions (including inspection, test, analysis, and acceptance criteria (ITAAC))," of the LNP COL application, the applicant proposed a license condition related to this subject.

20.2.3 Regulatory Basis

The requirements and guidance for mitigation strategies for beyond-design-basis external events are established or described in the following:

- Atomic Energy Act of 1954, as amended, § 161, authorizes the Commission to regulate the utilization of special nuclear material in a manner that is protective of public health and in accord with the common defense and security.

- 10 CFR 52.97(a)(1), which authorizes the Commission to issue a COL if it finds, among other things, that issuance of the license will not be inimical to the health and safety of the public. This regulation applies here because the Commission found in Order EA-12-049 that it is necessary for power reactor licensees to develop, implement and maintain guidance and strategies to restore or maintain core cooling, containment, and SFP cooling capabilities in the event of a beyond-design-basis external event in order to ensure adequate protection of the public health and safety.

- SRM-SECY-12-0025, "Staff Requirements – SECY-12-0025 – Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated March 9, 2012, approves issuance of orders for beyond-design-basis external events, as necessary for ensuring continued adequate protection under the 10 CFR 50.109(a)(4)(ii) exception to the Backfit Rule.

- JLD-ISG-2012-01, Revision 0, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," issued August 29, 2012, endorses NEI 12-06, Revision 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" (issued August 21, 2012), with exceptions/clarifications.

- Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012. Although Order EA-12-049 does not apply to LNP Units 1 and 2, the staff followed the current NRC and industry guidance for establishing mitigation strategies for beyond-design-basis external events at AP1000 reactors in evaluating the equipment used as part of the mitigation strategy for LNP Units 1 and 2.

20.2.4 Technical Evaluation

The NRC staff reviewed the information submitted by the LNP COL applicant regarding its proposed mitigation strategies for beyond-design-basis conditions resulting from an ELAP event. To assess whether the proposed mitigation strategies provided an acceptable approach, the staff applied JLD-ISG-2012-01, Revision 0, which endorses, with clarifications, the methodologies described in industry guidance document NEI 12-06, Revision 0. Appendix F, "Guidance for AP1000 Design," to NEI 12-06 outlines the process to be used by AP1000 COL licensees and applicants to define and implement the mitigation strategies for beyond-design-basis conditions resulting from an ELAP event.

In Section 7.0, "Guidance for AP1000 Design," of JLD-ISG-2012-01, the NRC staff states that the guidance in Appendix F of NEI 12-06 provides an acceptable means to meet the requirements of Order EA-12-049 or license conditions imposing similar requirements for the AP1000 reactor design. Appendix F to NEI 12-06 specifies that the underlying strategies for coping with ELAP events for AP1000 plants involve a three-phase approach as follows:

1. Initial coping through installed plant equipment without ac power or makeup to the ultimate heat sink. From 0 to 72 hours, the certified AP1000 design includes passive systems that provide core cooling, containment, and SFP cooling.

2. Following the 72-hour passive system coping time, support is necessary to continue passive system cooling. From 3 to 7 days, this support can be provided by installed plant ancillary equipment or by offsite equipment installed to connections provided in the AP1000 design.

3. To extend the passive system cooling time beyond 7 days to an indefinite time, offsite assistance is necessary, such as the delivery of diesel fuel oil. Appendix F includes provisions related to the qualification and use of equipment intended to mitigate an ELAP event.

As mentioned in Appendix F to NEI 12-06, APP-GW-GLR-171, referenced above, indicates that core cooling, containment, and SFP cooling is provided for the initial time period of 0 to 72 hours through installed, safety-related plant equipment that is part of the certified design. These systems do not rely on ac power or on access to any external water sources, because the containment vessel and the passive containment cooling system serve as the safety-related ultimate heat sink. The NRC staff reviewed and found acceptable the site-specific functional design, qualification, and inservice testing program descriptions for this safety-related equipment for LNP Units 1 and 2 as discussed in the applicable sections of this report.

Following the initial 72-hour coping period, APP-GW-GLR-171 indicates that support is necessary to continue passive system cooling, and this support can be provided by installed ancillary equipment or by offsite equipment interfacing with installed plant connections. For example, additional inventory for the passive containment cooling system (PCS) and SFP can be supplied from the onsite passive containment cooling ancillary water storage tank (PCCAWST) using the onsite PCS recirculation pumps, powered using the onsite ancillary diesel generators or offsite replacement generators. The installed ancillary equipment and stored cooling water are capable of supporting passive system cooling from 3 days after the

event to 7 days after the event. Beyond this time period, the report indicates that offsite assistance and resources are needed. For indefinite coping after 7 days, an offsite pump (PCCAWST makeup pump) and appropriate connection materials to refill the PCCAWST from the closest water source will be provided. In the event that the PCS recirculation pumps are unavailable, a second self-powered, offsite pump (PCS/SFP makeup pump) and appropriate connection materials will be available.

