22.0 REQUIREMENTS RESULTING FROM FUKUSHIMA NEAR-TERM TASK FORCE RECOMMENDATIONS

This Final Safety Analysis Report (FSAR) chapter addresses the requirements resulting from the Fukushima Near-Term Task Force (NTTF) recommendations that are applicable to the South Texas Project (STP), Units 3 and 4, Combined License (COL) application. 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 United States (U.S.) Nuclear Regulatory Commission (NRC or Commission) 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 its policy direction. In July 2011, the NTTF issued a 90-day report as SECY-11-0093, "Near Term Report and Recommendations for Agency Actions Following the Events in Japan," in which it made 12 recommendations (Agencywide Documents Access and Management System (ADAMS) Accession No. ML11186A950). On September 9, 2011, in SECY-11-0124, "Recommended Actions to Be Taken Without Delay From NTTF Report" (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" (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 as follows:

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

2. Implementation of SFP instrumentation proposed in Recommendation 7.1.

The staff also prioritized the NTTF recommendations into Tier 1, Tier 2, and Tier 3. Tier 1 recommendations represent those that the staff determined should be started without unnecessary delay. Tier 2 recommendations represent those that could not be initiated in the near term. Tier 3 recommendations 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" (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, and as described below, the Commission approved issuance of the proposed orders with some modifications in the staff requirements memorandum (SRM) to SECY-12-0025. On March 12, 2012, the NRC issued two orders, namely 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 (ML12054A735 and ML12054A679, respectively). The Commission determined that the requirements of EA-12-049, were necessary to provide adequate protection to the public health and safety, and EA-12-051, was issued to provide enhanced protection under an administrative exemption to the Backfit Rule in Title 10 of the *Code of Federal* Regulations (10 CFR) 50.109, "Backfitting," and the issue finality requirements in 10 CFR 52.63, "Finality of standard design certifications" and 10 CFR Part 52, Appendix D, Paragraph VIII, "Design Certification Rule for the AP1000 Design."

In a letter dated March 12, 2012 (ML12053A340), the staff also issued the request for information to the appropriate licensees and permit holders pursuant to 10 CFR 50.54(f) regarding Recommendations 2.1, 2.3 and 9.3, as described in SECY-12-0025.

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 STP, Units 3 and 4, 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).

The staff determined that the following four recommendations were applicable and should be addressed by the STP, Units 3 and 4, COL applicant:

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: Mitigation Strategies for Beyond-Design-Basis External Events – Order licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment and SFP cooling capabilities following a beyond-design-basis external event, including, among other things, for a multi-unit event.

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:

    - 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 STP 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 is no additional information that is needed to address Recommendation 2.1 for the STP, Units 3 and 4, COL application. Additionally, the staff determined that Recommendation 2.3 is not applicable to the STP COL because the plant is not yet constructed, and Recommendation 5.1 is not applicable because it applies to boiling-water reactor (BWR) type plant designs with Mark I and Mark II Containments, which differ significantly from the Advanced Boiling Water Reactor (ABWR) containment. 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.

SECY-12-0025 states that the staff will request all COL applicants to provide the information required by the orders and request for information letters through the review process. By letter dated April 25, 2012 (ML121080046), the staff informed the applicant that the staff had been directed by the Commission to implement the Fukushima NTTF recommendations in SECY-12-0025. Accordingly, for the STP COL application, the staff issued request for additional information (RAI) 417 (ML121230021), dated May 2, 2012, related to Implementation of Fukushima NTTF Recommendations pertaining to seismic hazard reevaluation, mitigation strategies for beyond design- basis external events, SFP 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 recommendations.

22.1 Recommendation 2.1 Seismic Hazard Reevaluation

22.1.1 Introduction

In SECY-12-0025 Enclosure 7, Attachment 1 to Seismic Enclosure 1 (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 staff issued NUREG–2115 in January 2012, as a replacement for 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 model for the CEUS (CEUS-SSC). Consistent with Regulatory Guide (RG) 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," which states that the latest available information should be used in the PSHA, the staff requested the applicant to evaluate the seismic hazards at the STP site using the CEUS-SSC in a PSHA.

This section of the safety evaluation report (SER) provides the staff's evaluation of the seismic hazards at the STP site performed in consideration of the CEUS-SSC model. The information discussed in this SER section supports the staff's evaluation in SER Sections 2.5.2, "Vibratory Ground Motion"; 2.5.4, "Stability of Subsurface Materials and Foundations"; and 3.7, "Seismic Design."

Summary of the CEUS-SSC Model

In this section, the staff summarizes the CEUS-SSC model that the applicant used for the seismic hazard reevaluation in response to the RAI 01.05-1, issued to the applicant on May 2, 2012 (ML121230021). This summary focuses on the parts of the CEUS-SSC model that are applicable to the STP site's seismic hazard and provides background and a framework for the staff's technical evaluation of the applicant's seismic hazard reevaluation in SER Section 22.1.4. In SER Section 22.1.4, the staff describes and evaluates the specific deviations taken by the applicant during model implementation from the as-is model published in NUREG–2115.

On January 31, 2012, the NRC, U.S. Department of Energy (DOE), and EPRI issued a new seismic source characterization model and report for use in seismic hazard assessments of 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 seismic source characterization 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) a comprehensive, participatory peer review. The cooperative project for this new model was conducted using processes described in the SSHAC guidance in 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 U.S. east of the Rocky Mountains. The CEUS-SSC study region is shown in Figure 22.1-1, "Map Showing the Locations of the CEUS Study Region, STP Units 3 and 4, and the Houston Test Site," of this SER. The CEUS-SSC Project resulted in products and methodological improvements that will be valuable to future users for: (1) data evaluation and data summary tables that identify all the data considered by the project team and indicate the team's views of the quality of the data and the degree of reliance placed on any given data set; (2) geologic, geophysical, and seismological databases; (3) earthquake catalogs with uniform moment — magnitudes (M); (4) updated paleoseismicity data and guidance; and (5) recommendations regarding future applications of the seismic source characterization model. For purposes of demonstrating the CEUS-SSC model, the project also incorporated sample calculations at the seven sites identified in Figure 8.1-1, "Map Showing the Study Area and Seven Test Sites for the CEUS SSC Project," of NUREG–2115—including the Houston Test Site shown in Figure 22.1-1.

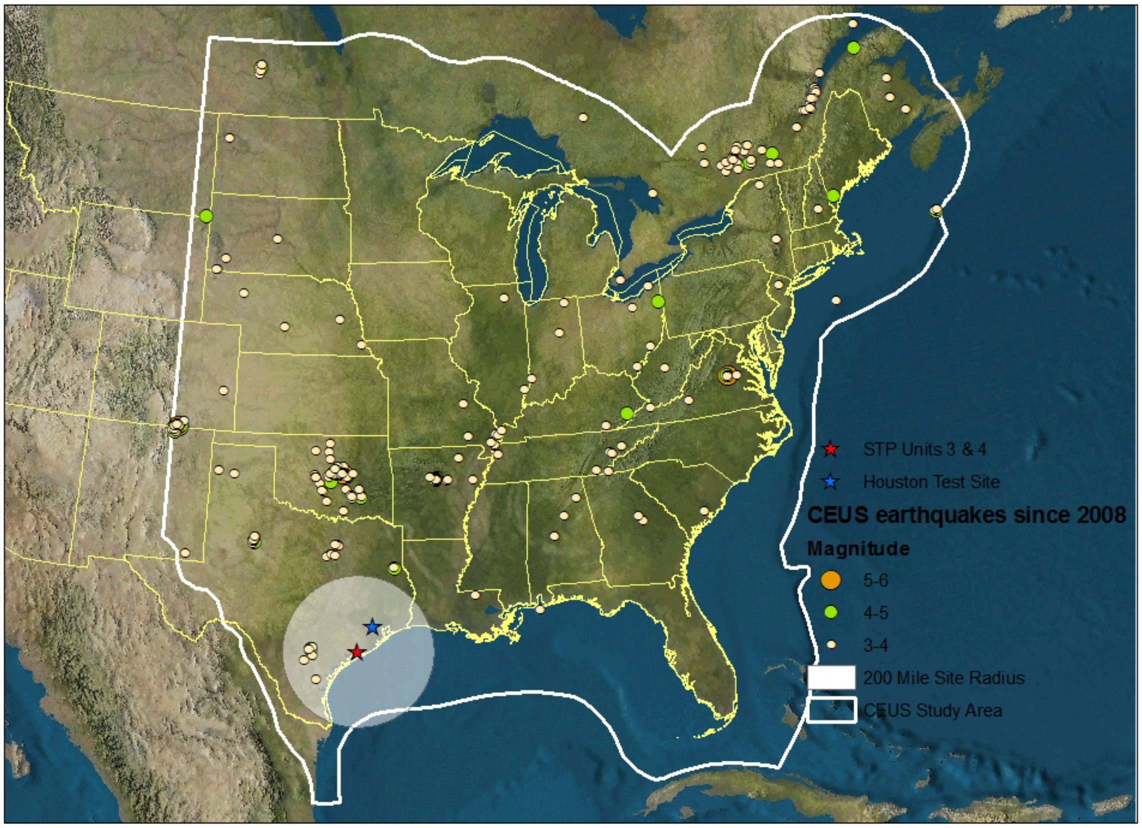

Figure 22.1-1 Map Showing the Locations of the CEUS Study Region, STP Units 3 and 4, and the Houston Test Site

The applicant used the EPRI-SOG model in the STP, Units 3 and 4, Final Safety Analysis Report (FSAR) evaluation of vibratory ground motion in Section 2.5S.2, "Vibratory Ground Motion." 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 STP 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. Although the applicant has updated the EPRI-SOG model on a site-specific basis, there has not been a systematic update of the full model in more than 20 years. The project to develop the CEUS-SSC model created an up-to-date CEUS seismic hazard model that takes 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 developed as part of the licensing actions for proposed and existing nuclear power facilities. Lastly, the CEUS-SSC model contains updated methods for evaluating the data and for quantifying uncertainties in the PSHA model. Because the STP applicant submitted the COL application to the NRC for review in September 2007, before the CEUS-SSC model was published in NUREG–2115 in January 2012, the applicant used the EPRI-SOG model in the initial application. The applicant later updated the application to include a sensitivity evaluation of the seismic hazard for the STP site using the newer CEUS-SSC model.

The CEUS-SSC model consists of three models of seismic sources—the M_{max} 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 to assess the spatial and temporal distribution of future seismicity. These are the M_{max} zones models and the seismotectonic zones model, which represent the background or distributed seismicity in the CEUS using two different approaches to characterize future earthquakes.

The M_{max} zones model is based on average or "default" characteristics that are representative of large areas of the CEUS, or the entire study area; M_{max} zones thus 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 in the M_{max} zones model. The seismotectonic zones model uses historical seismicity and regional-scale geologic and tectonic data to characterize seismic source zones.

Finally, the RLME sources model is the third type of seismic source. The RLME sources model is not based on distributed seismicity within an areal source as in the M_{max} and seismotectonic zones models. The RLME sources model is defined primarily by paleoseismic evidence and, as its name suggests, it represents the sources where repeated large magnitude earthquakes occur.

Figure 22.1-2, "CEUS-SSC Master Logic Tree," of this SER, 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 historical and paleoseismic records and are defined as having experienced two or more earthquakes with a moment magnitude of at least M 6.5. Figure 22.1-3, "The Geographic Locations of the RLME Sources," of this SER shows the geographic locations of the RLME sources.

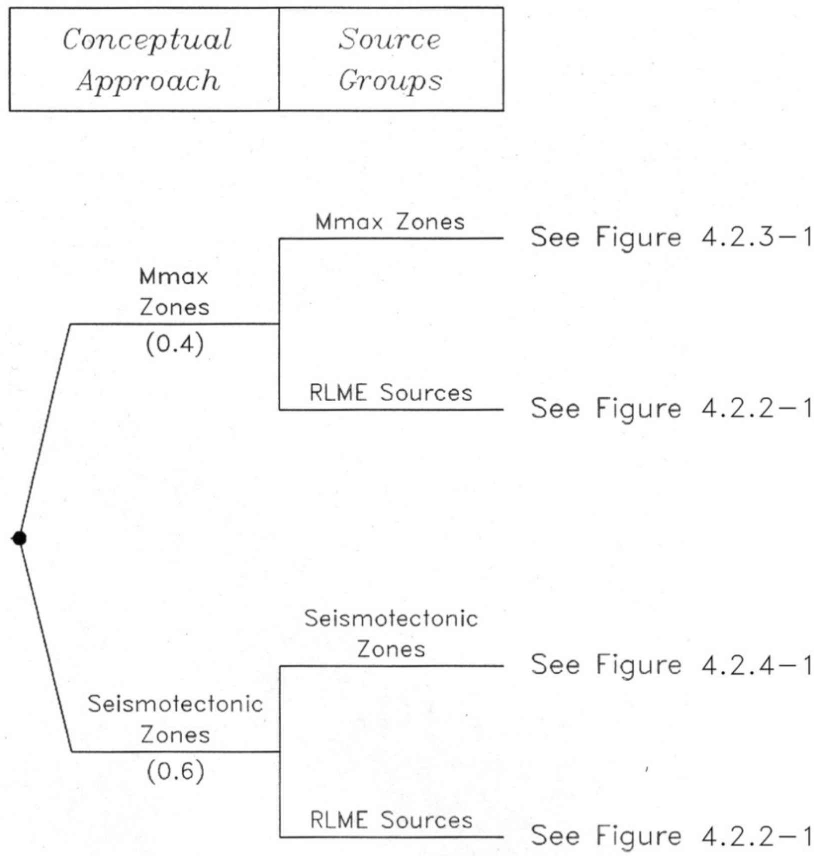

Figure 22.1-2 CEUS-SSC Master Logic Tree

(Taken from Figure 4.2.1-1 of NUREG–2115)

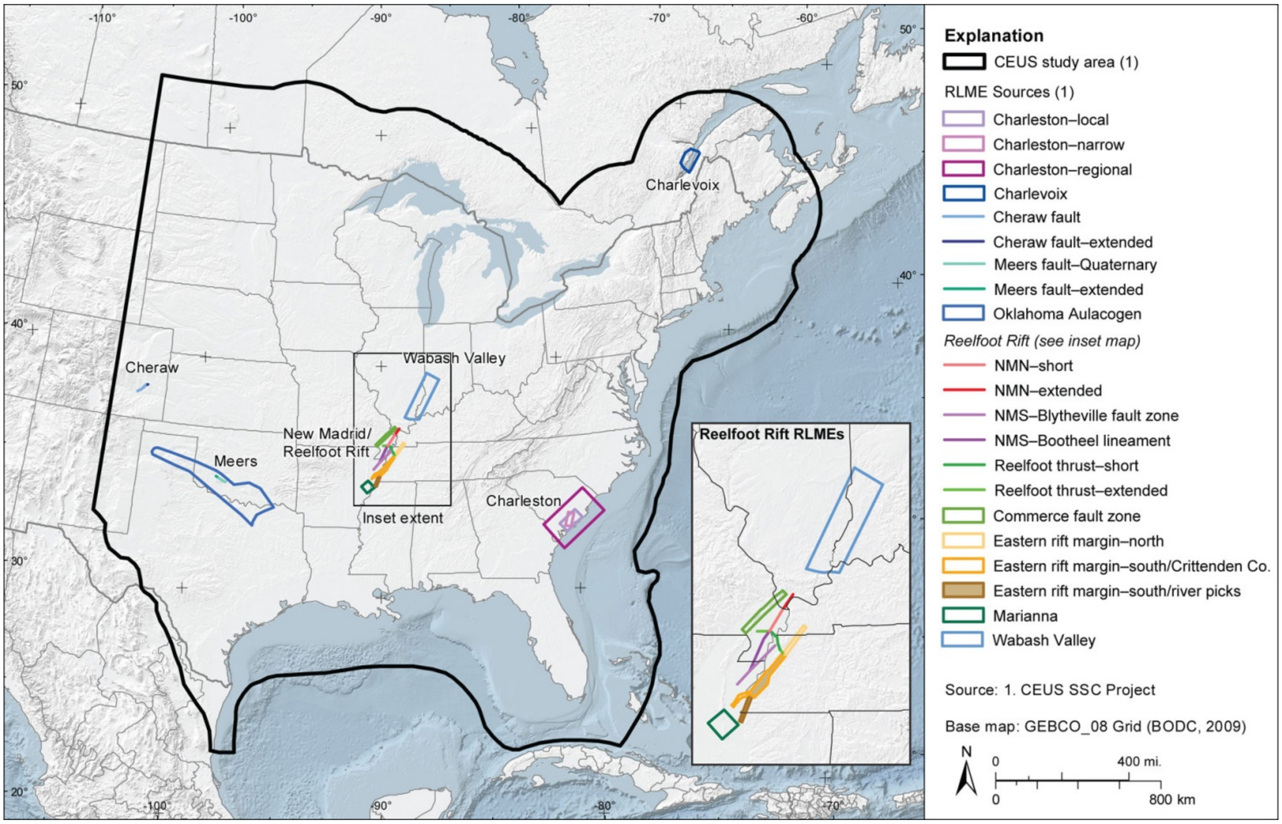

Figure 22.1-3 The Geographic Locations of the RLME Sources
(Taken from Figure 4.2.3-2 of NUREG–2115)

Each seismic source in the M_{max} zones, seismotectonic zones, and RLME sources models is defined by a source geometry; a set of maximum magnitude (M_{max}) 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 M_{max} zones, 17 sources characterized as seismotectonic zones, and 10 RLME sources. Each of the seismic source zones can have multiple alternative characterizations (geometries, M_{max} 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; 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 cells that were one-degree longitude by one-degree latitude. The smaller cell size used in the CEUS-SSC model achieves a higher resolution, which is particularly important for more active regions.

For the M_{max} zones model, the CEUS-SSC logic tree for the M_{max} zones is shown in Figure 4.2.3-1, “Logic Tree for the Mmax Zones Branch of the Master Logic Tree,” of NUREG–2115, and four source geometries are shown in Figure 4.2.3-2, “Subdivision Used in the Mmax Zones Branch of the Master Logic Tree (Narrow MESE Zone),” and Figure 4.2.3-3, “Subdivision Used in the Mmax Zones Branch of the Master Logic Tree (Wide MESE Zone),” of NUREG–2115; the fifth M_{max} zone covers the entire CEUS study region (See Figures 22.1-1 and 22.1-3, in this SER). The STP site is located in the "Mesozoic-and-younger extension" (MESE) M_{max} zone, where MESE-N and MESE-W distinguish between narrow (N) and wide (W) geometry interpretations. Figure 22.1-4, "Mesozoic-and-Younger Extension Zone (MESE) and Non-Mesozoic-and-Younger Zone (NMESE), where MESE-N and NMESE-N Refer to the Narrow (N) Geometry Interpretation," of this SER, depicts the narrow MESE and Non-Mesozoic Zones.

For the seismotectonic zones model, the CEUS-SSC logic tree for the seismotectonic zones is shown in Figure 4.2.4-1(a), "Logic Tree for the Seismotectonic Zones Branch of the Master Logic Tree," and Figure 4.2.4-1(b), "Logic Tree for the Seismotectonic Zones Branch of the Master Logic Tree," of NUREG–2115; the source geometries are shown in Figures 4.2.4-2 through 4.4.4-5 of NUREG–2115. The STP site is located in the "extended continental crust-Gulf Coast" (GHEX) seismotectonic source zone. Figure 22.1-5, "The Seismotectonic Zones in the Case where the Rough Creek Graben is Not Part of the Reelfoot Rift (RR), and the Paleozoic Extended Zone is Narrow (PEZ-N)," of this SER, shows the seismotectonic zones in the case where the Rough Creek Graben is not part of the Reelfoot Rift (RR), and the Paleozoic Extended Zone is narrow (PEZ-N).

For the RLME model, the CEUS-SSC logic trees for the Meers and New Madrid fault system (NMFS) RLME sources are shown in Figure 6.1.4-2, "Logic Tree for the Meers Fault Source," and Figure 6.1.5-1, "Logic Tree for the NMFS RLME Source," of NUREG–2115. The NMFS and the Meers RLME sources are the closest RLME sources to the STP site. Each of the 10 RLME sources (See Figure 22.1-3) has a logic tree defining the uncertainty in its characterization. The characterization of the NMFS RLME source in the CEUS-SSC model is similar to the updated New Madrid Seismic Zone (NMSZ) seismic source (Exelon, 2006) used by the applicant in FSAR Subsection 2.5S.2.4, "Probabilistic Seismic Hazard Analysis and Controlling Earthquake," and discussed and evaluated in Subsection 2.5.2.2 and Subsection 2.5.2.4, respectively, of this SER.