APP-GW-GLR-171 also includes several additional provisions related to the qualification and use of commercially procured equipment that will be used 72 hours after an ELAP event:

- Programmatic controls for this equipment include quality attributes, equipment design, equipment storage, procedure guidance, maintenance, testing, training, staffing, and configuration control.

- The quality assurance (QA) provisions in AP1000 DCD Tier 2, Table 17-1, “Quality Assurance Program Requirements for Systems, Structures, and Components Important to Investment Protection,” will be applied to this AP1000 FLEX equipment.

- The graded approach to availability and testing as shown in AP1000 DCD Tier 2, Section 16.3, “Investment Protection,” will be applied to the FLEX equipment.

- The design and maintenance of the FLEX equipment will be in accordance with Section 11.2, “Equipment Design,” and Section 11.5, “Maintenance and Testing,” respectively, of NEI 12-06.

- AP1000 DCD Tier 2, Section 1.9.5.4, “Additional Licensing Issue – Post-72 Hour Support Actions,” describes procedures that address actions that would be necessary 72 hours subsequent to an ELAP event to maintain core, containment, and SFP cooling for an indefinite period of time.

The NRC staff reviewed the applicable sections of the LNP FSAR, along with their respective AP1000 DCD sections, the final safety evaluation report (FSER) for the AP1000 design certification, and other sections of this report to verify the above information. For example, Table 8.1-201, “Site-Specific Guidelines for Electric Power Systems,” in the LNP FSAR indicates that station blackout is addressed as a design issue in the AP1000 DCD. The staff reviewed station blackout as part of its review of Chapter 8 of the AP1000 DCD Tier 2. Section 8.5.2.1, “Station Blackout,” of the AP1000 FSER states that the AP1000 safety-related passive systems automatically establish and maintain safe-shutdown conditions for the plant following design-basis events, including the loss of ac power sources, and the passive systems can maintain these safe-shutdown conditions after design-basis events for 72 hours, without operator action, following a loss of both onsite and offsite ac power sources. The staff reviewed the applicability of this FSER conclusion to LNP.

Section 8.3.2, “Direct Current Power and Uninterruptible Power Systems” of the AP1000 FSER, Supplement 2, states that Class 1E batteries will be sized adequately to perform their safety functions as designed and that ITAAC verifying that the batteries are adequately designed are identified in AP1000 DCD Tier 1, Table 2.6.3-3. APP-GW-GLR-171 discusses the connections

for the onsite ancillary diesel generators and the offsite portable generators. Electrical isolation between safety related power systems and power sources utilized in Phase 3 is addressed in APP-GW-GLR-171, which states that voltage regulating transformers are the connection point for the offsite portable generators. Section 8.3.2, “Direct Current Systems” of this document discusses how the voltage regulating transformer in combination with fuses and/or breakers will interrupt the input or output (ac) current under faulted conditions to achieve electrical isolation. As part of the license condition, part (c), as set forth in Section 20.2.5 of this SER, the capacity of the offsite portable generators will be assessed by DEF to ensure they are capable of providing power to the necessary loads described in AP1000 DCD Tier 2 Table 8.3.1-4, “Post-72 hours nominal load requirements.” Section 9.5.3 of this document addresses plant lighting systems, specifically emergency lighting which provides illumination in areas where emergency operations are performed.

Emergency core cooling for the LNP is accomplished using the AP1000 PXS, which is described in Section 6.3 of the AP1000 DCD Tier 2. The LNP FSAR specifies that Section 6.3 of the AP1000 DCD Tier 2 was incorporated by reference with identified departures. The staff reviewed LNP FSAR Section 6.3, and found that the departures have no impact on the capability of the PXS to establish and maintain safe-shutdown conditions for 72 hours following a loss of both onsite and offsite ac power sources. Therefore, core cooling for the initial phase (0 to 72 hours) of mitigation for LNP will be accomplished by its safety-related PXS, per the LNP licensing basis.

The mitigation of a station blackout, as required by 10 CFR 50.63, addresses the capability of a nuclear power plant to provide adequate core cooling during a loss of ac power. In addition to core cooling, the recommendations for mitigation strategies for beyond-design-basis external events also address containment function, and SFP cooling.

The control of containment pressure and temperature for LNP is accomplished using the AP1000 PCS, which is described in Section 6.2.2, “Passive Containment Cooling System,” of the AP1000 DCD Tier 2. In its review of the LNP FSAR, the staff found, with the exception of a departure related to the containment leak rate test program, that Section 6.2.2 of the AP1000 DCD Tier 2 was incorporated by reference into the LNP FSAR. In Section 6.2.2 of the AP1000 FSER, the staff stated the principal design basis for the PCS is to maintain the containment internal pressure below the design value for 3 days following a design-basis accident. The staff review, as documented in Section 6.2.1.1, “Containment Pressure and Temperature Response to High-Energy Line Breaks,” of the AP1000 FSER, found that the PCS met its design objectives. Therefore, the containment function for the initial phase of (0 to 72 hours) mitigation for LNP will be accomplished by its safety-related PCS per the LNP licensing basis.

The SFP cooling function for the LNP is accomplished by maintaining sufficient water inventory in the SFP to keep the fuel covered and, therefore, provide the necessary cooling in the event of an extended loss of SFP cooling due to the loss of ac power. In Section 9.1.3.2.3, “Increase in Number of Spent Fuel Storage Locations,” in Supplement 2 of the AP1000 FSER, the staff concluded that the SFP will maintain water coverage above the spent fuel assemblies for at least 72 hours following a loss of nonsafety-related SFP cooling, using only safety-related makeup water. Therefore, initial phase mitigation is accomplished through passive means. However, as indicated in Note 9 in the DCD Tier 2 Table 9.1-4, “Station Blackout/Seismic Event

Times," for the most limiting scenario (full core offload) operator action must occur at approximately 18 hours after the event. In Attachment 1, "Sequence of Events Timeline," to the AP1000 FLEX integrated plan, this action has been identified and the appropriate procedure cited to assure the task is performed. Hence, SFP cooling for the initial phase (0 to 72 hours) of mitigation for LNP will be accomplished by passive cooling of the SFP in accordance with the LNP licensing basis.

The NRC staff has reviewed the mitigation strategies for beyond-design-basis external events for LNP based on the information provided by the LNP COL applicant, including referenced mitigation guidance for beyond-design-basis external events applicable to AP1000 reactors. The staff finds that the LNP COL applicant has provided or referenced information to describe its mitigation strategies for beyond-design-basis external events in an acceptable manner. The staff recognizes that full implementation of the mitigation strategies for beyond-design-basis external events at AP1000 reactors cannot be established until after licensing (e.g., during procedure development). The staff prepared a license condition for implementation of the mitigation strategies for beyond-design-basis external events at LNP Units 1 and 2, based on the applicant's proposed license condition with specific enhancements to provide consistency with current NRC staff expectations. Completion of the activities associated with the license condition, including lessons learned from initial AP1000 implementation, can be verified through NRC inspection activities.

20.2.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following license condition related to the mitigation strategies program:

> License Condition (20-1) – Mitigation Strategies for Beyond-Design-Basis External Events:
>
> a. The Licensee shall complete development of an overall integrated plan of strategies to mitigate a beyond-design-basis external event at least 1 year before the completion of the last ITAAC on the schedule required by 10 CFR 52.99(a).
>
> b. The overall integrated plan required by this condition must include guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities. The overall integrated plan must include provisions to address all accident mitigation procedures and guidelines (including the guidance and strategies required by this section, emergency operating procedures, abnormal operating procedures, and extensive damage management guidelines).
>
> c. The guidance and strategies required by this condition must be capable of (i) mitigating a simultaneous loss of all alternating current (ac) power and loss of normal access to the normal heat sink and (ii) providing for adequate capacity to perform the functions upon which the guidance and

strategies rely for all units on the Levy site and in all modes at each unit on the site.

d. Before initial fuel load, the Licensee shall fully implement the guidance and strategies required by this condition, including:

   1. Procedures;

   2. Training;

   3. Acquisition, staging, or installation of equipment and consumables relied upon in the strategies; and

   4. Configuration controls and provisions for maintenance and testing (including testing procedures and frequencies for preventative maintenance) of the equipment upon which the strategies and guidance required by this condition rely.

e. The training required by condition d.2 must use a Systematic Approach to Training (SAT) to evaluate training for station personnel, and must be based upon plant equipment and procedures upon which the guidance and strategies required by this Condition rely.

f. The Licensee shall maintain the guidance and strategies described in the application upon issuance of the license, and the integrated plan of strategies upon its completion as required by condition a. The Licensee may change the strategies and guidelines required by this Condition provided that the Licensee evaluates each such change to ensure that the provisions of conditions b and c continue to be satisfied and the Licensee documents the evaluation in an auditable form.