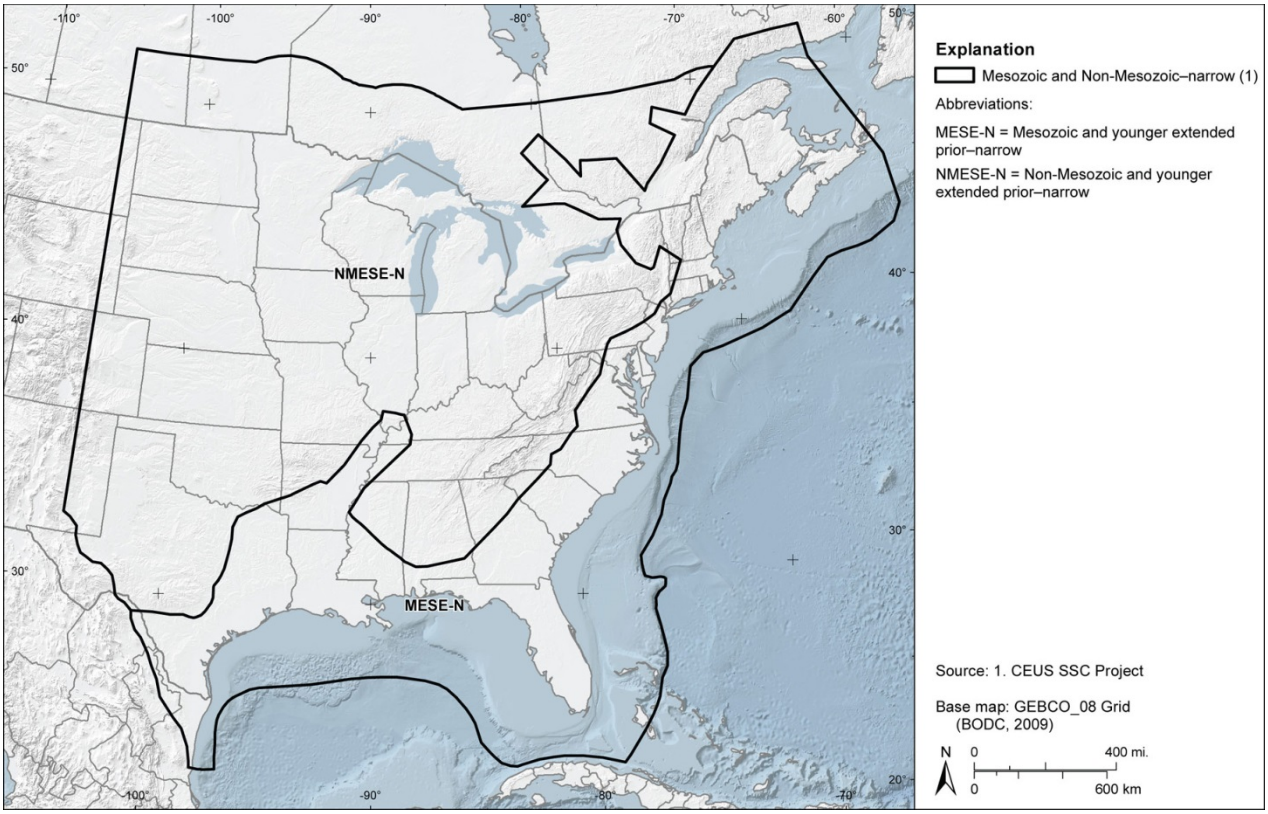

Figure 22.1-4 Mesozoic-and-Younger Extension Zone (MESE) and Non-Mesozoic-and-Younger Zone (NMESE), where MESE-N and NMESE-N Refer to the Narrow (N) Geometry Interpretation (Taken from Figure 4.2.3-2 of NUREG–2115)

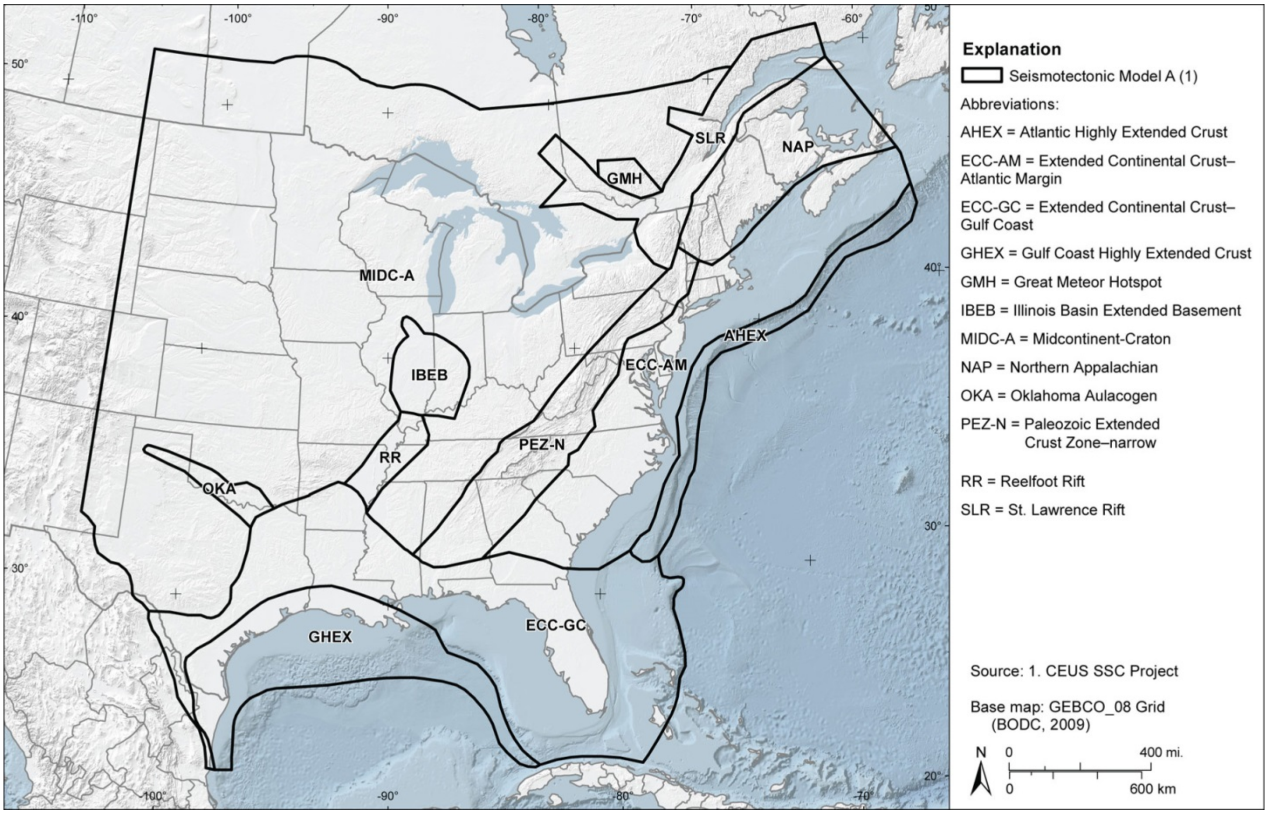

Figure 22.1-5 The Seismotectonic Zones in the Case where the Rough Creek Graben Is Not Part of the Reelfoot Rift (RR), and the Paleozoic Extended Zone Is Narrow (PEZ-N) (Taken from Figure 4.2.4-2 of NUREG–2115)

22.1.2 Summary of the Application

The applicant provided information to evaluate the seismic hazard at the STP site against current NRC requirements and guidance. The staff issued RAI 01.05-1, dated May 2, 2012 (ML121230021), which requested the applicant to evaluate the seismic hazard at the STP site using the CEUS-SSC model. The RAI also requested the applicant—if necessary—to update the design basis and the structures, systems, and components (SSCs) important to safety to protect against the updated hazards. In its response to RAI 01.05-1, dated December 6, 2012 (ML12346A445), the applicant proposed to incorporate changes into FSAR Appendix 1E, "Response to NRC Post-Fukushima Recommendations." Specifically, the applicant provided the seismic reevaluation in Subsection 1E.2.1.1 of Appendix 1E.

These changes are incorporated in Revision 12 of the STP COL FSAR.

22.1.3 Regulatory Basis

The applicable regulatory requirements for the seismic hazard reevaluation are established and described as follows:

- 10 CFR 100.23, "Geologic and Seismic Siting Criteria," with respect to obtaining geologic and seismic information necessary to determine site suitability and to establish site geologic and site characteristics as bases for design.

- 10 CFR 52.79(a)(1)(iii), as it relates to consideration of the most severe natural phenomena historically reported for the site and the surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data were 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 SSCs important to safety be designed to withstand the effects of natural phenomena—such as earthquakes—so the SSCs do not lose their capabilities to perform their safety functions.

- 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

In addition, the geologic and seismic characteristics of the STP site should be determined in accordance with the 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, Revision 1, "Design Response Spectra for Seismic Design of Nuclear Power Plants."

- RG 1.132, Revision 2, "Site Investigations for Foundations of Nuclear Power Plants."

- 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."

- Design Certification (DC)/COL Interim Staff Guidance (ISG)-017, "Interim Staff Guidance on Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses."

22.1.4 Technical Evaluation

This SER section provides the staff's evaluation of the applicant's response to RAI 01.05-1 dated December 6, 2012 (ML12346A445), as it relates to the applicant's evaluation of the seismic hazard at the STP site using the CEUS-SSC model, consistent with the need to consider the latest information in the PSHA as indicated in RG 1.208. The applicant evaluated potential seismic hazards at the STP site using the CEUS-SSC model (in NUREG–2115) and then performed a sensitivity study comparing the results with those that the applicant had previously generated using the EPRI-SOG model.

22.1.4.1 *Implementation of the CEUS-SSC Model and Sensitivity Evaluation*

The applicant evaluated the potential impact of the CEUS-SCC model on the characterization of seismic hazards at the STP site. For this evaluation, the applicant relied on the 1-, 10-, and 100-Hertz (Hz) hazard curves for the nearby Houston Test Site in Chapter 8 of NUREG–2115 because both sites share similar geologic and tectonic settings. The applicant noted that both sites have similar activity rates for the average of the STP-updated EPRI-SOG Earth Science Teams (ESTs) and for the CEUS-SSC. In its response to RAI 01.05-1, dated December 6, 2012 (ML12346A445), the applicant provided the results for its evaluation of the potential impact of the CEUS-SSC model on the characterization of seismic hazards at the STP site in Figure 2a, "Activity Rates for the Average of the STP-Updated EPRI-SOG ESTs (Earth Science Teams)," Figure 2b, "Activity Rates for the CEUS SSC," Figure 3a, "Relative Values of Activity Rates, $N_5/N_{5,Houston,}$ for the STP-Updated EPRI-SOG ESTs," and Figure 3b "Relative Values of Activity Rates, $N_5/N_{5,Houston}$, for the CEUS SSC Study." The applicant obtained the values for the CEUS-SSC rock spectral accelerations at the 10^{-4} and 10^{-5} mean annual exceedance from the curves in Figure 8.2-3d, "Houston 10 Hz Rock Hazard: Total and Contribution by RLME and Background," Figure 8.2-3e, "Houston 1Hz Rock Hazard: Total and Contribution by RLME and Background," and Figure 8.2-3f, "Houston PGA Rock Hazard: Total and Contribution by RLME and Background," of NUREG–2115 for 1, 10, and 100 Hz. The applicant estimated 10^{-4} and 10^{-5} spectral accelerations at 30 Hz by using the ratio of 100-Hz to 30-Hz rock motions from the STP, Units 3 and 4, FSAR and then applying this ratio to the CEUS-SSC peak ground acceleration (PGA) value. The applicant asserted that the ratio of 100-Hz to 30-Hz spectral acceleration developed for the STP, Units 3 and 4, site would closely approximate the Houston Test Site because: (1) this ratio is stable for a wide range of critical magnitudes and distances (McGuire et al., 2001); (2) both the CEUS-SSC and STP, Units 3 and 4, PSHA models use the EPRI (2004, 2006) ground motion models (GMMs); (3) the resulting 1-, 10-, and 100-Hz hazard curves in both models are very similar; and (4) the seismic hazard in the Houston-STP region varies slowly with location.

Using the procedure recommended in RG 1.208, the applicant then developed a hard rock ground motion response spectrum (GMRS) at frequencies of 1, 10, 30, and 100 Hz. Next, the applicant scaled the hard rock GMRS by the STP site-specific amplification factors [i.e., from FSAR Table 2.5S.2-21, "Horizontal 10^{-4} and 10^{-5} Site Specific UHRS (in g) and Calculation of GRMS (in g)]," at the corresponding frequencies. The applicant's GMRS values are in Figure 22.1-6, "Comparison of the Horizontal STP Units 3 and 4 Site-Specific SSE-the STP Units 3 and 4 COL Application GMRS-and the Houston Test Site CEUS SSC GMRS," of this SER, in addition to the STP COL application GMRS and the STP site-specific safe-shutdown earthquake (SSE) ground motion. Based on this comparison, the applicant concluded that the estimated CEUS-SSC STP GMRS is very close to—and not significantly above—the STP COL application GMRS, while the site-specific SSE envelopes the GMRS from both. The applicant further concluded that because the STP, Units 3 and 4, COL application GMRS and the estimated CEUS-SSC results for the STP site are not significantly different, the STP, Units 3 and 4, COL seismic design basis does not need to be revised.

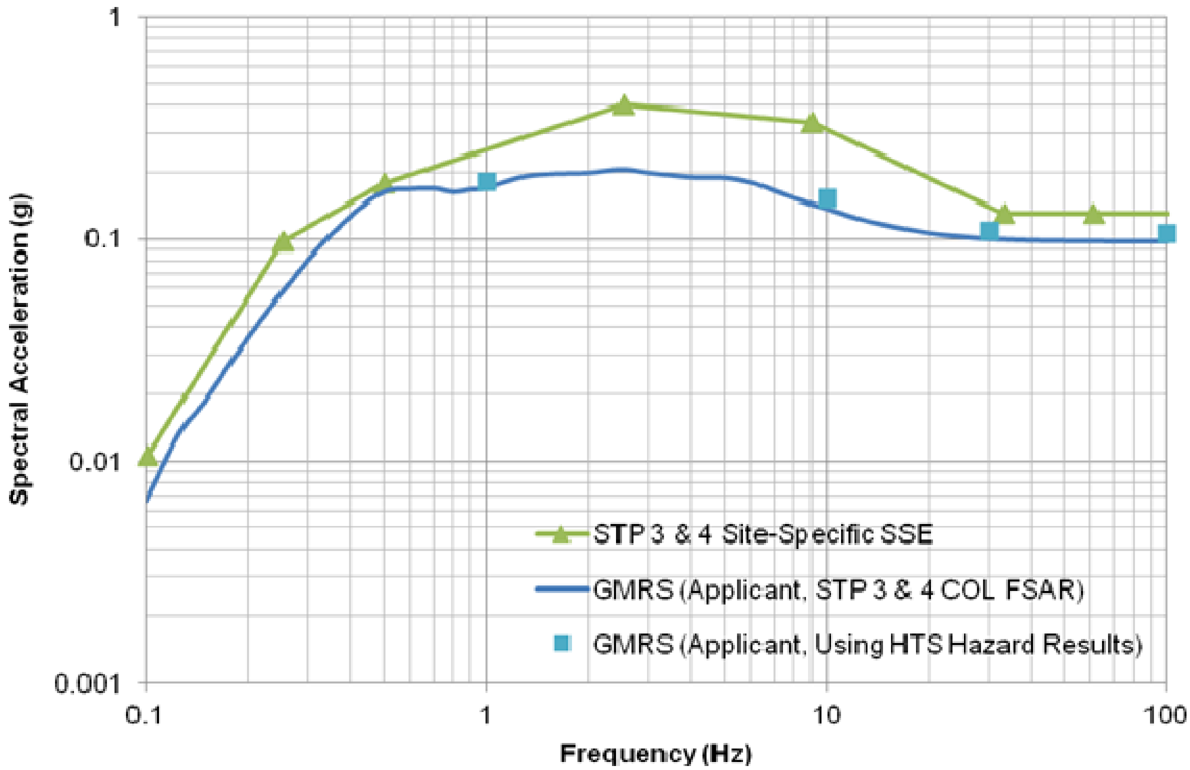

Figure 22.1-6 Comparison of the Horizontal STP Units 3 and 4 Site-Specific SSE—the STP Units 3 and 4 COL Application GMRS—and the Houston Test Site CEUS SSC GMRS (Taken from Figure 6 in the response to RAI 01.05-1 Dated December 6, 2012 [ML12346A445])

In order to confirm the applicant's assumption that the Houston Test Site hazard curves are appropriate for estimating reference rock seismic hazard curves at the STP site, the staff performed a confirmatory PSHA for the STP site and the Houston Test Site and compared the results with those in NUREG–2115 for the Houston Test Site. The staff used the CEUS-SSC model (NUREG–2115) with the EPRI (2004, 2006) GMM. The staff used the NUREG-2115 Houston Test Site distributed seismicity sources in this confirmatory PSHA. The staff compared the confirmatory 1-, 10-, and 100-Hz hazard curve results for both the STP site and the Houston Test Site with the Houston Test Site results in the NUREG–2115 report for the distributed seismicity sources. The staff determined that the two sets of results are almost identical. This comparison is illustrated in Figures 22.1-7 through 22.1-9 of this SER. The figures show the PSHA hard rock seismic hazard curve results for 1, 10, and 100 Hz, respectively, for the distributed seismicity sources. These figures also show a comparison of the staff's results for the STP site and the Houston Test Site, which indicates that the STP hazard curves are close to or less than the Houston Test Site hazard curves at frequencies of 1, 10, and 100 Hz.

Based on the above assessment, the staff concludes that the COL applicant's use of the NUREG–2115 hazard curves for the Houston Test Site at 1 Hz, 10 Hz, and 100 Hz is an adequate substitute for performing hazard calculations for the STP site using the CEUS-SSC model. Because the RLME sources are quite distant from both the Houston and STP sites, the staff only used the distributed seismicity sources in the confirmatory PSHA analysis. Including the RLME sources (i.e., the NMFS) would result in the same conclusion, because the STP site is located slightly further from the NMFS than the Houston Test Site is.

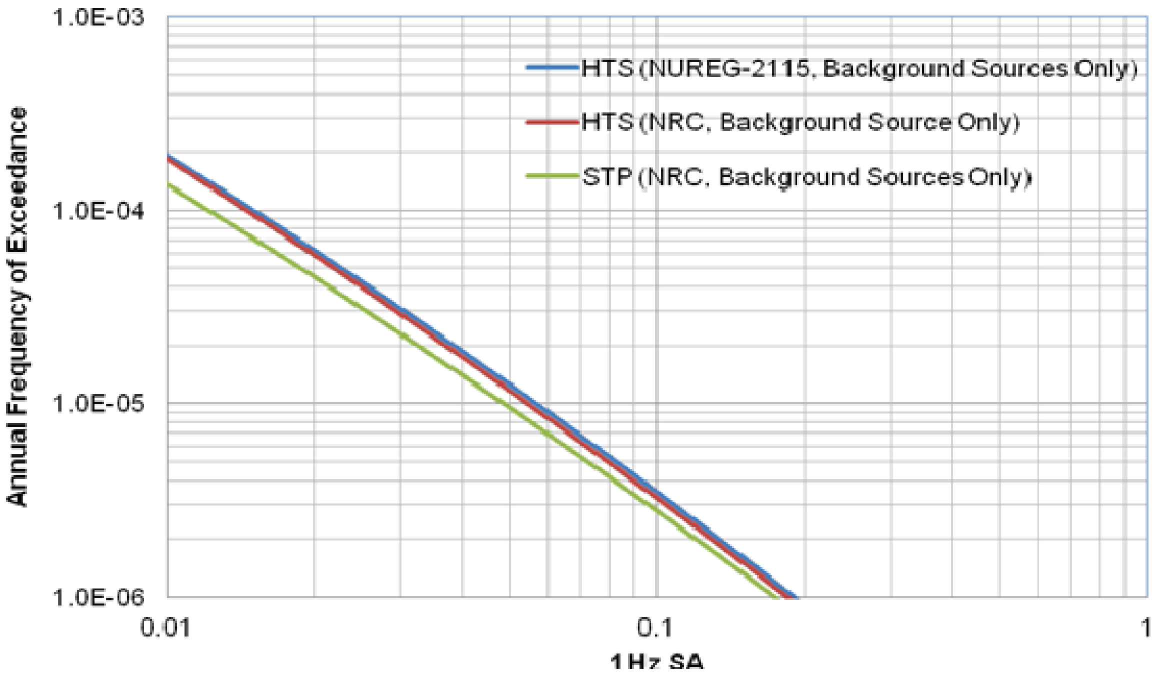

Figure 22.1-7 Plot Comparing the Staff's 1-Hz Total Mean Hazard Curves for the Distributed Seismicity Source Zones for the STP Site and the Houston Test Site (HTS). (Also Shown Is the NUREG–2115 1-Hz Total Mean Hazard Curve for the Distributed Seismicity Source Zones for the Houston Test Site)

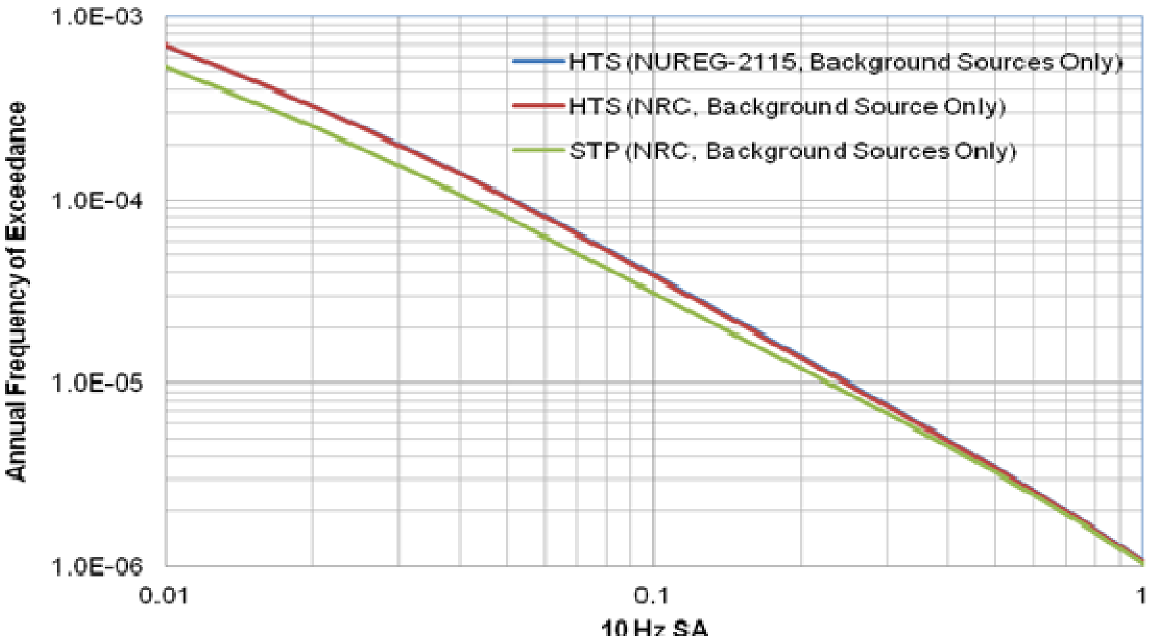

Figure 22.1-8 Plot Comparing the Staff's 10-Hz Total Mean Hazard Curves for the Distributed Seismicity Source Zones for the STP Site and the Houston Test Site (HTS). (Also Shown Is the NUREG–2115 10-Hz Total Mean Hazard Curve for the Distributed Seismicity Source Zones for the Houston Test Site)

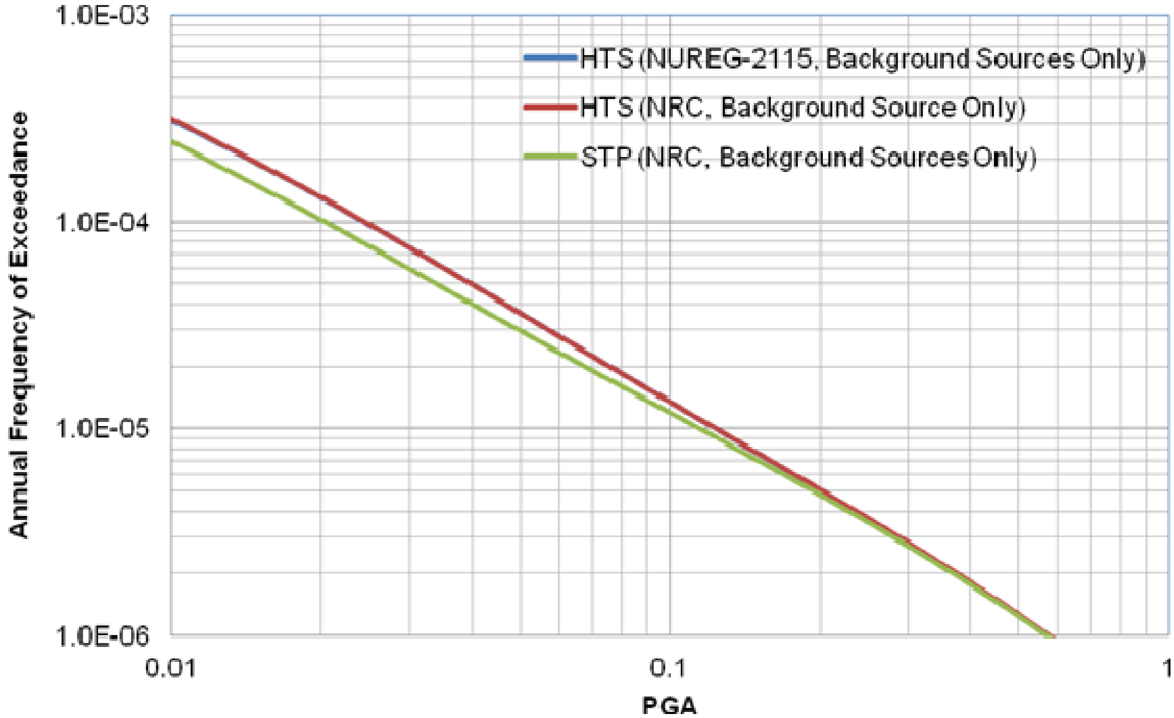

Figure 22.1-9 Plot Comparing the Staff's 100-Hz Total Mean Hazard Curves for the Distributed Seismicity Source Zones for the STP site and the Houston Test Site (HTS). (Also Shown Is the NUREG–2115 100-Hz Total Mean Hazard Curve for the Distributed Seismicity Source Zones for the Houston Test Site)

The staff also performed confirmatory site response calculations to determine the adequacy of the applicant's GMRS. For these calculations, the staff used the information on static and dynamic soil properties in COL FSAR Subsection 2.5S.2.5. The staff used 7 spectral frequencies and 11 input rock amplitudes. Figure 22.1-10, "Comparisons of the Staff's Site Response Amplification Functions with the COL Applicant's Amplification Functions," of this SER shows a comparison of the staff's mean site amplification function results with the COL applicant's results.

As shown in Figure 22.1-10, the COL applicant's and the staff's results are similar. The staff then used the amplification functions and the CEUS-SSC hard rock hazard curves for the Houston Test Site to develop probabilistic soil hazard curves that adhere to the methodology described in EPRI Technical Report 102587, "Seismic Evaluation Guidance," (EPRI, 2012) and NUREG/CR–6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines," (McGuire, et al., 2001). The staff then developed a GMRS at seven spectral frequencies using an approach that is consistent with RG 1.208. The staff's GMRS is depicted in Figure 22.1-11, "Comparisons of the Horizontal STP Units 3 and 4 Site-Specific SSE, the STP Units 3 and 4 COL Application GMRS, the Applicant's GMRS Based on Houston Test Site CEUS-SSC Hazard Results, and the NRC Staff's GMRS Using the Houston Test Site CEUS-SSC Hazard Results with the Updated EPRI GMM (EPRI, 2013)," of this SER.

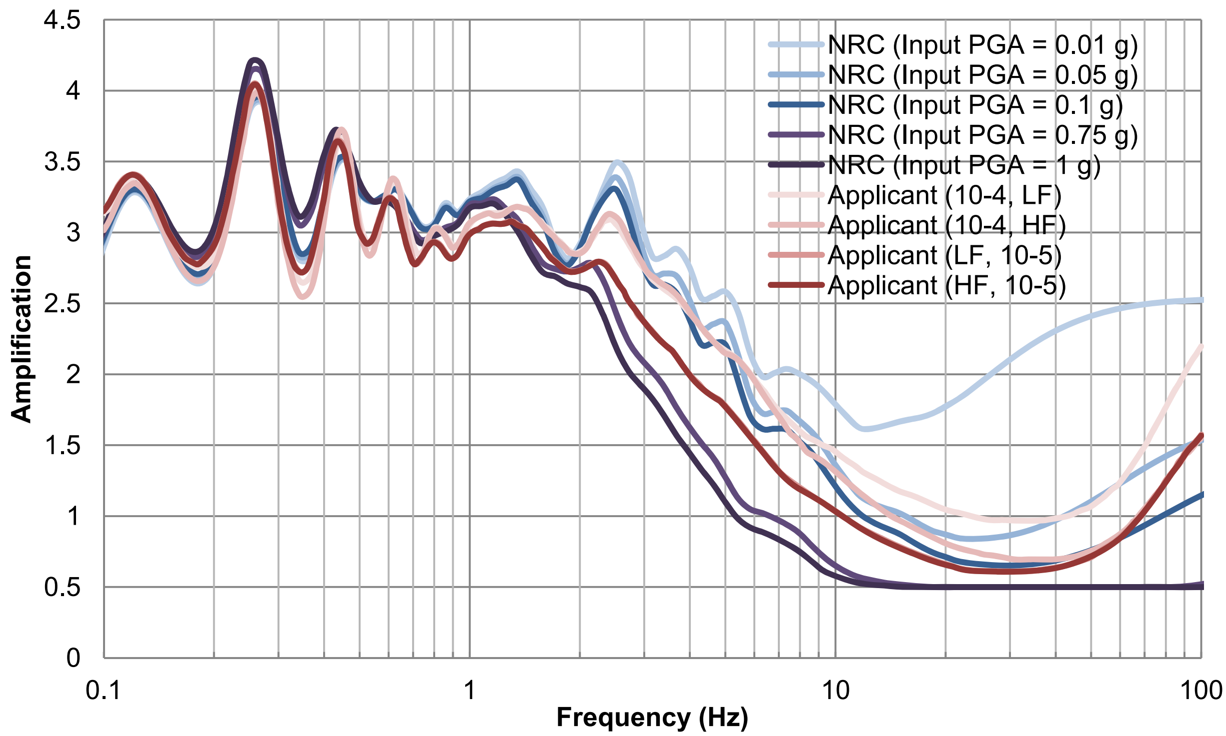

Figure 22.1-10 Comparisons of the Staff's Site Response Amplification Functions with the COL Applicant's Amplification Functions (Taken from FSAR Figures 2.5S.2-39, 2.5S.2-41, 2.5S.2-43, and 2.5S.2-45)

In a letter dated August 28, 2013 (ML13233A102), the staff stated that the updated GMM is an acceptable model for use by CEUS plants in developing a plant-specific GMRS. The staff calculated the STP GMRS using the CEUS-SSC Houston Test Site hazard results with the Updated EPRI (2004, 2006) GMM (EPRI, 2013). This GMRS is plotted in Figure 22.1-11 of this SER (the light brown curve) and is lower than the STP, Units 3 and 4, COL application site-specific GMRS (dark blue curve) for the entire 0.5- to 100-Hz frequency range with the exception of 10 Hz. Although the STP, Units 3 and 4, GMRS is exceeded by the staff's CEUS-SSC GMRS at 10 Hz by about 20 percent, the CEUS-SSC GMRS developed by the staff falls well below the STP, Units 3 and 4, site-specific SSE. The staff notes that it used a different site response method to develop the CEUS-SSC GMRS than the COL applicant used to develop the STP, Units 3 and 4, GMRS. The COL applicant used Approach 2 while the staff used Approach 3, which are described in NUREG/CR–6728. Using either Approach 2 or Approach 3 is consistent with the guidance provided in RG 1.208. Unlike Approach 3, however, Approach 2 does not explicitly incorporate the amplification function uncertainty into the development of the soil UHRS and subsequent GMRS. In a sensitivity calculation, the staff determined that neglecting the amplification function uncertainty resulted in a reduced GMRS across the entire frequency range and at 10 Hz, the STP, Units 3 and 4, GMRS is exceeded by the staff's CEUS-SSC GMRS by about 10 percent. In summary, the applicant's GMRS, which is set forth in FSAR Subsection 2.5S.2.6, "Ground Motion Response Spectra," and the staff's calculated GRMS are within the limits of uncertainty and both sets of results are bounded by the site-specific SSE. Based on the above comparison, the staff accepts the applicant's decision not to update the STP, Units 3 and 4, COL application site-specific GMRS and its site-specific SSE.

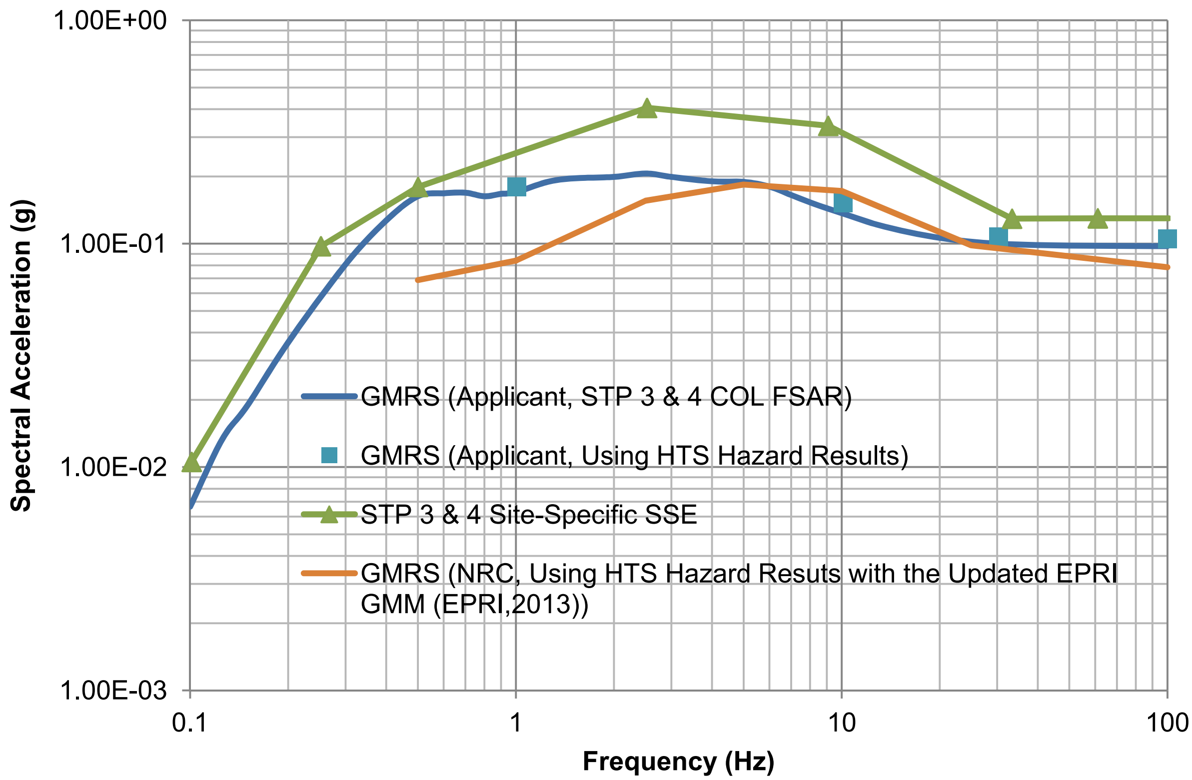

Figure 22.1-11 Comparisons of the Horizontal STP Units 3 and 4 Site-Specific SSE (Green Curve), the STP Units 3 and 4 COL Application GMRS (Dark Blue Curve), the Applicant's GMRS Based on the Houston Test Site CEUS-SSC Hazard Results (Blue Squares), and the NRC Staff's GMRS Using the Houston Test Site CEUS-SSC Hazard Results with the Updated EPRI GMM (EPRI, 2013) (Light Brown Curve)

22.1.4.2 NRC Staff's Conclusions Regarding the CEUS-SSC Sensitivity Calculations

The staff reviewed the applicant's response to RAI 01.05-1 (ML12346A445). Based on the staff's technical evaluation, which is set forth above, the staff reached the following conclusions:

1. The applicant's use of the NUREG–2115 demonstration hazard calculations for the Houston Test Site—instead of directly performing hard rock seismic hazard calculations for the STP site—is adequate because the staff's confirmatory analysis results showed that the CEUS-SSC hazard results at the STP site are similar to, or lower than, the hazard at the Houston Test Site.

2. The applicant adequately calculated the STP site-specific GMRS at 1, 10, 30, and 100 Hz using the CEUS-SSC model and the minimum magnitude cutoff of M 5, as recommended in SECY-12-0025, Enclosure 7, Attachment 1, to Seismic Enclosure 1 (ML12039A188).

3. The staff's site-specific GMRS, which is based on the CEUS-SSC model used for the Houston Test Site and the Updated EPRI (2004, 2006) GMM, is well below the STP, Units 3 and 4, COL FSAR GMRS calculated by the applicant using the updated EPRI-SOG model for the entire 0.5- to 100-Hz frequency range with the

exception of 10 Hz. The staff concludes that the about 20 percent exceedance of the STP, Units 3 and 4, GMRS by the staff's CEUS-SSC GMRS at 10 Hz is within the limits of uncertainties. Therefore, no revisions to the STP, Units 3 and 4, COL FSAR GMRS are necessary.

22.1.5 Post Combined License Activities

There are no post COL activities related to this section.

22.1.6 Conclusion

The staff reviewed the information submitted by the applicant in response to RAI 01.05-1 regarding the seismic hazard. As set forth above, the staff confirmed that the applicant has addressed the required information and has adequately evaluated the seismic hazards at the STP site against current NRC guidance in NUREG–0800, RG 1.60, RG 1.132, RG 1.206, RG 1.208, and DC/COL ISG-017; and the NRC requirements in 10 CFR 100.23; 10 CFR 52.79(a)(1)(iii); 10 CFR Part 50, Appendix A, GDC 2; and 10 CFR Part 50, Appendix S.

22.2 Mitigation Strategies (Recommendation 4.2)

22.2.1 Introduction

During the events at Fukushima Dai-ichi Nuclear Power Plant on March 11, 2011, the challenges faced by the operators were beyond any faced previously at a commercial nuclear reactor. In light of those events, the NRC determined that additional actions were needed to increase the capability of nuclear power plants to mitigate the effects of beyond-design-basis external events. The staff proposed Order EA-12-049 to impose additional requirements for such actions in Commission paper 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," (ML12039A103). In SECY-12-0025, the staff indicated that it would ensure that any additional actions necessary to increase the capability of currently licensed plants to mitigate beyond-design-basis external events would be addressed for design certification and COL applications submitted under 10 CFR Part 52 that were then under active staff review prior to certification or licensing. The staff also indicated that it would request all COL applicants to provide the information required of a licensee by any such order.

Subsequently, on March 12, 2012, the staff issued Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," (ML12054A735), which directed then-current licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities for a beyond-design-basis external event. The following documents the staff's evaluation of Nuclear Innovation North America LLC's (hereinafter referred to as the applicant) proposal for its mitigation strategies for a beyond-design-basis external event at STP, Units 3 and 4, in response to the matters consistent with Order EA 12-049.

22.2.2 Summary of Application

To address the Fukushima lessons learned, the applicant updated its FSAR and other documents to describe mitigation strategies for beyond-design-basis external events in response to staff RAIs. For example, on May 2, 2012, the staff issued to the applicant, RAI 01.05-4 (ML121230021). Based on SECY-12-0025, the letter requested the applicant to provide information related to the implementation of Fukushima NTTF Recommendation 4.2, regarding mitigation strategies. In its responses to RAI 01.05-4, dated June 25, 2012 (ML121850710), and December 6, 2012 (ML12346A445), the applicant submitted its responses to the Fukushima NTTF recommendations. The responses contained a copy of FSAR, Chapter 1, Appendix 1E, "Response to NRC Post-Fukushima Recommendations," addressing Recommendation 4.2, "Mitigation Strategies."

In its response to RAI 01.05-5 (ML13038A563), dated May 2, 2013 (ML13128A140), the applicant submitted a revised Appendix 1E, Section 1E.2.4, "Mitigation Strategies." The response also included the "STP 3&4 ABWR FLEX Integrated Plan," (hereinafter referred to as the "FLEX Integrated Plan"), which provides the basis for FSAR Appendix 1E, Section 1E.2.4. The "FLEX Integrated Plan" was updated to Revision 2, dated June 19, 2014, (ML14175A141). The "FLEX Integrated Plan" describes the guidance and strategies under development by the applicant for the maintenance or restoration of core cooling, containment, and SFP cooling capabilities following a beyond-design-basis external event, including necessary modifications to respond to the issues consistent with Order EA-12-049.

The applicant stated that all safety-related pumps, valves, and dynamic restraints used as part of the mitigation strategies for an extended loss of alternating current (ac) power (ELAP) event are permanently installed equipment. The applicant indicated that the design bases for these components, including the extent to which they are relied upon to perform, are discussed in the ABWR design control document (DCD) Tier 2 and in STP FSAR Chapter 3, "Design of Structures, Components, Equipment and Systems," Chapter 5, "Reactor Coolant System and Connected Systems," (Section 5.4.6), "Reactor Core Isolation Cooling System," and Chapter 6, "Engineered Safety Features," and Appendix 1E. The applicant indicated further that the components will not be relied upon to perform functions beyond those credited in the design basis during the implementation of the strategies discussed in the STP, Units 3 and 4, COL FSAR, Appendix 1E and the "FLEX Integrated Plan," except with respect to a specific aspect of the reactor core isolation cooling (RCIC) system which is discussed in Subsection 22.2.4.3.2 of this SER.

These changes are incorporated in Revision 12 of the STP COL FSAR.

22.2.3 Regulatory Basis

The regulatory basis for the staff review of beyond-design-basis external event mitigation strategies is as follows:

- The Atomic Energy Act of 1954, as amended, Section 161,"General Provisions," which authorizes the Commission to regulate the possession and use of special nuclear materials as necessary or desirable to protect public health and to promote 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.

The staff's guidance for beyond-design-basis external event mitigation strategies is established and described as follows:

- SRM-SECY-12-0025, which approves the 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.

- The Japan Lesson-Learned Project Directorate Interim Staff Guidance (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," (ML12229A174) dated August 29, 2012, which accepts the methodology described in Nuclear Energy Institute (NEI) industry guidance document NEI 12-06 Revision 0, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," (ML12242A378) with exceptions and clarifications.

22.2.4 Technical Evaluation

In Order EA-12-049, the Commission determined that ensuring adequate protection of public health and safety requires that power reactor licensees and construction permit holders 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. Therefore, the Commission modified all current licenses and construction permits to include requirements for such guidance and strategies. As a result, the staff is evaluating, among other things, the availability of diverse and flexible mitigation strategies (FLEX) equipment to mitigate beyond-design-basis external events to determine whether the designs of proposed nuclear power reactor facilities provide reasonable assurance of adequate protection of the public health and safety. With respect to STP, Units 3 and 4, the applicant relies on various safety-related, nonsafety-related, and portable equipment as part of its FLEX mitigation strategy. In addition, the staff used the current industry and NRC guidance for FLEX mitigation strategies in evaluating the equipment used as part of the FLEX mitigation strategy for STP, Units 3 and 4.

The staff reviewed the applicant's submittals consistent with Order EA-12-049 using JLD-ISG-2012-01, Revision 0 (ML12229A174). The ISG accepts, with clarifications, that the methodologies described in NEI 12-06, Revision 0 (ML12242A378) are consistent with Order EA-12-049.

Order EA-12-049 states that mitigation strategies must be capable of mitigating a simultaneous ELAP and a loss of normal access to the ultimate heat sink (LUHS), while maintaining adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all of a licensee's operating units on a site. The applicant submitted FSAR, Tier 2, Chapter 1, Appendix 1E, and the "FLEX Integrated Plan" in response to the issues consistent with Order EA-12-049.

The staff reviewed the following key safety functions and available equipment proposed to demonstrate the mitigation capabilities at STP, Units 3 and 4:

- phased approach and acceptance criteria,
- core cooling,
- containment function,
- SFP cooling,
- power supply,

- water and fuel supplies,
- ventilation (control room habitability and equipment cooling),
- instrumentation and emergency lighting, and
- FLEX equipment and offsite resources.

The staff reviewed the following areas with respect to protection and the availability of the equipment under external hazards:

- evaluation of external hazards,
- protection of equipment (structure), and
- mechanical equipment capability and programmatic controls.

As described in more detail below, the staff reviewed the following areas regarding procedures, administrative controls, guidance, and training; in addition to the acquisition, staging, or installation of equipment needed for mitigation strategies:

- multiple units at the site;
- programmatic controls, including equipment maintenance and availability testing, and procedures; and
- use of license conditions.

22.2.4.1 *Evaluation of External Hazards*

Sections 4 through 9 of NEI 12-06 provide an NRC-accepted methodology for determining the applicable extreme external hazards, in order to identify potential complicating factors for the protection and deployment of equipment needed to mitigate beyond-design-basis external events leading to an ELAP and an LUHS.

On pages 9 and 10 of the "FLEX Integrated Plan," the applicant states that STP, Units 3 and 4, screens all external hazards in accordance with Sections 4 through 9 of NEI 12-06. All site-specific external hazards identified in Section 4.1, "Site-Specific Identification of Applicable Hazards," of NEI 12-06 are applicable to STP, Units 3 and 4. The applicant stated that the following external hazards must be considered for the STP, Units 3 and 4, site:

- Seismic – FSAR Section 2.5S, "Geology, Seismology and Geotechnical Engineering," includes the seismic criteria for STP, Units 3 and 4.

- External Flooding – FSAR Section 2.4S.2, "Floods," defines the applicable criteria.

- Extreme Winds – FSAR Section 2.3S, "Meteorology," contains the defined extreme wind conditions for storms, such as hurricanes, high winds, and tornados.

- Extreme Cold – FSAR Section 2.3S contains the defined extreme cold conditions, including snow and ice.

- Extreme Heat –FSAR Section 2.3S contains the defined extreme heat conditions.

The applicant stated further that, in accordance with NEI 12-06, Section 2.2, "Determine Applicable Extreme External Hazards," all of the postulated external hazards identified as applicable in this section will be considered for the following:

- Protection of the FLEX equipment.
- Deployment of the FLEX equipment.
- Procedural interfaces.
- Utilization of offsite resources.

The applicant's acceptance criteria for installed equipment are as follows: Permanent plant equipment used for mitigation must be protected against the above external events. According to NEI 12-06, this equipment should be designed to be robust and housed in robust buildings. Robust is defined in NEI 12-06, Appendix A, "Glossary of Terms," as "the design of an SSC either meets the current plant design basis for the applicable external hazards or has been shown by analysis or test to meet or exceed the current plant design basis" with respect to seismic events, floods, and high winds and associated missiles. The applicant addresses the above considerations in various parts of the "FLEX Integrated Plan," including additional capability, equipment, protection of equipment, procedures, and offsite resources in the following subsections of this evaluation.

The staff reviewed the information in the "FLEX Integrated Plan" addressing issues associated with the protection of the SSCs from the external events listed above. The review of the mitigation equipment and the protection levels for external events is initially set at a design-basis or equivalent level.

NRC Order EA-12-049 addresses beyond-design-basis external events, which may damage internal plant SSCs or cause widespread damage to offsite power sources, thus preventing their rapid restoration. Accordingly, Order EA-12-049 specifies mitigation strategies beyond the coping capability required by 10 CFR 50.63, "Loss of all alternating current power," (referred to as the Station Blackout [SBO] rule). Order EA-12-049 specifies a diverse and flexible additional capability that uses guidance, strategies, and multiple sets of equipment to mitigate ELAP and LUHS conditions and to maintain or restore core cooling, containment, and SFP cooling capabilities. In contrast to the capabilities required by the SBO rule, the additional capabilities include SFP cooling, and the capability to address a loss of ac sources for an extended duration. Order EA-12-049 also addresses protection of equipment relied upon to accomplish the mitigation strategies from beyond-design-basis external events. The applicant addresses additional diverse and flexible capabilities in the "FLEX Integrated Plan," and the following SER subsections discuss the staff's evaluation.