20.2.6 Conclusion

The NRC staff reviewed the mitigating strategies for LNP to provide assurance of core cooling, containment, and SFP cooling capabilities in the event of a beyond-design-basis external event resulting in an ELAP event. The staff finds that the approach for mitigating beyond-design-basis external events to be used at LNP is consistent with NRC Order EA-12-049 and both general and AP1000-specific NRC guidance (including NEI 12-06, Appendix F, as endorsed by the NRC staff). Therefore, the staff concludes that the mitigating strategies for beyond-design-basis external events described for LNP are acceptable. The staff will impose a license condition as discussed in this SER section to verify the implementation of the mitigation strategies for beyond-design-basis external events at LNP Units 1 and 2 as described in the specified documentation.

20.3 Reliable Spent Fuel Pool Instrumentation (Based on Recommendation 7.1)

20.3.1 Introduction

During the events in Fukushima, responders were without reliable instrumentation to determine the water level in the spent fuel pool (SFP). This caused concerns that the pool may have boiled dry, resulting in fuel damage, and highlighted the need for reliable SFP instrumentation. The SFP level instrumentation at United States (U.S.) nuclear power plants is typically narrow range and, therefore, only capable of monitoring normal and slightly off-normal conditions. Although the likelihood of a catastrophic event affecting nuclear power plants and the associated SFPs in the U.S. remains very low, beyond-design-basis external events could challenge the ability of existing spent fuel pool instrumentation in providing emergency responders with reliable information on the condition of SFPs. Reliable and available indication is essential to ensure plant personnel can effectively prioritize emergency actions.

SECY-12-0025, Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami” states that the staff will request all combined license (COL) applicants to provide the information required by the orders and request for information letters described in SECY-12-0025, as applicable, through the review process. With regard to Recommendation 7.1 for reliable spent fuel pool instrumentation, SECY-12-0025 notes that the AP1000 standard design includes two permanently fixed safety related level instruments with the capability for a third instrument connection.

JLD-ISG-2012-03, Revision 0, “Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation,” (ADAMS Accession No. ML12221A339), endorses with exceptions and clarifications the methodologies described in the industry guidance document, NEI 12-02, Revision 1, “Industry Guidance for Compliance with Nuclear Regulatory Commission (NRC) Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,” (ADAMS Accession No. ML122400399) and provides an acceptable approach for satisfying the applicable requirements.

20.3.2 Summary of Application

The NRC issued RAI Letter No. 108 dated March 15, 2012, concerning spent fuel pool instrumentation. The applicant responded to the staff’s RAI in letters dated April 25, June 19, August 1, September 27, October 15, and October 31, 2012, and January 18 (ADAMS Accession No. ML130230378), April 5, and May 13, 2013. As part of the RAI response, the applicant submitted a Westinghouse report, APP-SFS-M3R-004, “Response to NRC Orders EA-12-051 and EA-12-063 and Background Information for Future Licensees on AP1000 Spent Fuel Instrumentation.” The RAI responses also proposed adding supplemental information to the final safety analysis report (FSAR) and proposed a license condition.

Supplemental Information

- LNP SUP 9.1-1

The applicant provided supplemental information LNP SUP 9.1-1 addressing spent fuel pool instrumentation in FSAR Section 9.1.3.7.

License Condition

- Part 10, License Condition 12.B

The applicant proposed a license condition related to personnel training for reliable spent fuel pool level instrumentation to Part 10 of the COL application.

20.3.3 Regulatory Basis and Guidance

The requirements and guidance for reliable spent fuel pool instrumentation are established or described in the following:

- SRM-SECY-12-0025, "Staff Requirements – SECY-12-0025 – Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated March 9, 2012, approves issuance of orders for reliable spent fuel pool instrumentation under an administrative exemption to the Backfit Rule and the issue finality requirements in 10 CFR 52.63 and 10 CFR Part 52, Appendix D, Paragraph VIII.

- Atomic Energy Act of 1954, as amended, (the Act), § 161, authorizes the Commission to regulate the utilization of special nuclear material in a manner that is protective of public health and in accord with the common defense and security.

- JLD-ISG-2012-03, Revision 0, "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," issued August 29, 2012, endorses NEI 12-02, Revision 1, "Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," with exceptions and clarifications.

20.3.4 Technical Evaluation

In light of the SECY-12-0025, the staff issued RAI Letter No. 108 requesting additional information in relation to the lessons learned from the Great Tohoku Earthquake and Tsunami. In RAI Letter No. 108, Question 1.5-1, third bullet, the staff requested the applicant to:

- Provide sufficient reliable instrumentation, able to withstand design-basis natural phenomena, to monitor key spent fuel pool parameters (i.e., water level, temperature, and area radiation levels) from the control room (detailed Recommendation 7.1 - Enclosure 6 of SECY-12-0025).

Out of these parameters, the most indicative of SFP conditions is the water level. The radiation monitors are used to confirm the integrity of the stored fuel, but cannot be used to determine how much time remains before the fuel integrity is compromised. The SFP water temperature can be used to monitor SFP water temperature from normal range up to boiling temperature. After the SFP water reaches the boiling point it will remain constant while the pool boils dry, therefore, water temperature cannot be used to determine how much time remains before the fuel integrity is compromised. SFP water level is the most useful parameter to indicate SFP condition. The water stored in the pool provides spent fuel cooling and radiation shielding for the operators on the SFP deck. Therefore, the SFP water level can be used to determine how much time remains before the fuel integrity is compromised.

In Commission Order EA-12-051, the Commission describes the key parameters used to determine that a level instrument is to be considered reliable. NEI 12-02, Appendix A4, "AP1000 Spent Fuel Pool Instrumentation Guidance," provides an AP1000-specific acceptable approach for satisfying the applicable requirements. In order to address the staff's RAI, the applicant submitted a series of letters that discussed how the Levy SFP level instrument is designed to be reliable, following the guidance provided in NEI 12-02, Appendix A4, and the applicant added supplemental information LNP SUP 9.1-1 to Section 9.1.3.7 of the FSAR.

Arrangement:

Commission Order EA-12-051, Attachment 2, Section 1.1 states that the spent fuel pool level instrument channels shall be arranged in a manner that provides reasonable protection of the level indication function against missiles that may result from damage to the structure over the spent fuel pool. This protection may be provided by locating the safety-related instruments to maintain instrument channel separation within the spent fuel pool area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the spent fuel pool structure.

The applicant's response states that the AP1000 design has three safety-related SFP level instrument channels (AP1000 DCD Revision 19, Table 7.5-1 (Sheet 7 of 12)). All three channels and associated instrument tubing lines are located below the fuel handling area operating deck and the cask washdown pit as stated in the supplemental information LNP SUP 9.1-1 added to LNP FSAR Section 9.1.3.7. This location provides level indication function protection from missiles that may result from damage to the structure over the spent fuel pool. In addition, the SFP level instruments associated with protection and safety monitoring system (PMS) Divisions A and C are physically separated from the SFP instrument associated with PMS Division B as stated in the supplemental information added to the LNP FSAR Section 9.1.3.7.

The staff evaluated the instrument description provided in the DCD and the proposed supplemental information added to LNP FSAR Section 9.1.3.7 and determined that the SFP level instrument will be arranged in a manner that provides reasonable protection against missiles, and therefore, the staff concludes that these features are in conformance with Commission Order EA-12-051, and the guidance provided by JLD-ISG-2012-03.

Qualification:

Commission Order EA-12-051, Attachment 2, Section 1.2 states that the level instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period.

The applicant's response states that the three safety-related SFP level instruments are seismically qualified and are located below the fuel handling area operating deck (AP1000 DCD Revision 19, Section 9.1.3.4.3.4 and Table 7.5-1 (Sheet 7 of 12)).[2] The environment in these areas is mild with respect to safety-related equipment qualification and affords access for post-accident actions. Even though they are not directly exposed to SFP boiling, the instruments are qualified to function at the conditions (temperature, humidity, and radiation) that could be seen where these instruments are located. This provides assurance that the SFP level transmitters exposed to these environmental conditions will remain available and functional for an extended period.