22.2.4.2 *Phased Approach and Acceptance Criteria*

NRC Order EA-12-049 specifies 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 key safety functions that include core cooling, containment, and SFP cooling. The transition phase involves the provision of sufficient portable onsite equipment and consumables to maintain or restore these functions, until they can be accomplished with offsite resources. The final phase involves sufficient offsite resources to sustain those functions indefinitely. Order EA-12-049 specifies that the strategies must be capable of mitigating a simultaneous ELAP and LUHS; and must provide capacity sufficient to address the challenges to core cooling, containment, and SFP cooling to prevent fuel damage for all of an applicant's units on a site.

While Order EA-12-049 specifies the guidance and strategies to mitigate the loss of all ac power sources and the loss of normal access to the ultimate heat sink (UHS), it does not explicitly address consideration of the loss of safety-related direct current (dc) batteries or any other equipment from such an event. The guidance in NEI 12-06 provides mitigation methods for beyond-design-basis external events that are consistent with Order EA-12-049 and acceptable to the staff. In regard to installed plant equipment, NEI 12-06 recommends that mitigation strategies which rely on such systems should be manually operated (i.e., without reliance on ac power) (NEI 12-06, Table C-1, "Summary of Performance Attributes for BWR Core Cooling Function," and Table D-1, "Summary of Performance Attributes for PWR Core Cooling Functions") and the strategies should provide for connecting instrument sensors to portable instrumentation at containment penetrations (NEI 12-06, Section 5.3.3, "Procedural Interfaces"). However, the guidance and strategies may rely on installed equipment other than ac sources if that equipment is designed to withstand the severe natural phenomena that are characteristic of the site or is otherwise protected from such phenomena, and is designed to function in the environmental conditions in which the strategies rely on that equipment and for the duration during which the strategies rely on it.

The STP, Units 3 and 4, UHS is a Seismic Category I structure, with an enclosed concrete flood-protected basin and UHS pumps and valves located in a subsurface protected structure. In addition, as stated in FSAR, Tier 2, Chapter 1, Appendix IE, Section 2.4, a permanent piping connection to allow the ac-Independent Water Addition (ACIWA) system to take suction from the water volume in the UHS basin will be installed with the appropriate separation of the safety-related and nonsafety-related systems. The ACIWA diesel-driven pump is operated manually; there is no automatic operation other than the start of the ACIWA pump on a loss of offsite power as discussed in DCD Tier 2, Subsection 5.4.7.1.1.10. Because of the robust nature of the UHS and the safety-related piping, the applicant expects the UHS (safety-related forced draft cooling tower and fans) to survive a beyond design basis external event. Therefore, even if normal access to the UHS is lost, the applicant described an alternative means of accessing the UHS. This is further discussed in the water and fuel supplies Subsection 22.2.4.2.5 and structure Subsection 22.2.4.3.1 of this SER. The applicant will also install permanent electrical connections to allow the Phase 3 FLEX 480V 1500 kW diesel generators (DGs) to be connected outside the Reactor Building and provide power to the Engineered Safety Feature (ESF) Load centers.

The applicant's proposed acceptance criteria for core cooling, containment function, and SFP cooling for STP, Units 3 and 4, mitigation strategies are as follows:

- Core Cooling – There is adequate core cooling, and no fuel damage will occur throughout the event.

- Containment Function – The structural integrity of the containment is maintained throughout the event. The containment overpressure protection system rupture disk actuates at a pressure of approximately 90 pounds per square inch gauge (psig) (0.62 mega Pascal gauge [MPaG]), which is below the drywell head and the ultimate containment pressure limits.

- Spent Fuel Pool Cooling – Adequate cooling for spent fuel is provided by keeping the spent fuel in the SFP covered with water 10 feet (3 meters) above the top of the fuel racks.

The staff issued RAI 01.05-4, requesting the applicant to address Fukushima NTTF Recommendation 4.2 (ML121230021). In its responses to RAI 01.05-4, dated June 25, 2012, (ML121850710) and December 6, 2012 (ML12346A445), the applicant addressed the question in RAI 01.05-4 related to NTTF Recommendation 4.2 in FSAR Appendix 1E, which was in

Attachment 5 of the December 6, 2012, submittal. The staff reviewed the FSAR Section 1E.2.4, and determined that the FSAR Section 1E.2.4 had not adequately documented capabilities or provided strategies for the specified mitigation functions. Therefore, the staff issued RAI 01.05-5, requesting the applicant to provide additional information on how the STP, Units 3 and 4, could address the issues specified in the NRC order. In the RAI, the staff provided additional guidance to clarify the staff's expectations for information contained in the application (i.e., in the FSAR or in supporting documents).

In its response to RAI-01.05-5 (ML13128A140), dated May 2, 2013, the applicant provided the "FLEX Integrated Plan" to address the phased approach, including core cooling, containment, and SFP cooling in the event of an ELAP. The applicant also provided a revised FSAR Appendix 1E. In its RAI response, the applicant stated that as a result of the advanced design features of the RCIC system, the ACIWA system (which is shared by both units), and the containment overpressure protection system (COPS), STP, Units 3 and 4, can support a Phase 1 coping ability for at least 36 hours using permanently installed plant equipment. The applicant reasoned that because Phase 1 is 36 hours in duration and offsite supplies can be delivered to the site from the regional response center within 32 hours from the start of the event, on-site temporary portable equipment is not needed to provide core cooling, containment, or SFP cooling for a Phase 2 coping ability. Thus, the applicant concluded that there will be a direct transition into Phase 3 at the end of Phase 1. The applicant described the two phases as follows:

(1) Initially cope by relying on installed plant equipment for 36 hours (Phase 1).

(2) Obtain additional capability and redundancy from offsite equipment until power, water, and coolant injection systems are restored or commissioned for an indefinite duration (Phase 3).

Because the duration of 36 hours is sufficiently long compared to the duration of 24 hours in the guidance of NEI 12-06 for the combination of initial phase and transition phases specified in Order EA-12-049, the staff finds that the proposed Phase 1 (36 hours) using the installed equipment for STP, Units 3 and 4, is adequate in regard to providing for the initial and transition phases. The proposed Phase 3 use of offsite equipment for an indefinite duration is consistent with Order EA-12-049. The proposed two-phase approach serves the same purpose as the three-phase approach consistent with Order EA-12-049 by providing for the same functions specified in Order EA-12-049. Therefore, the staff finds that the applicant's proposed two-phase approach is acceptable.

Included in the applicant's response to RAI 01.05-5, the "FLEX Integrated Plan" Figure 1, "FLEX Mitigating Strategy Summary Timeline," contains a summary level outline and provides the following estimated time of event occurrences for beyond-design-basis external event mitigation:

RCIC on Condensate storage tank (CST) Suction - 0 hours.

Containment Pressure Increasing - 0 hours.

No SFP Fill required - 0 hours.

RCIC on Suppression Pool (SP) Suction - <2 minutes.

RCIC on CST Suction - 10 hours.

Containment Vented after COPS Actuation - 20 hours.

ACIWA Available to Fill SFP as necessary - 20 hours.

ACIWA on Fire Water Storage Tank Suction - 36 hours.

Phase 3 FLEX Diesel Generator Operating - 36 hours.

ACIWA on UHS Suction - 72 hours.

Although not relied upon in Figure 1 of FSAR Section 1E.2.4, the applicant also indicated that additional portable equipment (including diesel-powered pumps, power supplies, hoses and fittings, and portable DGs) will be available on the site for use if needed.

The staff summarized the following mitigation criteria from the “FLEX Integrated Plan” for STP, Units 3 and 4:

- Core Cooling – Using RCIC and ACIWA systems to maintain adequate core cooling.
- Containment Function – The structural integrity of the containment is maintained throughout the event. The containment overpressure protection system rupture disk actuates at a pressure of approximately 90 psig (0.62 MPaG), which is below the drywell head and the ultimate containment pressure limits.
- Spent Fuel Pool Cooling – Adequate cooling for spent fuel is provided by keeping the spent fuel in the SFP covered with water 10 feet (3 meters) above the top of the fuel racks using ACIWA to makeup the water.

The staff reviewed the applicant’s criteria established for core cooling and SFP cooling, and finds them acceptable because fuel damage would be prevented if the proposed acceptance criteria are met. In addition, the staff finds the criteria to maintain containment function to be acceptable because the containment structural integrity is maintained, which will allow the containment to prevent an uncontrolled large release of radioactive materials. The staff also determined from the “FLEX Integrated Plan” that the containment is used as a heat sink for core cooling to prevent fuel damage. Based on the above, the staff finds that the applicant’s mitigation criteria stated above for core cooling, containment, and SFP cooling are acceptable.

As described above, the applicant’s information in the response to RAI 01.05-5, provides an overview of the FLEX Integrated Plan that is acceptable to the staff. In addition, the staff evaluated and documented its review of the individual features and activities depicted in Figure 1 (in the FLEX Integrated Plan, Revision 0, attached to the RAI response), in more detail in the following SER subsections. Therefore, RAI 01.05-5, is resolved and closed.

22.2.4.2.1 Core Cooling

In its response to RAI 01.05-5, dated May 2, 2013 (ML13128A140), the applicant addressed strategies to provide core cooling as described in the “FLEX Integrated Plan.” In addition, the applicant supplemented this response with subsequent responses to RAI 01.05-11 through RAI 01.05-15, in its letter dated July 23, 2013 (ML13211A303). The following information in the

ABWR DCD, STP, Units 3 and 4, FSAR Appendix 1E, and the above RAI responses are relevant to the staff's review of core cooling strategies.

ABWR DCD Subsection 19E.2.2.3, "Station Blackout with RCIC Available (SBRC)"

This subsection is incorporated by reference into the STP, Units 3 and 4, FSAR and describes an SBO sequence with a failure of the combustion turbine generator (CTG) (SBRC is the designation accident sequence code for an SBO with RCIC operating for eight hours). This event is characterized by the unavailability of all ac power sources, except for those obtained from batteries through inverters. Because the RCIC system is steam driven and the firewater system does not rely on ac sources other than obtained from safety-related batteries through inverters, they are available for core cooling. The DCD states that the RCIC can be operated in this SBO sequence for approximately eight hours due to the dc battery capabilities, to provide core cooling (see DCD Tier 2, Subsection 19E.2.1.2.2, "Performance During Station Blackout with Failure of the Combustion Turbine Generator"). However, as stated in the "FLEX Integrated Plan," the applicant states that the RCIC operation may be extended to about 36 hours, which is discussed below, and the ACIWA system can be manually started after about 36 hours of RCIC operation. The staff confirmed during audits performed on August 21, 2013, and September 20, 2013, (ML14129A248) that the Modular Accident Analysis Program (MAAP) analysis results of Calculation NSO-2013-000311/PSNN-2013-0513, Revision 001, showed that the steam driven RCIC will operate for 36 hours.

The staff reviewed and verified the assumptions and initial conditions used in the analysis. As described in the "FLEX Integrated Plan," before the RCIC stops operation, the operator depressurizes the reactor pressure vessel (RPV) and begins injection into the RPV using the ACIWA system, which is powered by a diesel-driven pump. The staff has determined that the ACIWA system uses a diesel-driven pump and can be aligned to take suction from several water sources, and the firewater addition system can be used to provide water to the core. In particular, the design includes permanent piping to allow the ACIWA system to take suction from the water volumes in the UHS basins and is discussed later in this SER. The calculation confirms that core cooling can indeed be maintained indefinitely in this scenario. Accordingly, the staff concludes that core cooling can be maintained indefinitely.

FSAR Appendix 1E, Section 1E.2.4, "Mitigation Strategies for Beyond Design Basis Events (4.2)"

The proposed revision to FSAR Section 1E.2.4 in Attachment 2 of the May 2, 2013, response stated that the basis in FSAR Section 1E.2.4 for mitigating a beyond-design-basis external event is the "FLEX Integrated Plan" in Attachment 1 of the same submittal. The "FLEX Integrated Plan" describes the design features, equipment capabilities, and the sequence of events timeline for an ELAP, and the phased approach.

In the "FLEX Integrated Plan," the applicant described how the facility will respond in the event of an ELAP, as follows: The reactor will trip on the loss of all ac power. The main steam isolation valves (MSIVs) will close and the reactor internal pumps will coast down. The RPV pressure will increase and the safety/relief valves (SRVs) will open with steam discharged into the SP. The RCIC will start automatically for core cooling on reactor water low-level. The operators will follow the Emergency Operating Procedures (EOPs) to start implementing the EOP/Abnormal Operating Procedure (AOP) guidance for an SBO within 30 minutes. The emergency DGs (EDGs) and CTGs are assumed to be unavailable. A determination will then be made as to whether an ELAP has occurred. Command and control will be transferred to the remote shutdown panel, and an operator will be dispatched to manually operate the RCIC. Deep load shedding for dc batteries to lengthen battery life will be performed within 60 minutes. With the RCIC turbine exhaust and the SRVs discharge into the SP, SP temperature will increase. Drywell cooling will be lost as the result of the loss of ac power, and the containment pressure will increase. The COPS is expected to actuate automatically and vent the containment at approximately 20 hours into the event. The RCIC will operate for about 36 hours. The ACIWA system valves will be manually aligned after about 36 hours of RCIC operation to start injection via the residual heat removal (RHR) Loop C in the low-pressure core flooder mode.

The applicant indicates that STP, Units 3 and 4, include design features to mitigate a simultaneous loss of all ac power and a LUHS, including the following systems, equipment, and instrumentation used for core cooling:

- RCIC.
- Condensate storage tank (CST) and CST level instrumentation.
- SRVs.
- RPV level and pressure instrumentation.
- Drywell pressure instrumentation.
- ACIWA.
- SP level and temperature instrumentation.

Core Cooling Method for an ELAP that Occurs in Modes 1, 2, and 3

In the staff's evaluation of the response to RAI 01.05-5, dated May 2, 2013 (ML13128A140), the following describes the core cooling methods using the systems and equipment listed above to mitigate an ELAP initiated during Mode 1, 2, or 3.

As described in the "FLEX Integrated Plan," each unit includes a RCIC system and each of these systems includes a turbine-pump that can take suction from the CST or the SP. The "FLEX Integrated Plan" further describes the RCIC system as follows: The turbine-pump is a single-shaft unit that requires no external services (electrical, pneumatic, or lube oil) to operate. Only steam is needed for the RCIC turbine-pump to function; the turbine governor controls are self-contained on the unit. The bearings are water lubricated from the pump discharge and are designed to operate at high temperatures of about 250 degrees Fahrenheit (°F) (121.1 degrees Celsius [°C]). The design-basis operating temperature of the RCIC room is 150.8 °F (66 °C) (see DCD Tier 2, Chapter 19, Table 19E 2-2, "ABWR Plant Ability to Cope with Station Blackout for up to 8 Hours"). The RCIC room is designed not to exceed this temperature for up to eight hours of RCIC operation during SBO conditions (see FSAR, Chapter 5, Subsection 5.4.15.2.1, "Analysis to Demonstrate the Facility has a 8 Hour Non-Design SBO Capability"). For extended RCIC operation, the operators will establish a natural circulation path by blocking open a RCIC door and removing an overhead hatch for additional cooling. At the onset of the event, the RCIC will take suction from the CST. When the SP level high alarm set point is reached, which is estimated to be less than two minutes into the event, as shown in Figure 1 of the "FLEX Integrated Plan," Revision 1 (M14114A194), suction will be automatically switched to the SP.

As the SP temperature approaches 250 °F (121 °C), operators will manually shift the suction back to the CST. This transition back to the CST is expected to occur at approximately 10 hours into the event.

The “FLEX Integrated Plan,” also stated that RCIC operation is expected to continue for about 36 hours. The operators will consider the CST level in deciding when to initiate injection via the ACIWA, which is shared by both STP, Units 3 and 4, and secure RCIC. In either unit, the ACIWA system will first be aligned to RHR Loop C in the low-pressure core flooder mode with the ACIWA pump already in operation. The operators will then depressurize the RPV to begin injection with the ACIWA system. One SRV will be opened to reduce the RPV pressure to below the shutoff head of the ACIWA pump, which is 284 psig (1.958 MPaG) (see ABWR DCD Tier 2, Chapter 19, Subsection 19E.2.2.3, “Station Blackout with RCIC Available (SBRC)”). Once the RPV pressure decreases below the shutoff head and the ACIWA system starts injecting, the RCIC can be secured and core cooling will be provided by the ACIWA. This transition is expected to be performed at approximately 36 hours. As discussed in DCD Tier 2, Chapter 5, Subsection 5.4.7.1.1.10.3, “ACIWA Flow rate,” the ACIWA design flow capacity is 634 gallons per minute (gpm) (40 liters per second [L/s]) at a backpressure of 90 psig (0.62 MPaG), and will increase to 951 gpm (60 L/s) at a backpressure of 0 psig (0 MPaG). At approximately 20 hours, the COPS rupture disk will activate at 90 psig (0.62 MPaG) to relieve drywell airspace pressure and after 36 hours, the RPV pressure should have fallen below 90 psig (0.62 MPaG) thus allowing the ACIWA injection flow capacity to be greater than 634 gpm (40 L/s). At 36 hours, the flow rate needed to remove decay heat will be about 165 gpm (10.4 L/s) per unit as described in Revision 1 of the “FLEX Integrated Plan.” Therefore, within 36 hours, the single ACIWA pump can provide sufficient flow to maintain the reactor vessel level for both units, if called upon to do so. If the pump fails, there are several backup portable pumps to perform the function as described in Subsections 22.2.4.2.5 and 22.2.4.2.8 of this SER.

For the core cooling calculations, the applicant assumes a CST volume of 250,000 gallons (946 cubic meters [m^3], out of a capacity of 550,000 gallons [2,080 m^3]), which is the low end of the normal operating range. The applicant states that the CST water is protected against extreme cold by immersion-type electric heaters.

As described above, RCIC initially takes suction from the CST and automatically switches to taking suction from the SP when the SP level high alarm set point is reached, after the loss of ac power. The “FLEX Integrated Plan,” Revision 2, Attachment 1, “Sequence of Events Timeline,” states that during the first hour of the event, command and control will be transferred from the control room to the remote shutdown panels, an operator is dispatched to manually operate RCIC. ABWR DCD Tier 2, Chapter 5, Subsection 5.4.6.1.2.2, “Manual Operation,” also states, “In addition to the automatic operational features, provisions are included for manual startup, operation, and shutdown of the RCIC System in the event initiation or shutdown signals do not exist or the control room is inaccessible.” Although the applicant’s preferred method of initially operating RCIC and switching RCIC suction to the SP uses ac power from the safety-related batteries through the inverters, RCIC operation in this manner can also be accomplished by manual RCIC operation. Because manual operation of RCIC can be accomplished within one hour, as stated in FLEX Integrated Plan, Attachment 1, “Sequence of Events Timeline,” core cooling can be maintained, which is acceptable. Moreover, the four independent Class 1E 125 volt dc (Vdc) batteries and their associated inverters are located within the control building, which is a Seismic Category I structure and has been evaluated for all site-specific external events and hazards with respect to seismic events, floods, and high winds and associated missiles as presented in FSAR Chapter 2. The remote shutdown system (RSS), located in the

Category I reactor building, provides a means to bring the reactor to hot shutdown and subsequent cold shutdown as described in ABWR DCD Tier 2, Subsection 7.4.1.4. The RSS is classified as a safety-related system in ABWR DCD Tier 2, Subsection 7.4.2.4. The system is designed to meet the applicable requirements in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," including GDC 2, "Design bases for protection against natural phenomena"; GDC 13, "Instrumentation and control"; and GDC 19, "Control room". It is also designed to conform to Institute of Electrical and Electronics Engineers (IEEE) Standard (Std) 279 "IEEE Criteria for Nuclear Power Plant Protection Systems," and IEEE Std 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits." Accordingly, the staff determined that the proposed provisions for initially operating RCIC as described in the "FLEX Integrated Plan" are acceptable.

Because significant credit is taken for the CST in the ELAP core cooling analysis, in RAI 01.05-12, the staff requested additional information to support the applicant's position that the CST is "robust." In its response to RAI 01.05-12, dated July 23, 2013 (ML13211A303), the applicant confirmed that the CST will be constructed to withstand the site-specific SSE of 0.13g, missile, flood, high wind, and other site-specific weather hazards. The staff also asked for a confirmation of the "robustness" of the RCIC pump automatic suction switchover from the CST to the SP. In this response, the applicant referred to ABWR DCD Tier 2, Chapter 9, Subsection 9.2.9, "Makeup Water Condensate System," and confirmed that the CST level instrumentation will be mounted in a safety-grade standpipe located in the reactor building secondary containment. Therefore, the switchover instrumentation will be protected from missiles, floods, high winds, and other site-specific severe weather hazards. In the evaluation of the response to RAI 01.05-12, the staff determined that the system(s) had a high probability of survival of a beyond-design-basis external event due to being designed as "robust" or located within Seismic Category I structures as provided in the guidance of JLD-ISG-2012-01. Therefore the applicant's response is acceptable, and RAI 01.05-12, is resolved and closed.

The staff determined that the operation of SRVs is critical to the mitigation strategy. Because the ACIWA is designed for low reactor pressure, the SRVs must open before the ACIWA can be used. To limit the amount of fluid leaving the reactor vessel, the reactor vessel low-low water level signal will also actuate the closure of the MSIVs. The SRVs may open automatically to relieve reactor pressure. As an alternative, the operator may also manually operate the pneumatic supply to open one or more of the SRVs to maintain a reactor pressure band, while there is sufficient dc power and pneumatic supply. Availability of dc power is further discussed in Subsection 22.2.4.2.4. Therefore in RAI 01.05-15, the staff requested the applicant to confirm that portable nitrogen bottles will be available to supplement the SRV accumulator. In its response to RAI 01.05-15, dated July 23, 2013 (ML13211A303), the applicant stated that there is no need for portable supplemental nitrogen bottles because the high-pressure gas supply system (see ABWR DCD Tier 2, Section 6.7) will have two divisions of permanently installed nitrogen gas bottles with each division having five installed nitrogen gas bottles with another five bottles installed in each division that provide backup to the normal supply. The applicant also stated that DCD Tier 2, Subsection 19E.2.1.2.2.2, Item (2)(b) further describes the backup nitrogen supply to the SRV accumulator and the operator actions that are necessary for its use during a SBO. In the evaluation of the response to RAI 01.05-15, the staff determined that the SRV accumulator system has built-in defense-in-depth, as the applicant described, and therefore is acceptable. The staff also requested the applicant in RAI 01.05-15, to discuss the impact that the containment backpressure may have on the operation of the SRVs due to the high drywell pressure. The applicant referred to DCD Tier 2, Subsection 19E.2.1.2.2.2 Item (2)(b), which confirms that the automatic depressurization system (ADS) accumulators will have sufficient pressure and capacity to fully open in spite of the containment backpressure. Therefore, the staff considers RAI 01.05-15 to be resolved and closed.