The staff reviewed the applicant's response and concludes that since the SFP level transmitters are not located on the pool area, they are not required to be designed to handle the pool area conditions. However, they must be designed to remain operational under the worst expected conditions for the area in which they are located. The AP1000 DCD does state that the instruments are designed to remain functional at the expected local conditions; therefore, the staff concludes that these features are in conformance with Commission Order EA-12-051, and the guidance provided by JLD-ISG-2012-03.

Power Sources:

Commission Order EA-12-051, Attachment 2, Section 1.3 states that the instrumentation channels shall provide for power connections from sources independent of the plant ac and direct current (dc) power distribution systems, such as portable generators or replaceable batteries. Power supply designs should provide for quick and accessible connection of sources independent of the plant ac and dc power distribution systems. Onsite generators used as an alternate power source and replaceable batteries used for instrument channel power shall have sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.

The applicant's response states that the AP1000 SFP level instruments are provided with Class 1E DC power supply for at least 72 hours of post-accident monitoring. One of these safety-related instruments is powered through PMS Division A which contains a 24-hour battery supply. The safety-related SFP level instrument PMS divisions are described in the supplemental information (LNP SUP 9.1-1) added to the LNP FSAR Section 9.1.3.7. A description of the AP1000 Class 1E DC and UPS system is contained in AP1000 DCD Revision 19, Section 8.3.2.1.1. Beyond the initial 72 hours, instrument power can be supplied by the use of onsite permanently installed ancillary diesel generators or offsite portable generators with quick and accessible connection points. Permanently installed onsite ancillary diesel generators are capable of providing power for Class 1E post-accident monitoring

[2] The RAI responses for this topic discuss a departure from the AP1000 DCD related to environmental zones for the level instruments. The departure is evaluated in FSER section 3.11.4

including SFP level instrumentation. This capability is described in Westinghouse AP1000 DCD Revision 19, Section 8.3.1.1.1. As described in Westinghouse AP1000 DCD Revision 19, Section 1.9.5.4, offsite portable generators are capable of being connected to distribution panels or to a safety-related connection.

As discussed in the applicant's response and as described in the AP1000 DCD, the safety related power distribution system has the capability of using portable generators to power safety related distribution panels, which power the level instruments. These panels are Seismic Category I and designed to remain operational following a safe shutdown earthquake. Based on the system description, the staff concludes that these design features are in conformance with Commission Order EA-12-051, and the guidance provided by JLD-ISG-2012-03.

Accuracy:

Commission Order EA-12-051, Attachment 2, Section 1.4 states that the instrument shall maintain its designed accuracy following a power interruption or change in power source without recalibration.

The applicant's response states that the measured range of the SFP level by the safety-related instruments is from the top of the SFP to the top of the fuel racks, the level instruments are calibrated at a reference temperature suitable for normal SFP operation and will read conservatively at elevated temperatures, including during boiling conditions. These instruments are calibrated on a regular basis and their accuracy is not affected by power interruptions. All these design features are described in the supplemental information (LNP SUP 9.1-1) added to LNP FSAR Section 9.1.3.7.

Based on the system description provided above, the staff concludes that these design features are in conformance with Commission Order EA-12-051, and the guidance provided by JLD-ISG-2012-03.

Display:

Commission Order EA-12-051, Attachment 2, Section 1.5 states that the display shall provide on-demand or continuous indication of spent fuel pool water level.

The applicant's response states that the safety-related SFP level sensors provide continuous indication of the SFP level to the main control room (MCR) as well as the Remote Shutdown Workstation (RSW) and are included in the Qualified Data Processing System (QDPS) PMS display as indicated in Westinghouse AP1000 DCD Revision 19, Table 7.5-1 (Sheet 7 of 12). Safety-related instrumentation gives an alarm in the MCR when the water level in the SFP reaches the low-low-level setpoint as stated in AP1000 DCD Revision 19, Section 9.1.3.7.D.

Based on the system description provided above, the staff concludes that these design features are in conformance with Commission Order EA-12-051, and the guidance provided by JLD-ISG-2012-03.

License Condition

Commission Order EA-12-051, Attachment 2, Section 2 states that the spent fuel pool instrumentation shall be maintained available and reliable through appropriate development and implementation of a training program. Personnel shall be trained in the use and the provision of alternate power to the safety-related level instrument channels.

The applicant's COLA Part 10 includes License Condition 12.B, which requires the development and implementation of a training program in accordance with the guidance contained in JLD-ISG-2012-03.