As indicated above, the ACIWA system is shared by both units. In its response to RAI 01.05-15, dated July 23, 2013 (ML13211A303), the applicant stated that the ACIWA system (including the fuel supply) will be protected against floods and severe weather events. The applicant described the ACIWA system as follows: The ACIWA system provides a means for introducing water directly into the RPV when ac power is not available from either onsite or offsite sources. The ACIWA system initially takes suction from the firewater storage tanks (FWSTs). Both FWSTs have robust designs and will be constructed to withstand a site-specific SSE of 0.13g, missile, flood, high wind, and other site-specific severe weather events. The RHR system provides the piping and valves that connect the ACIWA piping to the RHR Loop C pump discharge piping. The connections for the ACIWA valves that are used to introduce the water flow into RHR Loop C are located in the reactor building and are readily accessible. The primary means for supplying water through this connection is the diesel-driven pump in the ACIWA system. The ACIWA is operated only manually; there is no automatic operation other than the start of the ACIWA pump on a loss of offsite power. The ACIWA system is discussed in DCD Tier 2, Subsection 5.4.7.1.1.10.

The applicant also stated that the RCIC, SRVs, SP, and ACIWA are seismically qualified and are contained within “robust” structures such that they are adequately protected against the applicable site extreme hazards.

Furthermore, the applicant indicated that instrumentation used in this scenario is available on the RSS panels as hardwired instruments that are powered by Class 1E dc power. These instruments will not be affected by the shutdown of the plant’s computer systems and will therefore be available for use. The following key reactor and containment parameter indications and controls are relied upon in the evaluation:

- RPV level.
- RPV pressure.
- SRV control.
- Suppression pool level.
- Suppression pool temperature.
- Drywell pressure.
- CST level.

The core cooling strategy described above applies to Mode 1 (normal power operation). The strategy is similar in Mode 2 (startup) and Mode 3 (hot shutdown), except that the timing of events would be different. For Modes 4 and 5, the overall strategies for core cooling during cold shutdown and refueling are generally similar to those for power operation, startup, and hot shutdown except that the ACIWA system will be used first to inject water directly into the RPV. The core cooling strategy for Modes 4 and 5 is discussed below.

The staff reviewed the “FLEX Integrated Plan” regarding the RCIC operation for 36 hours. In RAI 01.05-14, the staff requested additional information to verify the basis for the 36-hour operation. In its response to RAI 01.05-14, dated July 23, 2013 (ML13211A303), the applicant referred to calculation NSO-2013-000311, Revision 1, dated August 8, 2013, which was made available for the staff's audit. During the audit on August 21, 2013, and September 20, 2013 (ML14129A248), the staff reviewed this engineering calculation, which was performed using the MAAP code. The calculation confirms that there will be adequate core cooling during the first 36 hours after the onset of the event. The calculation also confirms that RPV level is significantly higher than the top of the active fuel and clarifies the use of the RCIC operation for 36 hours. The assumptions, the methods used in the calculation, and the results confirm the RAI response. Therefore, RAI 01.05-14 is resolved and closed.

Core Cooling for an ELAP initiated during Cold Shutdown and Refueling (Modes 4 and 5)

In the "FLEX Integrated Plan," the applicant indicates that the overall strategies for core cooling during cold shutdown and refueling are generally similar to those for power operation, startup, and hot shutdown. The "FLEX Integrated Plan" states that if an ELAP occurs during cold shutdown (Mode 4), water in the vessel will heat up; an SRV will be opened; and the ACIWA system will be used to maintain the RPV level and thus provide water for core cooling. During refueling, the most limiting condition for providing cooling is the case in which the reactor head is de-tensioned, and the water level in the vessel is at or below the reactor vessel flange. If an ELAP occurs during this condition, the ACIWA system would be used to restore and maintain the water level in the reactor cavity above the vessel flange.

The staff determined that the ACIWA system is installed equipment that functions as Phase 1 equipment for the FLEX mitigation strategy. As previously discussed, the use of ACIWA is acceptable for core cooling. However, it should be noted that in Modes 4 and 5, the need for ACIWA operation and operator actions should be much sooner than that in Modes 1, 2, and 3, because the RCIC system is not available. This affects the operation of ACIWA and should be addressed in the procedures and training. Therefore, the staff issued RAI 01.05-23, requesting additional information. The evaluation of this issue is in Subsection 22.2.4.5 and Section 22.2.5, of this SER.

Core Cooling Summary

As described above, the design features of the RCIC and ACIWA systems in STP, Units 3 and 4, can support the plant's operation for at least 36 hours using installed plant equipment. Because Phase 1 is 36 hours in duration and offsite supplies can be delivered to the site from the regional response center within 32 hours from the start of the event, there is no gap between the use of installed equipment and the availability of offsite equipment. Because of the duration the RCIC and ACIWA systems can provide core cooling during Phase 1 and the arrival timing of offsite resources for Phase 3, Phase 2 is essentially eliminated. Accordingly, there is no need for temporary portable equipment to provide core cooling for a "Phase 2." Rather, the direct transition into Phase 3 at the end of Phase 1 will provide adequate protection of public health and safety for core cooling. Nonetheless, the applicant will provide portable equipment on the site to lend additional flexibility and defense-in-depth to the mitigation strategies as described above. Long-term water and fuel supplies for the ACIWA system will be discussed later in Subsection 22.2.4.2.5 of this SER.

As described above, the staff evaluated the information related to core cooling for an ELAP as presented in the "FLEX Integrated Plan" and the applicant's responses to the staff's RAIs. For the reasons set forth above, the staff concludes that with the systems and equipment listed in the evaluation adequate core cooling can be maintained for all modes of operation for both STP, Units 3 and 4. The adequacy of the procedures and training, which are not available and cannot be reviewed by the staff at this time, are addressed in Subsections 22.2.4.5 and Section 22.2.5, of this SER.

22.2.4.2.2 Containment Function

For operation in all modes, the "FLEX Integrated Plan" includes provisions for maintaining containment capability. Specifically, for operation in Modes 1, 2, 3 and 4, in the "FLEX Integrated Plan," Revision 1 (page 30 of 60) the applicant states the following (Mode 5 is discussed separately below):

> The ABWR has design features to mitigate a simultaneous ELAP and LUHS. The RCIC system and ACIWA are the primary systems used to provide core cooling. The COPS is the primary means for providing containment cooling.
>
> The containment design pressure is 45 psig (FSAR Subsection 6.2.1.1.2.1), the ultimate strength of the containment is 133.7 psig (FSAR Subsection 6.2.1.1.10), and the drywell head allowable pressure is 96.7 psig (FSAR Subsection 6.2.1.1.10). Containment structural integrity is maintained during the event because the COPS rupture disk actuates at a nominal pressure of approximately 90 psig (FSAR Subsection 6.2.5.2.6.3) prior to exceeding the drywell head and the ultimate containment pressure limits.

In the "FLEX Integrated Plan," the applicant describes the timing of the events and the decisions made during an ELAP without providing references to supporting calculations. For example, the "FLEX Integrated Plan" states that the steam-driven RCIC pump will operate for 36 hours and the COPS rupture disk will open at 20 hours into the event, without providing references to supporting calculations. The staff needed the supporting references in order to audit the calculations, if appropriate, and to determine whether the presented strategies are acceptable. Therefore, in RAI 01.05-16, the staff requested the applicant to provide this information.

In its response to RAI 01.05-16, dated July 31, 2013 (ML13218A290), the applicant stated that Calculation NSO-2013-000311/PSNN-2013-0513 provides the bases for the events and decisions documented in the "FLEX Integrated Plan," which would be made available for the staff's audit. On August 21, 2013, and September 20, 2013, the staff audited "STP-3/4 MAAP Analysis for SBO Sequence Design Report," NSO-2013-000311/PSNN-2013-0513, Revision 001, dated August 8, 2013. This analysis was performed by Toshiba Corporation under the Toshiba Quality Assurance (QA) Program, which complies with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The staff confirmed that the MAAP analysis results show that the steam-driven RCIC pump will operate for 36 hours and the COPS rupture disk will open at 22 hours into the event, which is 2 hours later than the 20 hours reported in the applicant's "FLEX Integrated Plan." Because a shorter COPS rupture disk opening represents a faster rate of containment pressurization, the staff determined that the applicant's 20 hours stated in the "FLEX Integrated Plan" is conservative. Therefore, the staff's concerns are resolved, and RAI 01.05-16 is resolved and closed.

ABWR DCD Tier 2, Subsection 6.2.4.3.2.2.2.3, states that the containment vent isolation valves "close on the following signals: high drywell pressure, RPV low water level 3, and high radioactivity in the purge and vent exhaust line." However, the essential containment instrumentation listed in the "FLEX Integrated Plan" Revision 0 (e.g., pp. 31 and 34) in the submittal dated May 2, 2013, does not include instrumentation for measuring radioactivity that is needed for isolating the containment on high radioactivity in the purge and vent exhaust line. In addition, the solenoids for the COPS isolation valves might not be able to operate during an ELAP if ac power is needed to operate them. The staff needed this information to determine whether the presented strategies are acceptable. Therefore, in RAI 01.05-17, the NRC requested the applicant to provide this information.

In its response to RAI 01.05-17, dated July 31, 2013 (ML13218A290), the applicant provided the following information to address the NRC staff's concern regarding the ability to isolate the containment during an ELAP, if needed:

> The STP 3&4 ABWR FLEX Integrated Plan describes mitigation strategies that increase defense-in-depth for a beyond-design basis external event including an Extended Loss of AC Power (ELAP) and loss of normal access to the ultimate heat sink (LUHS). The FLEX Plan does not require that the containment be isolated to control release of radioactivity during the ELAP and LUHS and specifically requires that the Containment Overpressure Protection System (COPS) remain in service until core and containment cooling have been restored.
>
> During Phase 1 of the response to an ELAP and LUHS, normal methods of core and containment cooling are not available and suppression pool temperature and pressure will increase until the COPS rupture disks open at a nominal pressure of approximately 90 psig (FSAR 6.2.5.2.6.3). There is no other mechanism to remove heat from the containment other than venting steam through the COPS. Use of the COPS to cool the containment will continue throughout Phase 1 and Phase 3 until other methods of core and containment cooling are established.
>
> [T]he use of the COPS during an ELAP and LUHS significantly reduces the potential for containment structural failure, which would result in a much larger release of radioactivity. Therefore, the COPS would not be isolated to control release of radioactivity to the environment in order to prevent containment structural failure, especially if the accident progresses or conditions worsen. In order to minimize the release of radioactivity through the COPS, the COPS release point is from the wetwell airspace. This ensures that fission products are directed to the suppression pool via the SRVs, scrubbing any potential release.
>
> The COPS does include provisions to isolate the venting path if required after the plant is stable (FSAR Subsection 6.2.5.2.1) following restoration of core and containment cooling.

The staff reviewed the applicant's response and determined that the applicant had addressed the staff's concern regarding the ability to isolate the containment during an ELAP, because the use of the COPS would reduce the potential for containment structural failure that would have resulted in a higher radioactivity release.

Regarding the staff's concern about the availability of power to the solenoids for the COPS isolation valves, the applicant stated that the power source of the solenoids for the vent line isolation valves is the "vital AC (VAC)." The VAC system provides a source of 120 Volt uninterruptible power that is normally supplied from the Class 1E 125V DC system. The Class 1E 125 V DC system would be powered from the Class 1E batteries during Phase 1 of the FLEX Plan and by the Class 1E battery chargers during Phase 3 of the FLEX Plan." The applicant added that the isolation valves, which are normally open, will not be operated (i.e., will not be closed) during an ELAP until the core and containment cooling are restored.

ABWR DCD Tier 2, Figure 6.2-39, “Atmospheric Control System P&ID,” shows that the conditions of the COPS isolation valves (F007 and F010) are normally open and fail open. ABWR DCD Tier 1, Table 2.14.6, “Atmospheric Control System,” states that the COPS isolation valves will fail open on loss of pneumatic pressure or loss of electrical power to the valve actuating solenoid. ABWR DCD Tier 2, Figure 6.2-39, also notes that the COPS isolation valves are not provided with an isolation signal.

After reviewing the applicant’s response and the information in the ABWR DCD, the staff determined that power will be available to the solenoids of the COPS isolation valves during an ELAP because that power is provided through inverters from the Class IE batteries. However, the valves will not be operated during an ELAP until core cooling and containment cooling is restored. This explanation addresses the staff’s concern regarding the availability of power to the solenoids for the COPS isolation valves, because the normally open COPS vent path will prevent pressurizing the containment by releasing heat through venting from the top of the SP air space to the atmosphere via the plant stack.

Regarding the staff’s concern about instrumentation for measuring radioactivity in the COPS vent line, the applicant stated the following:

> The COPS remains in service throughout Phase 1 and Phase 3 to provide containment cooling and minimize the potential for containment structural failure during an ELAP and LUHS.
>
> ABWR FSAR Tier 2, Section 6.2.4.3.2.2.2.3 describes ACS [atmospheric control system] isolation valves other than the COPS. These ACS containment isolation valves do “close on the high drywell pressure, RPV low water level 3, and high radioactivity in the purge and vent exhaust line.” These ACS containment isolation valves would receive an isolation signal from one or more isolation signals early in Phase 1 of the FLEX Plan.
>
> DCD Figure 6.2-39, Note 10, specifies that the COPS isolation valves “are not provided with an isolation signal,” which is consistent with the design function of the COPS.

ABWR DCD Tier 2, Subsection 6.2.4.3.2.2.2.3, states that “in the event of a radioactivity leak during inerting/de-inerting, the radiation detectors at the purge and vent exhaust line will detect the condition and isolate the ACS containment isolation valves.” ABWR DCD Tier 1, Section 2.14.6, states that the main control room has the control and the open/close status indication for the containment isolation valves, which fail closed upon the loss of pneumatic pressure or a loss of electrical power to the valve actuating solenoids.

In addition, FSAR Appendix 1E states COPS releases through the plant stack, which has a radiation monitor powered by Division 1E power. Therefore, the staff determined that the applicant’s justification for not including instrumentation for measuring radioactivity in the COPS vent line is acceptable, because radiation monitoring is performed through the plant stack and COPS remains in service (open) throughout Phase 1 and Phase 3 to provide containment cooling and to minimize the potential for the containment structural failure during an ELAP and

LUHS, and because the ACS containment isolation valves fail closed upon a loss of electrical power.

As described above, the applicant's response adequately addresses the staff's concerns raised in RAI 01.05-17. Therefore, this RAI is resolved and closed.

In Mode 5, the reactor vessel would be open to the secondary containment building. Water level will be maintained using ACIWA injection, and exterior doors in the reactor building will be opened to release heat to the atmosphere. While it should be possible to isolate the containment if the need arises to limit the radiological consequences, the "FLEX Integrated Plan" did not provide this information. The staff needed this information to determine whether the presented strategies are acceptable. Therefore, in RAI 01.05-18, the staff requested the applicant to provide this information.

In the July 31, 2013, response to RAI 01.05-18 (ML13218A290), the applicant stated the following:

> STP 3 & 4 Technical Specifications Limiting Condition for Operation 3.6.4.1 requires that the secondary containment will be OPERABLE in Mode 5 during movement of irradiated fuel in the secondary containment, during core alterations, and during operations with the potential for draining the reactor vessel. As such, during these periods the secondary containment will be isolated. At all other times in Mode 5 the secondary containment is not required to be OPERABLE.
>
> As described in FSAR section 13.5.3.1, "Plant Operating Procedure Development Plan", Emergency Operating Procedures and Abnormal Operating procedures will be developed prior to fuel load and will be inspected as part of the operational programs inspection. Procedures to respond to acts of nature (e.g., Tornado, flood, dam failure, earthquake) [FSAR 13.5.3.4.7(22)] are required as part of the Abnormal Operating Procedures. Response to a loss of all AC power is a required part of the Emergency Operating Procedures (FSAR 13.5.3.2). The operating procedures to respond to an Extended Loss of AC Power (ELAP) will provide a method to close the secondary containment.

The applicant's response adequately addresses the staff's concern, because a method to isolate the secondary containment will be provided in the EOPs and AOPs. Therefore, RAI 01.05-18 is resolved and closed.

The "FLEX Integrated Plan" maintains containment structural integrity solely by ensuring that the containment pressure limits are not exceeded. This is evident by the fact that the essential parameters for containment isolation listed on page 31 of the "FLEX Integrated Plan" do not include a drywell temperature. In general, excessive temperatures could result in a loss of containment integrity resulting from the failure of containment penetration seals or other portions of the containment boundary. Furthermore, excessive temperatures may need to be monitored to ensure that the qualification range of necessary measurement instruments located in the drywell is not exceeded. For these reasons, the staff requested the applicant to provide the basis for concluding that monitoring the drywell temperature is not necessary for purposes such as validating the qualification range of measurement instruments located in the drywell or establishing the survivability of penetration seals or other equipment. The staff needed this information to determine whether the strategies presented will be effective in maintaining containment in all phases of an ELAP. Therefore, in RAI 01.05-19, the staff requested the applicant to provide this information.

In the July 31, 2013, response to RAI 01.05-19 (ML13218A290), the applicant stated the following:

> Equipment in the drywell is to be designed and qualified for 171 degrees Celsius (339.8 degrees F).
>
> Calculation NSO-2013-000311/PSNN-2013-0513 documents that the maximum drywell temperature expected during an ELAP condition is approximately 332 degrees F. Since this temperature is less than the containment equipment design temperature, the penetration seals, instrumentation, and other equipment in the drywell will remain functional during ELAP conditions and monitoring of drywell temperature to validate equipment survivability is unnecessary.

On August 21, 2013, and September 20, 2013, the staff audited Calculation NSO-2013-000311/PSNN-2013-0513 and confirmed from the MAAP analysis results in the calculation report that the maximum drywell temperature during an ELAP condition is 332 °F (167 °C), which is below the equipment design temperature of 339.8 °F (171 °C). The staff determined that the applicant's justification for not monitoring the drywell temperature during an ELAP is acceptable, because it is based on MAAP analysis results showing a drywell temperature lower than the equipment design temperature. This information addresses the staff's concern, and RAI 01.05-19 is resolved and closed.

As described above, the staff evaluated the information related to containment cooling for an ELAP as presented in the "FLEX Integrated Plan," in the applicant's responses to RAIs and relevant sections in the ABWR DCD, and Calculation NSO-2013-000311/PSNN-2013-0513, which was available for the staff's audit. For the reasons described above, the staff concludes that with the noted systems and equipment in the "FLEX Integrated Plan," adequate containment cooling can be maintained for all modes of operation.

In the "FLEX Integrated Plan," the applicant stated that during Phase 1 the structural integrity of the containment is maintained by its normal design features, such as the COPS and the containment isolation valves. After the SP water becomes saturated, the containment will begin to heat up and pressurize. Additionally, the SP level will rise from the transfer of inventory to the SP (via the RCIC and then the ACIWA). The containment will continue to pressurize to approximately 90 psig (0.62 MPaG), at which point the COPS rupture disk will actuate and containment venting will commence at approximately 20 hours after event initiation. The structural integrity of the containment is maintained throughout the duration of the ELAP/LUHS. No portable equipment is relied upon to maintain the containment during Phase 1. The same methods used in Phase 1 are used in Phase 3 to maintain the structural integrity of the containment.

Therefore, the containment can be maintained for all modes of operation by using the COPS and the containment isolation valves throughout Phases 1 and 3 of the ELAP mitigation strategies consistent with the order. Portable equipment is not necessary to fulfill the COPS function. Therefore, RAI 01.05-19 is resolved and closed. Procedures and training are discussed below in Subsection 22.2.4.5 and Section 22.2.5.

22.2.4.2.3 Spent Fuel Pool Cooling

In the "FLEX Integrated Plan," the applicant states that there are no required actions during Phase 1 other than to monitor the SFP level. Maintaining the SFP full of water at all times during the ELAP event is not required; instead, the requirement is to maintain adequate level at all times to protect the stored spent fuel and limit exposure to personnel onsite and offsite. During normal operation, the SFP water level is 23 feet (7 meters) above the top of the fuel racks. During an ELAP event, the applicant's mitigation strategy is to allow the water in the SFP to boil, and the SFP water level to drop to 10 feet (3 meters) above the top of the fuel racks before any compensatory action. Assuming a full core offload (including a recently discharged full core) and no makeup, the SFP level will not reach 10 feet (3 meters) above the top of the fuel racks in the first 36 hours of the event. In the time-line for addressing the cooling of the SFP, the applicant indicates that the ACIWA is available before Phase 3 to fill the SFP after 20 hours from the start of the event. The installed ACIWA system will be aligned to provide water to the SFP while still providing core cooling. The applicant indicated in the "FLEX Integrated Plan" that calculations show that it will take a total of 76 hours from the start of the event for the water level to drop to 10 feet (3 meters) above the top of the fuel racks, assuming that there is no makeup.

On August 7, 2013, the staff audited (ML14129A248) Calculation SAV-2013-000105, Revision 1, dated July 19, 2013, titled "Spent Fuel Pool Heat-up and Inventory Loss Calculation," and confirmed the calculation results: due to boiling, and without any makeup, it takes 76 hours for the water level to reach 10 feet (3 meters) above the top of the fuel. Therefore, there is adequate margin (time) for the ACIWA equipment to start its makeup function, and the staff found these results acceptable.

Fuel in the SFP is cooled by maintaining an adequate water level above the top of the fuel. The staff finds that the proposed mitigation strategies can prevent fuel damage by maintaining SFP water more than 10 feet (3 meters) above the fuel racks for the first 36 hours by boiling the water in the pool. The duration of 36 hours is consistent with the provisions of the NRC Order regarding the initial and transition phases, because this strategy can maintain SFP cooling longer than the 24 hours recommended in the NEI 12-06 guidance for the combination of the initial and transition phases. The proposed Phase 3 strategies start at 36 hours into the event and use the ACIWA system to makeup the pool water as needed, consistent with the order for the final phase. The ACIWA can provide water makeup for an extended period of time without ac power. There are onsite and offsite portable pumps available as a backup to the ACIWA pump, which is described in Sections 22.2.4.2.5 and 22.2.4.2.8 of this SER. Therefore, other than the ACIWA realignment, the staff finds the proposed SFP mitigation strategies acceptable because the strategies can prevent fuel damage in the SFP assuming loss of ac power for an extended period of time.

In its response to RAI 01.05-22 dated October 16, 2013 (ML13294A121), the applicant clarified the ACIWA realignment. In Phase 3, SFP water makeup, if needed, is provided by manually opening valves F14C and F15C (Loop C double isolation between RHR Loop C and the fuel pool cooling). The connection at RHR Loop C is in the reactor building. These valves are ac motor operated valves (MOVs) as indicated in FSAR Figure 5.4-10, "Residual Heat Removal System P&ID," Sh. 7. The installed ACIWA pump provides flow to the RHR system piping and then to the SFP, as described in FSAR Section 9.1.3. Therefore, RAI 01.05-22, with respect to the valves being used for ACIWA realignment, is resolved and closed. Procedures and training for the realignment are discussed below in Subsection 22.2.4.5 and Section 22.2.5.

22.2.4.2.4 Power Supply

Section 8.3 of the applicant's FSAR incorporates by reference Section 8.3 of the certified ABWR DCD, Revision 4, referenced in 10 CFR Part 52, Appendix A.

As shown in Figure 8.3-4, "Plant Vital DC Power Supply System SLD," of the ABWR DCD Tier 2, Revision 4, the STP dc power systems design includes four independent Class 1E 125 VDC divisions, three independent non-Class 1E 125 VDC load groups, and one non-Class 1E 250 VDC computer and motor power supply. The 125 VDC Class 1E power is credited to supply ac power for emergency lighting, diesel-generator field flashing, control and switching functions such as the control of medium voltage and 480V switchgear, control relays, meters and indicators, vital ac power supplies, and dc components used in the RCIC system. Each 125 VDC battery is provided with a charger, and a standby charger is shared by two divisions. A non-Class 1E 125 VDC power supply is provided for non-Class 1E switchgear, valves, converters, transducers, controls and instrumentation. A non-Class 1E 250 VDC power supply is provided for the computers and the turbine turning gear motor.