The applicant's proposed license condition states:

> B. RELIABLE SPENT FUEL POOL LEVEL INSTRUMENTATION
>
> Prior to initial fuel load, DEF shall fully implement the following requirements for spent fuel pool level indication using the guidance contained in JLD-ISG-2012-03, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0.
>
> - The spent fuel pool instrumentation shall be maintained available and reliable through the development and implementation of a training program. The training program shall include provisions to insure trained personnel can route the temporary power lines from the alternate power source to the appropriate connection points and connect the alternate power source to the safety-related level instrument channels.

The proposed license condition is consistent with the guidance provided in JLD-ISG-2012-03, and is intended to ensure that the operators will be properly trained in the adequate equipment maintenance procedures and the proper operational procedures in order to establish the necessary alternate power connections. Based on this, the staff concludes that the proposed license condition is acceptable because the development and implementation of a training program is consistent with Commission Order EA-12-051 and the guidance provided by JLD-ISG-2012-03.

20.3.5 Post Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following license condition related to development and implementation of a training program:

- License Condition (20-2) – Prior to initial fuel load, the Licensee shall address the following requirements using the guidance contained in JLD-ISG-2012-03, Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation, Revision 0:

  The spent fuel pool instrumentation shall be maintained available and reliable through the development and implementation of a training program. The training program shall include provisions to ensure trained personnel can route the temporary power lines from

the alternate power source to the appropriate connection points, and connect the alternate power source to the safety-related level instrument channels.

20.3.6 Conclusion

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to SFP instrument reliability, and there is no outstanding information expected to be addressed in the LNP COL FSAR.

The staff evaluated the applicant's and the AP1000 design description of the SFP water level instrument and determined that the instruments are in accordance with the guidance provided in JLD-ISG-2012-03. Therefore, the staff concludes that the applicant's SFP level instruments are considered reliable, able to withstand design-basis natural phenomena and monitor key spent fuel pool level parameters as described in Commission Order EA-12-051. In addition, the staff concludes that the information presented in the LNP COL FSAR is acceptable because it conforms to the guidance provided in JLD-ISG-2012-03. The staff based its conclusions on the following:

- LNP SUP 9.1-1 is acceptable because, when combined with the information in Table 7.5-1 and Sections 8.3.1.1.1 and 9.1.3.7.D of the AP1000 DCD, it includes provisions for SFP instrumentation arrangement, qualification, power sources, accuracy and display that are consistent with the requirements described in SECY-12-0025 and Commission Order EA-12-051.

- The proposed license condition is acceptable because it provides that, prior to fuel load, the licensee will have in place procedures for the proper maintenance of the level instruments and for the connection and use of an alternate power source in order to power the level instruments.

20.4 Emergency Preparedness (Based on Recommendation 9.3)

20.4.1 Introduction

The accident at Fukushima reinforced the need for effective emergency preparedness, the objective of which is to ensure the capability exists for a licensee (or COL applicant) to implement measures that mitigate the consequences of a radiological emergency and provide for protective actions of the public. The accident at Fukushima highlighted the need to determine and implement the required staff to fill all necessary positions of the emergency organization responding to a multi-unit event with impeded access to the site. Additionally, there is a need to ensure that the communication equipment relied on has adequate power to coordinate the response to an event during an extended loss of ac power.

20.4.2 Summary of Application

In Revision 9 of the LNP Units 1 and 2 COL application, Part 10, the applicant proposed a license condition related to emergency preparedness communications and staffing. The staff's discussion is located in the Technical Evaluation section below.

20.4.3 Regulatory Basis

The requirements and guidance for emergency preparedness for beyond-design-basis external events are established or described in the following:

- 10 CFR 50.47(b)(6) states that provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

- 10 CFR 50.47(b)(1) states, in part: ". . . each principal response organization has staff to respond and to augment its initial response on a continuous basis."

- 10 CFR 50.47(b)(2) states, in part: ". . . adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, . . ."

- 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section IV. E. 9. states that adequate provisions shall be made and described for emergency facilities and equipment, including "at least one onsite and one offsite communications system; each system shall have a backup power source."

- SECY-12-0025 states, in part, that the staff will also request all COL applicants to provide the information required by the orders and request for information letters described in this paper, as applicable, through the review process.

- NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," Revision 0 - By NRC letter from David Skeen, Director, Japan Lessons-Learned Directorate, to NEI, Susan Perkins-Grew, Director, Emergency Preparedness, dated May 15, 2012, NRC finds the guidance in NEI 12-01 to be an acceptable method for licensees to employ when responding to the 10 CFR 50.54(f) letters regarding NTTF Recommendation 9.3.

- NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Section B, Onsite Emergency Organization, states in part:

> 5. Each licensee shall specify . . . functional areas of emergency activity...These assignments shall cover the emergency functions in Table B-1 entitled, 'Minimum Staffing Requirements for Nuclear Power Plant Emergencies.' The minimum on-shift staffing shall be as indicated in Table B-1. The licensee must be able to augment

on-shift capabilities within a short period after declaration of an emergency. This capability shall be as indicated in Table B-1. . .

- NUREG-0696, "Functional Criteria for Emergency Response Facilities," issued February 1981, offers guidance on how to meet the requirements of Appendix E to 10 CFR Part 50 and describes the onsite and offsite communications requirements for the licensee's emergency response facilities.

20.4.4 Technical Evaluation

The NRC issued RAI Letter No. 108 dated March 15, 2012 to the applicant, concerning implementation of the Fukushima NTTF Recommendation 9.3 in the combined license application for LNP Units 1 and 2. In response, the applicant proposed a license condition in Revision 5 of the LNP COL application submitted on July 31, 2012, to address the 10 CFR 50.54(f) request for information letters sent to existing licensees – including COL applicants - regarding communications and staffing for NTTF Recommendation 9.3. This license condition was subsequently revised in Revision 7 of the license application. As part of its proposed license condition, the applicant committed to perform assessments for NTTF Recommendation 9.3 using NEI 12-01, Revision 0. By letter from the NRC to NEI dated May 15, 2012 (ADAMS Accession No. ML1213A043), the NRC stated that the guidance in NEI 12-01, Revision 0, provides an acceptable method for licensees to employ when responding to the 10 CFR 50.54(f) letters regarding NTTF Recommendation 9.3. The applicant proposed the license condition on communications and staffing in License Condition 12, Section C to Part 10 of the COL application. The staff reviewed the applicant's proposed license condition and revised it to reflect the NRC's expectation when addressing NTTF Recommendation 9.3 as stated below in Section 20.4.5 of this SER. The NRC staff has revised the timeframe of the completion of this license condition to be consistent with the schedules provided in 10 CFR 52.99(a) and 10 CFR 52.103(a).

20.4.5 Post Combined License Activities

The license condition language in this section has been clarified from previously considered language. In a letter dated March 22, 2016 (ADAMS Accession No. ML16084A099), the applicant did not identify any concerns with the clarified license condition language. The changes do not affect the staff's above analysis of the conditions, and therefore, for the reasons discussed in the technical evaluation section above, the staff finds the following license conditions acceptable:

- License Condition (20-3) – No later than eighteen (18) months before the latest date set forth in the schedule submitted in accordance with 10 CFR § 52.99(a) for completing the inspections, tests, and analyses in the ITAAC, the licensee shall have performed an assessment of the on-site and augmented staffing capability for response to a multi-unit event. The staffing assessment shall be performed in accordance with NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," Revision 0.

No later than one hundred eighty (180) days before the date scheduled for initial fuel load, as set forth in the notification submitted in accordance with 10 CFR § 52.103(a), the licensee shall revise the Emergency Plan to include the following:

(a) Incorporation of corrective actions identified in the staffing assessment required by this license condition; and
(b) Identification of how the augmented staff will be notified, given degraded communications capabilities.

- License Condition (20-4) – No later than eighteen (18) months before the latest date set forth in the schedule submitted in accordance with 10 CFR § 52.99(a) for completing the inspections, tests, and analyses in the ITAAC, the licensee shall have performed an assessment of on-site and off-site communications systems and equipment relied upon during an emergency event to ensure communications capabilities can be maintained during an extended loss of alternating current power. The communications capability assessment shall be performed in accordance with NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," Revision 0.

  No later than one hundred eighty (180) days before the date scheduled for initial fuel load set forth in the notification submitted in accordance with 10 CFR § 52.103(a), the licensee shall have completed implementation of corrective actions identified in the communications capability assessment, including revisions to the Emergency Plan.

20.4.6 Conclusion

Based on the staff's review, the staff finds that the license condition, as revised by the staff above, is acceptable because it conforms to the guidance provided in SECY-12-0025 and NEI 12-01 regarding communications and staffing to address NTTF Recommendation 9.3, in NUREG-0654/FEMA-REP-1, and in NUREG-0696, and meets the applicable requirements in 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50.