The applicant's mitigating strategy for a beyond-design-basis external event assumes that the only available power sources during the first phase are the Class 1E station batteries. The applicant performed an analysis to determine how long the Class 1E battery capacity can be extended beyond that capacity credited in the design bases, and concluded that the battery discharge duration can be extended beyond 36 hours.

The staff in JLD-ISG-2012-01 provides guidance for a three-phase approach for mitigating beyond design basis external events. However, as described above, the staff determined STP, Units 3 and 4, can support a Phase 1 coping ability for at least 36 hours using permanently installed plant equipment. Because Phase 1 is 36 hours in duration and offsite supplies can be delivered to the site within 32 hours from the start of the event, there is no need for temporary portable equipment to provide core cooling, containment, or SFP cooling for Phase 2. Thus, there will be a direct transition into Phase 3 at the end of Phase 1, as stated in "FLEX Integrated Plan," Revision 2, dated June 19, 2014 (ML14175A141).

Because of the duration in which the RCIC/ACIWA systems can provide core cooling during Phase 1 and the arrival timing of offsite resources for Phase 3, Phase 2 is essentially eliminated. Accordingly, there is no need for temporary portable equipment to provide core cooling for a Phase 2. Rather, the direct transition into Phase 3 at the end of Phase 1 will provide adequate protection of public health and safety for core cooling.

During Phase 1, the Class 1E batteries must support reactor core and SFP cooling as well as maintaining containment capabilities. The applicant's analyses/calculations provide an estimate of the duration for which the Class 1E DC system can supply these loads. The estimate provides the maximum time period in which the transition to portable offsite equipment (Phase 3) can be achieved.

Because the station batteries were initially qualified for a two-hour duty cycle, the staff requested additional information to assess the methodology and calculations by which the applicant extended the discharge duration to 36 hours (via load shedding). Load shedding is an established industry practice to extend battery mission time, the time during which the batteries supply dc power without interruption. Given the known capacity of nuclear plant station batteries, load shedding is a strategy to extend battery runtime, the amount of time a battery can supply power before it needs to be recharged. Load shedding was explicitly identified in Section 3.2.2 of NEI 12-06, as a means to extend battery runtime.

In the “FLEX Integrated Plan,” Revision 0 (ML13128A140), the applicant indicated that the deep load shed of the dc batteries can extend battery life beyond 36 hours. However, it was not clear that the power supply needed for Phase 1 core cooling will be available for the valve operation following the deep load shed. Therefore, the staff requested additional information in RAI 01.05-11 Part (1). In its response to this RAI dated July 23, 2013 (ML13211A303), the applicant referred to Calculation U7-DCE-CALC-DESN-6001, “STP Units 3 & 4 Class 1E 125 VDC Battery Sizing Analysis.” The referenced calculation was used as input to the “Extended Station Blackout Scenario” analysis (U7-LB-140004), which provided the dc load shedding analysis. The staff audited the calculation on August 7, 2013, and August 26, 2013 (ML14129A248). Further audits were conducted on April 2, 2014, and April 24, 2014 (ML14170A685) and on June 2, 2014 (ML14182A495), to review the battery sizing calculation and confirm the adequacy of the power supply needed for Phase 1. The applicant stated that it followed the battery sizing methodology described in Appendix A of IEEE Std 485, “IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications,” based on number of positive plates.

During the April 2014, audits, the staff determined that the values shown in the worksheets did not identify the parameters associated with battery sizing as specified in the examples in Annex A of IEEE Std 485. RG 1.212, “Sizing of Large lead-Acid Storage Batteries,” endorses IEEE Std 485, which is the industry standard that provides the methodology for sizing batteries for safety-related and nonsafety-related applications in nuclear power plants. Therefore, the staff determined that the response to RAI 01.05-11, Part (1) was not acceptable. Accordingly, in RAI 01.05-33, the staff requested the applicant to either provide the necessary parameters in a revised worksheet for the “Extended Station Blackout Scenario,” or use the format of the applicable example provided in Annex A of IEEE Std 485. In its response to RAI 01.05-33 (ML14161A390), dated June 5, 2014, the applicant provided an example worksheet, outlining the parameters used in battery sizing, and in addition, provided a summary of the sizing methodology, which was to be added in FSAR Section 1E.4, “DC Electrical Equipment Loading Considerations.” The staff confirmed that the applicant developed the battery sizing in accordance with IEEE Std 485, and properly identified the parameters used in the battery sizing calculation. The battery sizing reflects the loads that the Class 1E batteries are relied upon to supply for the extended station blackout scenario, and shows they are sized per the methodology in IEEE Std 485. Therefore, the response to this portion of RAI 01.05-33 is acceptable. The staff confirmed that the applicant incorporated the proposed revision into FSAR, Appendix 1E, Section 1E.2.4, Revision 11. Therefore, this issue in RAI 01.05-33 is resolved and closed.

In RAI 01.05-11 Part (3), RAI 01.05-22, and RAI 01.05-31, the staff requested the applicant to list the power sources and manual operations upon which the applicant relies in the event of an ELAP to power all valves and pumps for all phases of the mitigation strategies including core cooling, containment overpressure protection, and SFP cooling. In the “FLEX Integrated Plan,” Revision 2, Attachment 3, dated June 19, 2014 (ML14175A141), the applicant provided a list of the valves and pumps upon which it relies for Phases 1 and 3. The staff finds the list acceptable because the list addressed the staff’s concern of identifying the power supplies for all power-operated pumps and valves, and the response to this portion of RAI 01.05-11 Part (3), RAI 01.05-22, and RAI 01.05-31, are acceptable. The staff continued to evaluate the capability of the power sources relied upon in Phase 1 and 3 to supply the loads identified by the applicant.

In RAI 01.05-33, the staff also asked the applicant to provide the battery duty cycle diagram that depicted the dc load profile and the battery division(s) providing power to the corresponding loads along the timeline for the mitigating strategies to maintain core cooling, containment, and SFP cooling during all modes of operation. In its response to RAI 01.05-33, dated June 5, 2014 (ML14161A390), the applicant provided a duty cycle diagram depicting the dc load profile for each of the battery divisions and their corresponding loads along the timeline for mitigating strategies. The diagram provides information on the loads and respective timing, including RCIC valve operation. Furthermore, the applicant provided the instrumentation and controls available at the remote shutdown panel for monitoring parameters listed in NEI 12-06, Section 3.2.1.10, "Instrumentation and Controls." The staff finds this response acceptable, as the applicant has shown the battery divisions can provide power to the corresponding loads necessary to maintain core cooling, containment, and SFP cooling capabilities over the duration for which they are relied upon. Therefore, RAI 01.05-11 Part (3), RAI 01.05-22, and RAI 01.05-31, are resolved and closed.

Even though the staff accepted the information in the response to RAI 01.05-22, there were some inconsistencies between this RAI response and the information reviewed during the audit of the battery Calculation Report U7-DCE-E-CALC-DESN-6001 regarding the dc power supply for RCIC valves (F004, F011, and F037) during Phase 1. In addition, two valves (F001 and F006) listed in the response to RAI 01.05-22, were not included in the calculation of the power the batteries would need to supply during the event. Therefore, the staff issued RAI 01.05-30 requesting the applicant to clarify these inconsistencies. In its response to RAI 01.05-30, dated March 19, 2014 (ML14084A369), the applicant explained that the inconsistency resulted from the difference in the nomenclature of the valve numbers in these two documents. The applicant has initiated Condition Report CR-14-2498 to correct the valve nomenclature and numbering to be consistent with the STP, Units 3 and 4, plant and instrumentation diagram (P&ID) numbers and nomenclature in the next revision of the calculation, which is discussed below. The applicant also indicated that the two valves in question F001 (supply from the CST) and F006 (supply from the SP), were not included in the calculation because these two valves can be operated manually and they are only needed once in Phase 1 and will have no impact on the analysis. The applicant added that the "FLEX Integrated Plan" will be revised to indicate that the initial automatic shift of the RCIC suction to the SP will take place automatically before the load shed and the subsequent shift back to the CST will be performed manually by hand wheels, which is discussed below.

Based on the Attachment 3 of the "FLEX Integrated Plan," Revision 2, the staff determined that each of the valves needed for mitigation strategies, with the exception of the solenoid SRVs, can be manually operated, and that RCIC can be manually operated. The staff finds the response to RAI 01.05-30, acceptable because the applicant clarified the inconsistency and revised the documents accordingly. Therefore, RAI 01.05-30 is resolved and closed. The "FLEX Integrated Plan," Revision 2, dated June 19, 2014 (ML14175A141), was revised to indicate the initial shift of the RCIC suction to the SP will take place automatically before the load shed and the supplemental response to RAI 01.05-30, dated July 21, 2014 (ML14210A052), confirmed Condition Report CR-14-2498 was completed. Therefore, RAI 01.05-30 is resolved and closed.

In RAI 01.05-33, the staff also requested the applicant to provide the basis for the minimum battery voltage to ensure proper operation of all electrical equipment as included in the load profile. In the June 5, 2014, response (ML14161A390), the applicant provided the maximum voltage drops to ensure the minimum battery voltage is met for the Division I and II batteries. In addition, the staff conducted audits on April 2, 2014, and April 24, 2014 (ML14170A685), and on June 2, 2014 (ML14182A495), to review Calculation U7-DCE-E-CALC-DESN-6001, “STP Units 3 &4 Class 1E 125 VDC Battery Sizing Analysis, Voltage Drop, Short Circuit, and Charger Sizing Calculation,” Revision A, to confirm the battery sizing, capacity and capability are valid under the ELAP scheme. The staff finds the basis for the assumed minimum battery voltage acceptable, as confirmed in the calculation. Because the minimum battery voltage is met for the Division I and II batteries and confirmed during the audit of the calculation, the staff also determined that the batteries can provide power to the equipment, as depicted in the load profile. Therefore, this portion of RAI 01.05-33, is resolved and closed.

In RAI 01.05-11 Part (2), the staff requested the applicant to discuss the temperature in the battery room following an ELAP relative to the qualification temperature for the batteries. In its response to RAI 01.05-11, dated July 23, 2013 (ML13211A303), stated the following:

> The STP 3&4 ABWR FLEX Integrated Plan requires that the Class 1E 125V DC power, including the batteries, function as assumed during the 36-hour battery discharge during Phase 1 of the FLEX response and are capable of being recharged and restored to service during Phase 3 of the FLEX response.
>
> Class 1E 125V DC power, including the batteries, are required to satisfy the requirements of COLA Part 2, Tier 2, Section 3.11, “Environmental Qualification of Safety-Related Mechanical and Electrical Equipment,” which provides assurance that the batteries will function as required during and after a design basis event or during and after the 8 hour Station Blackout (SBO) described in COLA Part 2, Tier 2, Table 1C-1, “ABWR Design Compliance with 10CFR 50.63 Regulations.” NINA concluded that Class 1E 125V DC power, including the batteries, will function as assumed in the FLEX plan because the thermal performance (i.e., temperature increase) of the battery during the 36 hour discharge during Phase 1 of the FLEX response is enveloped by the thermal performance of the battery during both the 2-hour battery discharge assumed during a design basis event and the 8-hour battery discharge assumed during the SBO event.

The staff had additional questions regarding the environmental conditions, specifically, the temperature in the battery rooms. Specifically, NEI 12-06 in Section 3.2.1.12, “Qualification of Installed Equipment,” states “equipment relied upon to support FLEX implementation does not need to be qualified to all extreme environments that may be posed, but some basis should be provided for the capability of the equipment to continue to function.” Therefore the response to Part (2) of RAI 01.05-11 was not acceptable and the issue was addressed in another RAI. In RAI 01.05-33, the staff asked the applicant to provide a discussion on the list of equipment serviced by each Class 1E battery division, and its operability under the expected environmental conditions. In its response to RAI 01.05-33, dated June 5, 2014, (ML14161A390), the applicant stated that safety-related instrumentation inside the reactor building will be qualified for temperatures above the estimated temperatures for the reactor building spaces during the FLEX scenario. In addition, the applicant provided information regarding the solenoid valves, which will not experience radiation levels as high as those assumed for the loss of coolant accident. Furthermore, the peak temperature inside containment does not exceed the drywell design temperature. As a result, the applicant has shown that the equipment can operate under the expected environmental conditions.

In RAI 01.05-33, the staff asked the applicant to provide a license condition that ensures that the final calculation for the "Extended Blackout Scenario" reflects the characteristics of the equipment actually installed in each plant prior to fuel load. The current version of the calculations was drafted based on available information; however, the level of detail provided in the calculations is constrained because detailed design for STP, Units 3 and 4, is not yet finalized. The staff therefore requested a license condition to ensure that upon receipt of final procurement information for the pertinent equipment, the applicant would prepare a final calculation based on the as-built equipment characteristics. In its response to RAI 01.05-33, dated June 5, 2014, the applicant provided a license condition stating that, "The 'Extended Station Blackout Scenario' calculation will be updated to incorporate 'as-built' plant design information to verify that the Class 1E battery discharge duration is adequate to support Phase 1 of the mitigating strategies discussed in FSAR Appendix 1E." The staff has modified this license condition, as set forth in Section 22.2.5 of this SER, to more specifically identify the calculation to be verified, and important battery parameters covered. As modified, the license condition will enable the staff to verify the adequacy of the "Extended Station Blackout Scenario" calculation and the supporting documentation, including, but not limited to, vendor information. Specifically, the license condition will require the licensee: 1) to verify that the installed Class IE batteries are capable of the duty cycle length relied upon in the "FLEX Integrated Plan," to the extent such relied-upon duty cycle length is greater than the maximum design-basis duty cycle length for which the battery is qualified, and 2) to complete an integrated system validation of the ELAP timeline in accordance with guidance in NUREG-0711, "Human Factors Engineering Program Review Model," Section 11.4.3, "Integrated System Validation." The staff finds the applicant's response acceptable because the license condition ensures the calculation will be finalized to ensure the Class 1E battery can support Phase 1 of the mitigating strategies. Therefore, the response to this portion of RAI 01.04-33, is resolved and closed.

With respect to the validity of the qualification of the batteries relied on to respond to an ELAP, the current regulatory guidance on battery duty cycles for safety-related batteries limits qualification to eight hours. IEEE Standard 535-1986, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations," as endorsed by RG 1.158, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants," provides guidance for qualifying nuclear-grade batteries and describes a method acceptable to the staff for complying with Commission regulations with regard to qualification of safety-related lead storage batteries for nuclear power plants. Based on a previous concern with extended battery duty cycle durations, the staff requested the applicant to interpret IEEE Standard 535-1986. The staff specifically requested the applicant to identify the length of the duty cycle for which a vented lead-acid battery is qualified per IEEE Standard 535 and to identify any limitations on the length of the duty cycle for a vented lead-acid battery. In its response to the NRC's interpretation request (ML13094A397), the applicant stated that in order to meet IEEE Standard 535, applications with duty cycles over eight hours will need to demonstrate that the battery cells fully comply with the qualification principles in Clause 5 and meet the basis in Clause 8.2 of IEEE Standard 535. Based on the background considerations stated above and the fact that the applicant is proposing to rely on the use of the Class 1E batteries for mitigation strategies, the staff was concerned about the capability of the STP, Units 3 and 4, batteries to provide dc power for the durations specified in the "FLEX Integrated Plan," Revision 1. The applicant, however, did not provide sufficient information to support a conclusion that batteries with duty cycles greater than eight hours can meet the ELAP battery duty cycles as relied upon in the mitigation strategies.

In RAI 01.05-34, the staff requested the applicant to explain how STP, Units 3 and 4, will validate battery duty cycles greater than eight hours, and justify that the methodology used is consistent with applicable regulatory guidance regarding determination of battery duty cycles. In its response to RAI 01.05-34, dated June 5, 2014 (ML14161A390), the applicant included the discharge rates provided by the battery manufacturer to show that the batteries are able to support a duty cycle greater than eight hours. The staff finds the applicant's response acceptable as a means to support this preliminary calculation because manufacturers have tested and obtained data for battery discharge rates up to 72 hours, which indicate that batteries can support a duty cycle greater than eight hours. Because the applicant has not yet procured any particular battery, however, vendor information to validate duty cycles greater than the maximum duty cycle length for which the battery is qualified will be verified before fuel load as part of the "Extended Station Blackout Scenario" calculation. This is part of the license condition described above, and RAI 01.05-34 is resolved and closed.

The "FLEX Integrated Plan," Revision 1, dated April 16, 2014 (ML14114A194), in its "Discussion of Maintaining Safety Function Support in Phase 1," for Communications states:

> Each unit will have 4 portable satellite phones and additional batteries. Also each unit will have additional batteries and a charger for radio communications. The radio communication system will have a backup supply from the 250V DC non-Class 1E battery located in the Control Building (FSAR Appendix 1E). This will be sufficient to keep the radio communication system operable for more than 36 hours.

NEI 12-06, Section 3.2.1.3, states that for initial conditions:

> Other equipment, such as portable ac power sources, portable back up dc power supplies, spare batteries, and equipment for [10 CFR] 50.54(hh)(2), may be used provided it is reasonably protected from the applicable external hazards per Sections 5 through 9 and Section 11.3 of this guidance and has predetermined hookup strategies with appropriate procedures/guidance and the equipment is stored in a relative close vicinity of the site.

As such, the 250 VDC non-Class 1E battery located in the control building is assumed to be unavailable because it is not protected and is not a safety-related battery system, per the guidance in NEI 12-06. However, NEI 12-06 states that for baseline coping capability, installed equipment that is designed to be robust with respect to design-basis external events is assumed to be fully available. Because the 250 VDC non-Class 1E battery is relied upon, the battery should be protected from the impact of earthquakes; flooding; severe storms with high winds; snow, ice, and extreme cold; and high temperature events. However, in the "FLEX Integrated Plan," Revision 2, dated June 19, 2014, the applicant removed the statement that a 250 VDC non-Class 1E battery will be used as a backup to communication equipment. The applicant stated that the batteries for the portable satellite phones will be charged with a portable diesel generator, which, as stated in FSAR Section 1.E.2.4, will be adequately protected from external events. The staff considered this to be acceptable because the applicant's approach conformed to the guidance in NEI 12-06, which recommends the use of portable satellite phones during this phase and has provided a means to charge the satellite phones that does not rely on installed equipment.

Based on the additional information and audits performed on April 2, 2014, and April 24, 2014 (ML14170A685), and on June 2, 2014 (ML14182A495), the staff confirmed that the applicant combined license application addresses the specific provisions related to baseline coping capability utilizing dc power in JLD-ISG-2012-01; confirmed that the battery sizing, qualification, capacity, and capability are still valid insofar as the mitigation strategies rely on the batteries in the event of an ELAP; and verified that the Class 1E batteries, when utilized per their proposed procedures for dealing with a beyond-design-basis external event that includes ELAP, can provide the necessary dc power to support the Phase 1 of such event, which is the first 36 hours. Therefore, the staff finds that the applicant acceptably addressed the specific provisions related to baseline coping capability utilizing dc power in JLD-SG-2012-01, and conforms to the guidance in NEI 12-06.

For Phase 3, the applicant states in “FLEX Integrated Plan,” Revision 2, that two FLEX 480V, 1500 kilowatt (kW) DGs from offsite would be connected and started to provide ac power for battery charger operation, limited ventilation system operation, and other uses. In RAI 01.05-36, the staff requested the applicant to provide additional information describing the STP, Units 3 and 4, considerations of electrical isolation for portable equipment used to provide temporary power to the battery chargers or to locally energize other components during an ELAP event. In addition, the staff requested additional information regarding how the portable generators, and the CTGs are isolated to prevent simultaneously supplying power to the same Class 1E bus in order to conform to NEI 12-06, Section 3.2.2, guidelines (13), which states, “The use of portable equipment to charge batteries or locally energized equipment may be needed under ELAP/LUHS conditions. Appropriate electrical isolations and interactions should be addressed in procedures/guidance.” The applicant stated that these DGs will be connected to pre-run cabling, for connection to the Class 1E load centers, as discussed in FSAR Appendix 1E. The staff has verified that the DGs available from offsite have sufficient capacities to power the loads on which the applicant relies to mitigate a beyond-design-basis external event. In its response to RAI 01.05-36, dated June 5, 2014 (ML14161A390), the applicant stated that portable FLEX power supplies will be connected to the 480 VAC switchgear via disconnect switches and breakers. In addition, the applicant stated that isolation is accomplished by procedures that direct the opening of the circuit breaker between the transformer and the bus on the 480 VAC switchgear to isolate the FLEX power supply from the normal power supply path. The applicant further stated that FSAR Section 1.E.2.4 would be revised to include that “the input circuit breaker from the 4160/480 VAC transformer to the applicable 480 VAC power centers will be opened to isolate the FLEX DGs from the rest of the safety-related distribution system.” The staff finds that electrical isolation between the safety-related power system and FLEX power supplies is accomplished via circuit breaker and is addressed in procedures. The staff finds the electrical isolation acceptable and conforms to the guidance in NEI 12-06. The staff concludes that the DGs conform to the guidance in NEI 12-06 and in JLD-ISG-2012-01, and is therefore acceptable. The staff confirmed that Revision 11 of FSAR Section 1.E.2.4 was revised to include that the input circuit breaker from the 4160/480 VAC transformer to the applicable 480 VAC power centers will be opened to isolate the FLEX DGs from the rest of the safety-related distribution system. Therefore, this issue in RAI 01.05-36 is resolved and closed.

22.2.4.2.5 Water and Fuel Supplies

As described in the “FLEX Integrated Plan,” in Phase 1, the RCIC takes suction from the CST or SP. The “FLEX Integrated Plan” further describes the RCIC and ACIWA water and fuel supplies as follows: The onsite CST has a capacity of more than 550,000 gallons (2,080 m^3) of water, which is sufficient for the 36-hour Phase 1 period. The RCIC steam-driven pump and the ACIWA diesel-driven fire pump, which has a minimum 36 hours of fuel supply (storage tank capacity), have sufficient pumping capacity to provide core cooling for this period. The RCIC and ACIWA are installed equipment. In addition to being seismically qualified, RCIC is inside the reactor building, a Seismic Category I structure, and the ACIWA pump is inside the fire pump house, a robust structure that is designed to withstand the applicable site-specific severe natural phenomena, including an SSE. The CST is outside the reactor building, but is designed and constructed to withstand the site-specific SSE, flood, high wind, and other site-specific external hazards. The applicant's FLEX analysis assumes a CST volume of 250,000 gallons (963 m^3), which is at the low end of the normal operating range.

The “FLEX Integrated Plan” indicates further that after 36 hours, Phase 3 begins with core cooling provided by the installed and seismically qualified ACIWA system using water in the FWSTs. There is one ACIWA system and two FWSTs shared between both units. As indicated in FSAR Appendix IE, the primary means for supplying water is by use of the diesel-driven pump in the Fire Protection System (FPS). A backup to this pump is provided by a connection on the outside of the reactor building at grade level, which allows hookup of the ACIWA to a fire truck pump or a trailer mounted portable pump. The trailer mounted portable diesel-driven pump is stored in a Seismic Category I structure. In addition, one of the two diesel driven pumps to be procured in accordance with FLEX guidance will be stored in a Seismic Category I structure. Each FWST contains a minimum usable volume of 300,000 gallons (1,135.5 m^3) (see FSAR Subsection 9.5.1.3.5). Each FWST is designed and constructed to withstand the site-specific SSE, flood, high wind, and other site-specific severe weather hazards (see FSAR Appendix 1E, Section 1E.2.4). Once the water in the FWSTs is depleted, operator action will be necessary to shift the ACIWA suction to the volume of water in the UHS basin(s) which is an engineered cell, as described below, and has a water volume of approximately 16 million gallons (60,560 m^3). The UHS (see DCD Tier 2, Section 9.2.5, “Ultimate Heat Sink”) and any pumps, fans, valves, structures, or other components that remove heat from safety systems are designed to Seismic Category I. The UHS basin can be filled as needed via a restored well water system or tanker truck. A permanent piping connection to allow the ACIWA system to take suction from the water volume in the UHS basins will be installed.

In RAI 01.05-32, the staff requested the applicant to clarify that the piping connection from the UHS to the connection of the ACIWA system is designed so that it can survive a beyond-design-basis external event to perform its intended function of a long-term water supply. In its response to RAI 01.05-32, dated March 19, 2014 (ML14084A370), the applicant clarified that the permanent piping to allow the ACIWA system to take suction from the water volumes in the UHS basins will be seismically designed consistent with the design requirements of the ACIWA system (see DCD Tier 2, Section 19I.4, “System Analysis”). The applicant stated further that this piping will be robust, sub-surface, and protected from site hazards. The response proposes to revise FSAR Appendix 1E, Section 1E.2.4 to reflect this clarification. The STP, Units 3 and 4, UHS is a Seismic Category I structure (see FSAR Section 9.2.5, “Ultimate Heat Sink”), with an enclosed concrete flood-protected basin and UHS pumps and valves located in a subsurface protected structure. Therefore, the staff determined that the proposed FSAR clarification adequately addresses the staff’s concern. The staff confirmed that the applicant incorporated the proposed revision into FSAR Appendix 1E, Section 1E.2.4, Revision 11. Therefore, RAI 01.05-32 is resolved and closed

The order specifies that a loss of normal access to the UHS be considered. The applicant’s approach is that the UHS will remain structurally functional and available after the external event. The piping connections from the UHS to ACIWA and the ACIWA pump are addressed above. The newly installed valves to transfer ACIWA suction to the UHS are addressed in Attachment 3 of the “FLEX Integrated Plan.” Because the piping system connecting the UHS to the ACIWA and the associated pumps and valves are robust or are protected by a Seismic Category I structure, the staff finds that using water from the UHS for Phase 3 is acceptable.

In regard to makeup flow, the staff reviewed the "FLEX Integrated Plan" and verified the following. The ACIWA design flow capacity is 634 gpm (40 L/s) at a backpressure of 90 psig (0.62 MPaG) and this flow capacity increases to 951 gpm (60 L/s) at a backpressure of 0 psig (0 MPaG). After 36 hours from the onset of the event, the RPV pressure is predicted to fall below 90 psig (0.62 MPaG) due to the open COPS rupture disk, which is predicted to open at 20 hours; and the injection flow capacity will be greater than 634 gpm (40 L/s). After 36 hours, the minimum flow rate needed to remove core decay heat is predicted to be approximately 165 gpm (10.4 L/s) for each unit. Therefore, the staff determined that a single ACIWA pump can provide enough flow to maintain the vessel level for both units and still have more than 150 gpm (9.46 L/s) available for the SFP makeup to each pool (or 300 gpm [18.92 L/s] total for both pools). Accordingly, the staff concludes that flow and pressure provided by the ACIWA pump are sufficient to provide makeup for both SFPs.

The staff reviewed ABWR DCD Tier 2, Subsection 5.4.7.1.1.10, "AC-Independent Water Addition (ACIWA) Mode," the STP FSAR Section 1E.2.4; and the "FLEX Integrated Plan" and determined the ACIWA system is permanently installed in the fire pump house, which is a seismically qualified robust structure. The "FLEX Integrated Plan" states that the RHR system provides the piping and valves that connect the ACIWA piping with the RHR Loop C pump discharge piping. The "FLEX Integrated Plan" indicates that the primary means for supplying water through this connection is by use of the diesel-driven pump in the ACIWA system. According to ABWR DCD Tier 2, Subsection 5.4.7.1.1.10, manual valves permit adding water from the FPS to the RHR system. The staff also determined that the ACIWA pump is designed with a minimum 36-hour fuel supply, and the connections for the manually operated ACIWA valves, which are used to introduce flow into RHR Loop C, are located in the reactor building and are readily accessible. A backup to the pump is provided by a connection on the outside of the reactor building at grade level, which allows hookup of the ACIWA to a fire truck pump.

The fuel supply for ACIWA is described in the "FLEX Integrated Plan." In Phase 3, core cooling is shifted from RCIC to ACIWA. Operators will need to transfer diesel fuel, as necessary, from one of the three underground EDG fuel oil storage tanks, which are protected by Seismic Category I structures (FSAR Subsection 3.8.6.4, "Identification of Seismic Category I Structures," and Table 3.2-1, "Classification Summary") to the ACIWA fuel storage tank using a staged portable pump and a small portable DG.

In the "FLEX Integrated Plan" for safety functions support with the "ABWR Portable Equipment Phase 3," the applicant states the following:

> A 120V portable AC generator will be stored in two of the Emergency Diesel Generator Fuel Oil Storage Vaults along with a pump that will pump fuel oil from the ESF DG Fuel Oil Storage Tanks to the ACIWA DG fuel oil tank in the fire pump house. It will also be capable of filling the portable ACIWA pump tank as well as other diesel fueled equipment. Approximately 300 feet of 1" hose will be staged to support this strategy. Five-gallon fill cans will be used to fuel the 120V portable AC generators. They will be filled at either the chemical sampling line of one of the ESF DGs or using the small fuel oil transfer pump, whichever is easiest. (FSAR Appendix 1E)

The DG fuel oil storage vaults are Seismic Category I structures, as described in FSAR Subsection 3.8.6.4 and Table 3.2-1. Based on the above information, the staff finds that the applicant has demonstrated sufficient capability regarding water and fuel supplies and pumping mechanisms for Phase 1 and Phase 3 mitigation strategies. In addition, manual operations, procedures, and guidance are addressed in Subsections 22.2.4.5 and 22.4.5 of this SER.

22.2.4.2.6 Ventilation

The staff reviewed the "FLEX Integrated Plan" Revision 0, submitted on May 2, 2013 (ML13128A140), to address conformance to agency guidance on loss of ventilation. The guidance in NEI 12-06 states that the effects of a loss of heating, ventilating, and air conditioning in an ELAP can be addressed consistent with the guidance in the Nuclear Management and Resources Council (NUMARC) report NUMARC 87-00 Revision 1, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," or by plant-specific thermal hydraulic calculations.

With regard to ventilation, the "FLEX Integrated Plan" describes the overall strategy using the RCIC, ACIWA, and COPS to provide the necessary core cooling, containment, and SFP cooling capabilities as follows. Under an ELAP, the main control room command and control will be transferred to the RSS room in the reactor building. Because the RSS room temperature will begin to increase under ELAP conditions, the Phase 1 FLEX strategy is to block open (keep in open position) the RSS room door and the stairwell doors in the reactor building to limit the temperature rise in the room until normal ventilation is restored in Phase 3. When power is restored to the plant computer, command and control will be restored to the main control room and the RSS room will no longer need to be occupied. During Phase 3, the control room temperature rise is expected to be slow, and additional measures could be implemented such as the placement of portable fans. As described in DCD Tier 2, Table 19E.2-2, "ABWR Plant Ability to Cope with Station Blackout for up to 8 Hours," the RCIC room is designed not to exceed 150.8 °F (66 °C) for at least eight hours of continuous RCIC pump operation. During the first 11 hours of the FLEX strategy using RCIC, operators may make periodic RCIC pump room entries to verify proper pump operation. But for the purpose of mitigation strategies for beyond-design-basis external events, continuous habitability will not be necessary. The door to the hallway from the RCIC pump room and its stairwell doors will be blocked open. In addition, the roof access hatch directly above the RCIC pump will be removed using the installed monorail and manually operated chain hoist. This will provide additional natural circulation cooling for the RCIC room.

With respect to the RCIC system, the staff noted that ABWR DCD Tier 2, Subsection 5.4.6.1 states the following:

> The RCIC system is designed to perform its vessel water inventory control function without AC power for at least 2 hours. Supporting systems such as DC power and the RCIC water supply are designed to support the RCIC system during this time period. Without AC power, RCIC room cooling will not be available. However, room temperature during the 2 hour period will not reach the maximum temperature for which the RCIC equipment has been qualified.

Furthermore, the staff noted that in STP FSAR Subsection 5.4.15.2.1, "Analysis to Demonstrate the Facility has 8 Hour Non-Design SBO Capability," the applicant commits to the following:

> The capability of the RCIC System to operate for 8 hours as discussed in Subsection 5.4.6 and NUREG-1503 will be demonstrated during the Initial Test Program as described in section 14.2.12.1.9. A best estimate analysis will be available for NRC review by the end of preoperational testing demonstrating that the RCIC system can function for 8 hours in an SBO event. This analysis will reflect Class 1E loadings based on expected plant and operator response during this event. Additionally, an evaluation of room temperature response during the transient will ensure that equipment remains within its qualification envelope. Similar evaluations have been satisfactorily performed on other ABWRs. (COM 5.4-1)

Because the RCIC pump room heat-up analysis only covers eight hours into the onset of an SBO, the staff was concerned that the RCIC room would continue to heat up after eight hours and potentially challenge the operability of the RCIC pump for the duration of Phase 1 (i.e., 36 hours).

In addition, with respect to the RSS rooms and the control room, the "FLEX Integrated Plan" describes the high-temperature coping strategy during Phase 1 for the RSS room (page 41), and notes that during Phase 3, command and control can be re-established in the main control room (page 27). The staff needed clarifications of the technical basis supporting these statements, and whether there is an analysis addressing habitability of the RSS rooms during Phase 1 and the control room during Phase 3. The staff was concerned about the heat-up of those spaces and the ability of the operators to perform the functions relied upon in the "FLEX Integrated Plan."

To address these issues, the staff issued RAI 01.05-29. In its response to RAI 01.05-29, dated March 19, 2014 (ML14084A368), the applicant clarified these items with proposed license conditions to address them, and which are discussed below.

In response to the staff's concern about the operability of the RCIC pump, the applicant stated that Tier 1 Departure STD DEP T1 2.4-3 was taken from the certified design to install a steam-driven mono-block pump that can be operated manually without ac or dc power. The applicant also proposed a license condition to perform an analysis confirming the following:

> [t]he RCIC room temperature will not exceed the maximum temperature at which the equipment can meet the FLEX Integrated Plan requirements using the environmental qualification of the equipment as the acceptance criteria (Reference DCD Chapter 3, Appendix 3I).

Because the substance of this proposed license condition requires confirmation that RCIC, as installed, will be capable of performing its function in the conditions in the RCIC room during a beyond-design-basis external event, the substance of the condition is acceptable to the staff. However, as set forth in Section 22.2.5 of this SER, the staff has rewritten the proposed license condition in a format appropriate for inclusion in a license, and the staff may further reformat the proposed license condition for inclusion in any license that may be issued.

In response to the staff's concern about RSS room and main control room habitability, the applicant proposed a license condition to perform a habitability analysis of the RSS room heat-up during Phase 1 and the control room heat-up during Phase 3 under an ELAP event, with the acceptance criteria in Table D-2, "Stay Times for Different WBGTs by Combinations of Clothing and Metabolism (from EPRI NP-4453)," of NUREG/CR–6146, "Local Control Stations: Human

Engineering Issues and Insights." The staff finds this proposed license condition acceptable because it requires the licensee to verify that operators will be able to occupy the RSS room and the main control room when the "FLEX Integrated Plan" relies upon operator action in those locations.

Because NUREG/CR–6146, Table D-2 is used as the acceptance criteria, the results of the heat-up analyses will also be used to determine operator stay time. The staff also finds this acceptable, because NUREG/CR–6146, Table D-2 provides guidance for determining operator stay time. However, the staff has also included a provision to govern staffing based on operator stay time in this license condition. Specifically, as set forth in Section 22.2.5 of this SER, the license condition will require the number of operators needed to staff the RSS or main control room at any time, based on operator stay time, shall be considered in the "FLEX Integrated Plan." The heat-up analysis, implementing procedures, and training plans are subject to staff inspection.

In summary, the staff reviewed the "FLEX Integrated Plan" together with the proposed license conditions. For the reasons set forth above, the staff concluded that there is reasonable assurance that the COL application meets the underlying purpose of NRC Order EA-12-049 with respect to the ventilation functions if the capabilities outlined in the application are implemented as described. Therefore, RAI 01.05-29 is resolved and closed. The NRC will impose the license condition as described in Subsection 22.2.4.5 of this SER.

22.2.4.2.7 Instrumentation and Emergency Lighting

In the "FLEX Integrated Plan," the applicant indicates that Class 1E 125 VDC power (see FSAR Subsection 8.3.2.1.3) is available for remote shutdown instrumentation for more than 72 hours following deep load shedding and implementation of a division cross-connection strategy. The "FLEX Integration Plan" describes the power available for instrumentation and lighting. The battery-supported lighting (reactor and control buildings) necessary for the ELAP operation will last at least 72 hours based on deep load shedding of the battery systems. This was confirmed by the staff in audits on April 2, 2014, and April 24, 2014 (ML14170A685), and on June 2, 2014 (ML14182A495), on the Calculation U7-DCE-CALC-DESN-6001, "STP Units 3 & 4 Class 1E 125 VDC Battery Sizing Analysis." Battery chargers will be reenergized and will operate at about 36 hours into the event, so sufficient battery margin will be available. Lighting (e.g., battle lanterns) located inside the power block will provide additional lighting. The staff's safety evaluation of the dc power design is in Subsection 22.2.4.2.4 of this SER, and the staff finds the design acceptable. Therefore, no further review is necessary.

22.2.4.2.8 FLEX Equipment and Offsite Resources

Table 3, "ABWR Portable Equipment Phase 3 Offsite FLEX Equipment," of the "FLEX Integrated Plan" provides a list of Phase 3 offsite FLEX equipment for the ABWR. This list includes two 480 V, 1500 kW portable diesel generators, six portable ventilation fans, and 300 standard cubic feet per minute (scfm) (about 510 m^3 per hour) instrument air compressors. The "FLEX Integrated Plan" describes the other Phase 3 equipment as follows. The two portable offsite DGs will provide ac power for the battery charger operation, limited ventilation system operation, and other limited uses relied on in the "FLEX Integrated Plan." Despite the fact that no portable Phase 2 equipment is needed or relied upon in the "FLEX Integrated Plan," STP, Units 3 and 4, does have portable equipment (including diesel-powered pumps, power supplies, hoses and fittings, and portable DGs) that provide additional defense-in-depth for mitigating a simultaneous ELAP and LUHS. This additional equipment is described in FSAR Section 1E.2.4. NEI 12-06 guidance states the regional response center will be able to provide the equipment listed above to a particular location within 24 hours of the time of the request, and in the “FLEX Integrated Plan,” the applicant assumed an additional six hours to relocate the equipment to the site. The “FLEX Integrated Plan” indicates the equipment and resources needed for Phase 3 will begin arriving onsite no later than 32 hours from the start of the event. The duration of 32 hours is conservative when comparing anticipated delivery of the equipment to the beginning of Phase 3.

NEI 12-06 guidance in Section F.3.2, “Qualification of Installed Equipment,” states that equipment that is stored far enough from the site such that it would not be subjected to the hazard that affected the site need not be designed or qualified for any of the assumed hazards. In addition, the storage arrangements (building, etc.) need not be designed to withstand any severe hazard from natural phenomena. The staff determined that the Phase 3 offsite FLEX equipment satisfied this condition in the guidance, because it is located in Phoenix, Arizona, or in Memphis, Tennessee.

FSAR Section 1E.2.4 describes the following three staged ac-independent portable pumping systems at STP, Units 3 and 4, that can be used as a backup to the fire water pump in the ACIWA system:

– Two pumps (a fire truck and a trailer mounted portable pump) shared between STP, Units 3 and 4, provide core, SFP, and containment cooling water to the RHR system via the ACIWA system.

– The fire truck is stored in the Turbine Building Truck Bay and is protected from site hazards with the exception of floods.

– The trailer mounted portable diesel-driven pump is stored in a Seismic Category I structure as required for protection from severe weather events (FSAR Subsection19.4.6). In addition, one of the two diesel driven pumps to be procured in accordance with FLEX guidance will be stored in a Seismic Category I structure.

  One trailer mounted pump shared among between STP, Units 1, 2, 3, and 4, provides water in the event of the loss of large areas of the plant (FSAR Part 11, Subsection 5.1.2).

– This trailer mounted pump is protected primarily by distance.

  In addition to the above pumps, two additional portable high capacity pumps will be procured as described in the paragraph below. The applicant indicated further that this will result in one high capacity portable trailer mounted diesel driven pump stored in a Seismic Category I structure in each unit, two fire trucks stored in a turbine building, and one trailer mounted diesel driven pump shared between the four units.

  Additional equipment to be procured to provide defense in depth mitigation capability, includes:

    Two diesel driven high capacity pumps (one/unit) one of which will be required to be kept in a Seismic Category I structure.

Six portable diesel generators (three/unit).

Four portable DC power supplies (two/unit).

Eight handheld satellite phones (four/unit).

Various hoses, fittings, cables, and jumpers necessary to connect the above equipment.

NEI 12-06, Section 3.2.2, "Minimum Baseline Capabilities," states that in order to assure reliability and availability of the FLEX equipment relied upon to meet these capabilities, the site should have sufficient equipment to address all functions at all units on-site, plus one additional spare, i.e., an N+1 capability, where "N" is the number of units on-site, assuming each set is a full capacity set. The staff reviewed the above list of Phase 3 offsite equipment and confirmed that the listed equipment adequately addresses the need identified in the previous sections of the SER for Phase 3 mitigation and provides N+1 capability in Phase 3 because the applicant does not rely on on-site portable equipment for mitigation strategies. The staff also confirmed that installed systems being relied on for mitigation strategies are located within their respective Seismic Category I structures and are independent of each other but are connected according to the system configuration needed for mitigation strategies (e.g., piping). Furthermore, the staff determined that equipment the applicant listed as on-site but not necessary for mitigating the event are all stored in Seismic Category I structures or will be protected from an external event.

For the reasons set forth above, the staff finds the information in the Phase 3 offsite equipment list adequate to address the staff's questions in the RAIs identified in the previous sections of this SER for Phase 3 mitigation strategies. The additional equipment will provide additional defense-in-depth and enhance the capability for these strategies. The 32-hour time frame for offsite equipment and resources to be available onsite is acceptable because the offsite equipment arrives before the Phase 1 duration of 36 hours during which the applicant has shown the onsite equipment acceptable.

22.2.4.3 *Protection of Equipment*

22.2.4.3.1 Structure

NRC Order EA-12-049 specifies that equipment used for mitigation strategies must be protected from external events. Guidance in NEI 12-06 was accepted in JLD-ISG-2012-01 provides criteria for the storage and protection of equipment used for FLEX mitigation strategies for applicable site-specific external events.

Phase 1 of the FLEX mitigation strategies relies on installed plant equipment. In NEI 12-06, Section 3.2, addresses a baseline assumption that installed plant equipment associated with FLEX strategies is fully available following a design-basis external event. Specifically, NEI 12-06 Subsection 3.2.1.3, Item 6 addresses an initial condition that permanent plant equipment is available and contained in structures with a robust design against seismic events, floods, high winds, and associated missiles.

FSAR Section 1E.2.4, states that the "FLEX Integrated Plan" relies on the installed RCIC, ACIWA, and COPS to provide core cooling, containment, and SFP cooling capabilities. Also, Class 1E batteries stored in the battery rooms provide electric power during Phase 1 of the "FLEX Integrated Plan."

On page 22 of the “FLEX Integrated Plan,” the applicant states that RCIC, COPS, and ACIWA are seismically qualified and are contained within robust structures that provide adequate protection against the applicable extreme hazards for the site.

The staff verified that the RCIC, COPS, and ACIWA valves connecting to the RHR Loop C are contained in the reactor building and the battery rooms in the control building. Both buildings are Seismic Category I structures designed to withstand the effects of design-basis external hazards described in Chapter 3 of the ABWR DCD Tier 2, including the effects of hurricane winds and associated missiles. Therefore, the staff concludes that these systems and components are protected from external hazards by structures sufficient to withstand those external hazards.

Phase 2 of the mitigation strategies involves transitioning from installed plant equipment to onsite portable equipment. NEI-12-06, Section 11.3, states that a technical basis should be developed for equipment storage for portable equipment that performs a FLEX mitigation strategy for the core cooling, containment, and SFP cooling capabilities; and the support equipment will be reasonably protected from applicable external events.

FSAR Section 1E.2.4 indicates that the “FLEX Integrated Plan” provides guidance and strategies to restore core cooling, containment, and SFP cooling capabilities following a beyond-design-basis external event. The guidance is based on a two-phase approach, where Phase 1 uses installed equipment for 36 hours until there is a direct transition into Phase 3 where offsite equipment and resources are obtained to sustain the mitigation capabilities. Because of the extended availability of Phase 1 installed equipment, the “FLEX Integrated Plan” states that there is no need for a Phase 2 that relies on onsite portable equipment. The staff reviewed and accepted this approach as documented in Subsection 22.2.4.2 of this SER.

Despite the fact that Phase 2 onsite portable equipment is not relied upon in the “FLEX Integrated Plan,” STP, Units 3 and 4, will have onsite portable equipment, including portable DGs and diesel-powered pumps and associated hoses and fittings, as described in FSAR Section 1E.2.4, which provides additional defense-in-depth for mitigating a simultaneous ELAP and LUHS. This onsite portable equipment is independent from the installed plant equipment relied upon for mitigation strategies and is stored either in Seismic Category I structures or structures that are designed to withstand the effects of applicable external events.

Phase 3 of the mitigation strategies involves obtaining offsite equipment and resources as well as using available onsite facilities to sustain long-term mitigation capabilities. The “Flex Integrated Plan” relies on the UHS basin water inventory to sustain long-term core cooling, containment and SFP cooling capabilities.

As explained above, the STP, Units 3 and 4, UHS is a Seismic Category 1 structure reviewed in Chapter 3 of this SER and is designed to withstand the effects of design-basis external hazards including the SSE and floods, tornados, hurricanes, and associated missiles.

In conclusion, for the reasons set forth above, the staff finds that the installed plant equipment used for Phase 1 and Phase 3 of the “FLEX Integrated Plan” will be protected from the applicable site-specific external hazards, consistent with the provisions of Order EA-12-049.

22.2.4.3.2 Mechanical Equipment Capability and Programmatic Controls

The NRC regulations in 10 CFR Part 50, Appendix A, GDC 1, "Quality standards and records," require that the "[s]tructures, systems, and components important to safety … be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed." In Order EA-12-049, the Commission ordered licensees and COL holders to establish a three-phase approach for mitigating beyond-design-basis external events.

Section 6.2, "Equipment Quality," of JLD-ISG-2012-01 accepts NEI 12-06 with additional provisions for the quality of equipment used to respond to beyond-design-basis external events. In Subsection 3.2.1.12, "Qualification of Installed Equipment," NEI 12-06 states that equipment relied upon to support the FLEX implementation does not need to be qualified to all extreme environments that may be posed, but some basis should be provided for the capability of the equipment to continue to function. In Section 11.2, "Equipment Design," NEI 12-06 states in Item 1 that design requirements and supporting analyses should be developed for portable equipment that directly performs a FLEX mitigation strategy for the core cooling, containment, and SFP cooling capabilities. This information provides the inputs, assumptions, and documented analyses to show that the mitigation strategy and support equipment will perform as intended. Footnote 3 in NEI 12-06 states that the FLEX documentation should be auditable, but the 10 CFR Part 50, Appendix B QA provisions are not required, and that manufacturer information may be used to establish the basis for the equipment use. In Section 11.2 of NEI 12-06, Item 3 notes that the functionality of the equipment may be outside of the manufacturer's specifications if justified in a documented engineering evaluation. In Section 11.5, "Maintenance and Testing," NEI 12-06 indicates in Item 1 that FLEX mitigation equipment should be initially tested; or other reasonable means used to verify that equipment performance conforms to the limiting functions on which the FLEX strategies and guidelines rely. Item (1) notes that validation of source manufacturer quality is not required. Section 11.5 in NEI 12-06, also states that portable equipment that directly performs a FLEX mitigation strategy for the core cooling, containment, or SFP cooling capabilities should be subject to maintenance and testing (including surveillances and inspections) to verify proper function.

FSAR Appendix 1E, Section 1E.2.4 indicates that the "FLEX Integrated Plan" provides guidance and strategies to restore core cooling, containment, and SFP cooling capabilities following a beyond-design-basis external event. Section 1E.2.4 states that the guidance employs a two-phase approach, where Phase 1 uses installed equipment for 36 hours until there is a transition to Phase 3 with offsite resources. Because of the extended availability of Phase 1 equipment and supplies, the "FLEX Integrated Plan" states that the Phase 2 interval specified in Order EA-12-049, is not needed.

In RAI 01.05-24, the staff requested the applicant to describe the performance of the following categories of equipment used as part of the mitigation strategies for an ELAP event:

a. Existing and new safety-related pumps, valves, and dynamic restraints.

b. Existing and new nonsafety-related pumps, valves, and dynamic restraints.

c. Any portable equipment such as pumps and valves.

In its response to RAI 01.05-24, dated March 19, 2014 (ML14084A371), the applicant provided clarifying information to each of the aspects of RAI 01.05-24. The applicant's response to each aspect of RAI 01.05-24, and the staff's evaluation of each response are discussed below:

(a) The applicant clarified that all safety-related pumps, valves, and dynamic restraints used as part of the mitigation strategies for an ELAP event are permanently installed equipment. The applicant indicated that the design bases for these components, including the extent to which they are credited to perform, are discussed in the ABWR DCD Tier 2 and FSAR Chapters 3, 5 (Section 5.4.6), and 6 and in FSAR Appendix 1E. The applicant indicated further that the components will not be relied upon to perform functions beyond those credited in the design basis during the implementation of the strategies discussed in FSAR Appendix 1E and the "FLEX Integrated Plan," except with respect to a specific aspect of the RCIC system. In particular, the applicant stated that the suction temperature for the RCIC system while connected to the suppression pool will be allowed to increase to 250 °F (121.1 °C), which is the qualification temperature of the limiting component part of the RCIC system (specifically the pump bearings). The applicant evaluated the RCIC net positive suction head (NPSH) for the FLEX operation in accordance with design basis assumptions based on SP pressure, temperature, and level generated during the beyond-design-basis external event. To address piping expansion for the increased RCIC suction temperature, the applicant stated that FSAR Appendix 1E, Section 1E.2.4 would be revised to specify that the RCIC piping expansion calculations will be performed at 250 °F (121.1 °C). The planned revision of the FSAR would specify the provisions for RCIC piping expansion calculations of 250 °F (121.1 °C).

The staff finds that the application of installed safety-related pumps, valves, and dynamic restraints within the limits of their design-basis credited performance specified in the COL FSAR (with one indicated exception) is acceptable, because the staff has already evaluated the capability provisions for ABWR safety-related components as part of the rulemaking certifying the ABWR standard design or in the COL application review. Therefore, this portion of RAI 01.05-24 is resolved and closed. The staff confirmed that in Revision 11 of the FSAR, the revised RCIC piping expansion calculations was referenced in Appendix 1E, Section 1E.2.4. Therefore, this issue in RAI 01.05-24, is resolved and closed.

(b) The applicant clarified that all nonsafety-related pumps, valves, and dynamic restraints used as part of the mitigation strategies for an ELAP are permanently installed plant equipment. The applicant indicated that these components will not be relied upon to perform functions beyond those discussed in the ABWR DCD Tier 2 and FSAR Chapters 3, 5, 6, and 9 (Section 9.5.1) during the implementation of the strategies discussed in FSAR Section 1E.2.4, and the "FLEX Integrated Plan." The applicant indicated that the water flow rate for the ACIWA cooling mode was selected to optimize the containment pressurization after the onset of core damage. The applicant references ABWR DCD Tier 2 Subsection 5.4.7.1.1.10.3, "ACIWA Flow Rate," which specifies that the flow rate supplied by the ACIWA mode of the RHR system using either the diesel-driven pump or the fire pump truck is between 634 gpm (40 L/s) and 951 gpm (60 L/s), for conditions between no containment backpressure and a backpressure equal to the COPS setpoint. The applicant stated that the ACIWA flow rate relied upon for a single unit is 230 gpm (14.5 L/s) at 11 hours and 165 gpm (10.4 L/s) at 36 hours. The applicant indicates that the single ACIWA diesel-driven pump is capable of supplying the necessary flow rate to maintain the vessel level for both units throughout the ACIWA portion of the makeup.

The staff finds this portion of the response acceptable because it clarifies that all nonsafety-related pumps, valves, and dynamic restraints utilized as part of the ELAP mitigation strategies will perform consistent with the functions specified for them in the DCD and the FSAR. Therefore, this portion of RAI 01.05-24, is resolved and closed.

(c) The applicant clarified that the only portable pumps that will be relied upon in STP, Units 3 and 4, as part of the mitigation strategies for an ELAP event are two portable pumps used to transfer fuel oil to the ACIWA fuel oil storage tank from the EDG fuel oil storage tanks. The applicant indicates that this will involve placing into service a vane-type positive displacement pump with at least a 100-foot (30.5-meter) head and a 5 gpm (0.315 L/s) flow rate. The applicant notes that additional pumps discussed in FSAR Appendix 1E, Section 1E.2.4 are "defense-in-depth" equipment and are not relied upon in the "FLEX Integrated Plan."

The staff finds the applicant's clarification of the description of the portable pumps used to transfer fuel oil to the ACIWA fuel oil storage tank and their use to be acceptable. The performance capability of these pumps is discussed in response to RAI 01.05-25, later in this SER section. Therefore, this portion of RAI 01.05-24, is resolved and closed.

In RAI 01.05-25, the staff requested the applicant to describe the provisions for the design, manufacture, testing, installation, and surveillance to provide assurance of the seismic, environmental, and functional capability of existing and new safety-related pumps, valves, and dynamic restraints to perform their intended functions as part of the mitigation strategies for an ELAP event at STP, Units 3 and 4. The staff also requested the applicant to indicate whether any safety-related pumps, valves, and dynamic restraints used as part of the mitigation strategies for an ELAP event will be relied upon to perform in a manner that differs from their original design and performance specifications (such as differences in pump NPSH available).

In its responses to RAI 01.05-25, dated March 19, 2014, and May 21, 2014 (ML14084A371 and ML14148A410, respectively), the applicant indicated that all safety-related pumps, valves, and dynamic restraints utilized as part of the mitigation strategies for an ELAP are permanently installed plant equipment. The applicant stated that the design, manufacture, testing, installation, and surveillance used to provide assurance of the seismic, environmental, and functional capability requirements for these components are discussed in the ABWR DCD Tier 2 and in STP FSAR Chapters 3, 5, and 6 and in Appendix 1E. The applicant clarified that these SSCs are not relied upon to perform FLEX functions that differ from their original design and performance specifications, because these specifications will not be exceeded during implementation of the strategies discussed in FSAR Appendix 1E, Section 1E.2.4 and in the "FLEX Integrated Plan." The applicant indicated that the NPSH for the RCIC system with the increase in the suction temperature had been evaluated and found acceptable based on the containment overpressure generated during the event. The applicant noted that the reduced NPSH for the RCIC system will be resolved when the suction is shifted to the CST. The applicant also noted that the testing of this safety-related equipment will be conducted in accordance with regulatory requirements such as Technical Specifications and the Maintenance Rule Program.

The staff reviewed the DCD and FSAR chapters and sections referenced in the applicant's response to RAI 01.05-25. For example, ABWR DCD Tier 2, Section 3.9, "Mechanical Systems and Components," supplemented by STP, Units 3 and 4, FSAR Tier 2, Section 3.9 of the same title, specifies provisions for the functional design and qualification and inservice testing of safety-related pumps, valves, and dynamic restraints in an ABWR nuclear power plant. The staff documented its review of provisions for functional design and qualification, and in service testing, of safety-related pumps, valves, and dynamic restraints in the SER on the ABWR DC application (NUREG–1503). The evaluation of additional information provided by the applicant is included in Section 3.9 of this SER.

During the telephone conferences on May 6, 2014 (ML14190A327) and May 13, 2014 (ML14154A015), the staff indicated to the applicant that the initial response to RAI 01.05-25 dated March 19, 2014, did not provide a sufficient basis for assumptions regarding containment overpressure in demonstrating the adequate NPSH (available) for the RCIC system during an ELAP event at STP, Units 3 and 4. The applicant stated that the revised response to RAI 01.05-25, would provide additional justification for the capability of the RCIC system to respond to an ELAP event (including the basis for its containment overpressure assumptions), and to specify planned modifications to the "FLEX Integrated Plan" to describe the operation of the RCIC system to avoid potential NPSH concerns. On May 21, 2014, the applicant submitted a revised response to RAI 01.05-25, that replaced the initial response in its entirety to address the staff's comments.

In its revised response to RAI 01-05-25, dated May 21, 2014 (ML14148A410), the applicant clarified that after the initial automatic start and suction shift from the CST to the SP, the RCIC system will be operated manually during the postulated ELAP event. The RCIC system will operate within the design and perform as described in the ABWR DCD and STP, Units 3 and 4, FSAR, except that while connected to the suppression pool, the suction temperature will be allowed to increase to 250 °F (121.1 °C), which is the qualification temperature for the limiting RCIC system component (pump bearings). The applicant described an evaluation of the NPSH for the RCIC system in the submittal dated May 21, 2014. In particular, the applicant summarized the evaluation of the NPSH (available) and NPSH (required) during the time that the RCIC suction would be aligned to the SP. (In this discussion, "NPSH (required)" refers to the NPSH that the RCIC pumps need to function properly, and not any regulatory requirement, and "NPSH (available)" refers to the calculated NPSH based on the postulated conditions at the pump inlet.) The evaluation determined that the minimum NPSH (available) occurs at eight hours into the ELAP event, at which time the containment pressure is calculated to be 27.2 psia (0.187 MPa) and the SP water temperature is calculated to be 227 °F (108.3 °C). The applicant stated that when the SP water temperature nears 250 °F (121.1 °C) at about 10 hours into the ELAP event, STP, Units 3 and 4, plant operators will switch the RCIC pump suction to the CST.

In response to the NRC questions regarding the use of containment pressure in the NPSH evaluation for the RCIC system, the applicant clarified that the determination of the NPSH margin is conservative for several reasons. First, NPSH (required) is calculated based on full RCIC flow, even though full flow will not be needed to remove the decay heat. Second, the friction head loss resulting from suction pipe friction and the suction strainer head will be less than the assumed value because the assumed head loss was not adjusted due to lower water viscosity at a higher temperature and a reduced RCIC flow. Third, the RCIC pump supplier has indicated that the NPSH (required) for the STP, Units 3 and 4, RCIC pump will be lower than the assumed value of 23 feet (7 meters). With these conservatisms, the applicant provides support for the assumption that cavitation of the RCIC pump will not occur. Nevertheless, the applicant stated that the STP, Units 3 and 4, plant operators will monitor the RPV level at the RSS panel and the RCIC pump operation locally in the RCIC room. If there are indications of RCIC pump cavitation, the plant operators will switch the RCIC suction to the CST. The applicant indicated further that if the RCIC system becomes unavailable for any reason, the plant operators can take steps to shift the reactor vessel injection to the ACIWA system.

In the submission dated May 21, 2014, the applicant stated that the “FLEX Integrated Plan” would be revised to specify that containment overpressure will be credited to maintain the RCIC NPSH during an ELAP event. In the revised plan, if the RCIC becomes unavailable before the planned end of its use, STP, Units 3 and 4, plant operators will reduce the reactor pressure and shift to injection using the ACIWA system. Furthermore, in the response to the staff’s questions dated May 21, 2014 (ML14148A410), the applicant clarified that it had evaluated the recommendations in BWROG Project Task Report DRF 0000-0155-1541, “RCIC Pump and Turbine Durability Evaluation-Pinch Point Study,” and found that they were not applicable to the integrated turbine-pump monoblock design to be used by STP, Units 3 and 4. The staff finds the applicant’s evaluation of RCIC pump NPSH to be reasonable in that RCIC full flow is not needed to remove decay heat for mitigation strategies, and water viscosity effects were neglected. Accordingly, the staff finds the applicant’s RCIC NPSH analysis acceptable.

For the reasons discussed above, the staff finds that the applicant has provided sufficient provisions to demonstrate the seismic, environmental, and functional capability of existing and new safety-related pumps, valves, and dynamic restraints to perform their intended functions as part of the mitigation strategies for an ELAP event at STP, Units 3 and 4; including the planned modifications to the “FLEX Integrated Plan.” The staff verified that Revision 2 to the “FLEX Integrated Plan” specifies that containment overpressure is credited to maintain the RCIC NPSH during an ELAP event. Revision 2 of the “FLEX Integrated Plan” also specifies that if the RCIC system becomes unavailable before the planned end of its use, STP, Units 3 and 4, plant operators will reduce the reactor pressure and shift the injection using the ACIWA system. Therefore, this issue in RAI 01.05-25, is resolved and closed.

In RAI 01.05-26, the staff requested the applicant to describe the provisions for design, manufacture, testing, installation, and surveillance to provide assurance of the seismic, environmental, and functional capability of existing and new nonsafety-related pumps, valves, and dynamic restraints to perform the FLEX functions upon which the mitigation strategies rely for an ELAP event at STP, Units 3 and 4.

In its response to RAI 01.05-26, dated March 19, 2014 (ML14084A371), the applicant clarified that all nonsafety-related pumps, valves, and dynamic restraints used as part of the mitigation strategies for an ELAP event are permanently installed plant equipment and part of the ACIWA system that is part of the FPS. The design, manufacture, testing, installation, and surveillance requirements for these components are discussed in the ABWR DCD Tier 2 and FSAR Chapters 3 and 9 (Section 9.5.1); and in FSAR Appendix 1E. The operation of these components during normal and emergency conditions is described in DCD Tier 2, Subsection 5.4.7.1.1.10. The applicant noted that testing this nonsafety-related equipment is performed in accordance with the Fire Protection Program, and the Maintenance Rule Program. The ACIWA system and the FPS are included in the Design-Reliability Assurance Program (D-RAP) and are within the scope of the Maintenance Rule.

The staff reviewed the DCD and FSAR chapters and sections referenced in the applicant’s response to RAI 01.05-26. The “FLEX Integrated Plan” referenced in FSAR Appendix 1E specifies that the RCIC system will be used for the first 36 hours during an ELAP event to provide core cooling. Subsection 5.4.6.1, “Design Basis,” of the ABWR DCD Tier 2 indicates that the RCIC system is a safety system designed to assure that sufficient water inventory is maintained in the reactor vessel to permit adequate core cooling. Subsection 5.4.6.1.2.1, "Reliability and Operability," of the ABWR DCD Tier 2 specifies that the RCIC system is designed commensurate with the safety importance of the system and its equipment. The DCD states that the system is tested during both the startup and preoperational phases of the plant, with functional and operability testing performed at predetermined intervals throughout the life of the plant. Table 3.2-1, "Classification Summary," in the ABWR DCD Tier 2 provides the classification provisions for the RCIC system, including quality assurance requirements in accordance with 10 CFR Part 50, Appendix B, for the most significant RCIC components. Furthermore, STP, Units 3 and 4, FSAR Table 3.9S-1, "Inservice Testing Program Plan (OM-2004)," specifies periodic testing provisions for the RCIC pump and valves. Because the ABWR DCD and STP Units 3 and 4, FSAR provisions are acceptable to ensure that RCIC will function as designed when called upon to do so, the staff determined the functional design, qualification, and inservice testing (IST) provisions for the RCIC system to be acceptable in regard to the use of RCIC during the first 36 hours of the response to an ELAP event at STP, Units 3 and 4.

The "FLEX Integrated Plan" referenced in FSAR Appendix 1E specifies that the ACIWA system will be used after approximately 36 hours to provide core cooling and to refill the SFP as needed. Subsection 5.4.7.1.1.10, "AC-Independent Water Addition (ACIWA) Mode," of the ABWR DCD Tier 2 specifies that the ACIWA system provides a means of introducing water from the FPS directly into the reactor pressure vessel, into the drywell spray header, or into the wetwell spray header under degraded plant conditions when ac power is not available from either onsite or offsite sources. The "FLEX Integrated Plan" states that the RHR system provides the piping and valves that connect the ACIWA piping with the RHR Loop C pump discharge piping. The "FLEX Integrated Plan" indicates that the primary means for supplying water through this connection is by use of the diesel-driven pump in the ACIWA system. According to ABWR DCD Tier 2, Subsection 5.4.7.1.1.10, manual valves permit adding water from the FPS to the RHR system.

Subsection 9.5.1.6.2, "Quality Assurance (QA) Program," of the ABWR DCD Tier 2 specifies that QA policies and procedures for the FPS are in Chapter 17, "Quality Assurance," of the ABWR DCD Tier 2 for compliance with the criteria in 10 CFR Part 50, Appendix B. Subsection 9.5.1.3.5, "General Description of Fire Protection System," of the ABWR DCD Tier 2 specifies that the fire protection sprinkler systems and other portions of the FPS are designed in compliance with ANSI Standard B31.1, "Power Piping," and are analyzed to remain functional following an SSE. The "FLEX Integrated Plan" in the general overview section states that the diesel-driven pump is designed to withstand a design-basis earthquake. In discussing installed equipment, the "FLEX Integrated Plan" states that the RCIC system, COPS, and ACIWA system are seismically qualified and are contained within robust structures so they are adequately protected against the applicable site's extreme hazards. Furthermore, Subsection 9.5.1.5, "Inspection and Testing Requirements," of the ABWR DCD Tier 2 states that preoperational inspection and testing requirements will be prepared for each FPS as described in ABWR DCD Tier 2, Section 14.2.12, "Individual Test Descriptions." In particular, Subsection 14.2.12.1.48, "Fire Protection System Preoperational Test," specifies preoperational tests for individual components of the FPS, including proper performance of all remote-operated valves; the diesel engine-driven fire pump; and various components of water-based suppression systems such as the automatic wet pipe sprinkler system.

Section 17.4S, "Reliability Assurance Program," of the STP, Units 3 and 4, FSAR specifies provisions for the post-certification D-RAP and the transition to reliability assurance activities during operations. For example, Section 17.4S.2, "Procurement, Fabrication, Construction, and

Test Specifications," specifies that provisions for procurement, fabrication, construction, and test specifications for safety-related and nonsafety-related SSCs within the scope of the RAP are prepared and implemented under the approved "STP 3 & 4 Quality Assurance Program Description" referenced in Section 17.5S, "Quality Assurance Program Guidance." Section 17.4S.2 states that Section 2, "Nonsafety-Related SSCs Credited for Regulatory Events," in Part III, "Nonsafety-Related SSC Quality Control," of the "STP 3 & 4 Quality Assurance Program Description" specifies the quality requirements for nonsafety-related SSCs credited in mitigating defined events such as the FPS. When SSCs are risk-significant because of their role in mitigating defined events, Section 17.4S.2 states that the specified quality requirements for these SSCs will be satisfied. Section 1, "Nonsafety-Related SSCs–Significant Contributors to Plant Safety," in Part III of the "STP 3 & 4 Quality Assurance Program Description" describes specific program controls to be applied to nonsafety-related SSCs to which 10 CFR Part 50 Appendix B is not applicable but where the SSCs are significant contributors to plant safety. Section 1 in Part III in the "STP 3 & 4 Quality Assurance Program Description" states that the specific program controls that are consistent with the applicable sections of the "STP 3 & 4 Quality Assurance Program Description" are applied to those items in a selected manner, by targeting those characteristics or critical attributes that render the SSC a significant contributor to plant safety. Section 2 in Part III in the "STP 3 & 4 Quality Assurance Program Description" specifies that the guidance regarding quality in RG 1.189 (Revision 2), "Fire Protection for Operating Nuclear Power Plants," shall be implemented for the Fire Protection System.

The "FLEX Integrated Plan" specifies that the applicant will implement an administrative program whereby the equipment used in these strategies will be controlled with respect to configuration control, maintenance, and testing. The plan indicates that preventative maintenance (PM) activities and inventories will be established for components, on which the strategies rely, and testing procedures will be developed and frequencies established based on the type of equipment and considerations in EPRI guidelines. The plan states that these programs, maintenance practices, and procedures, which are part of the STP, Units 3 and 4, operational programs, will be in place 180 days before the initial STP, Unit 3, fuel load. The NRC plans to inspect the STP, Units 3 and 4, operational programs before the fuel load. As discussed later in this SER section, the applicant proposed a license condition to specify an administrative program addressing equipment and procedures used as part of the mitigation strategies at STP, Units 3 and 4, for an ELAP event. The staff determined that the ABWR DCD and STP, Units 3 and 4, FSAR provisions for the ACIWA system, with verification required by the license condition specified later in this SER section, provide acceptable support for use of this system as part of the mitigation strategies for an ELAP event beyond 36 hours. In particular, the applicant has provided support for the performance capability of the FLEX equipment as specified in the ABWR DCD and the applicant's FSAR. In addition, the applicant has described testing and maintenance activities for FLEX equipment to provide confidence in the readiness of this equipment to perform its FLEX functions, and the staff verified these activities are sufficient to provide that confidence. Therefore, this issue in RAI 01.05-26 is resolved and closed.

In RAI 01.05-27, the staff requested the applicant to describe the provisions for design, manufacture, testing, installation, and surveillance that provide assurance of the seismic, environmental, and functional capability of portable pumps, valves, and dynamic restraints that are relied on to accomplish the mitigation strategies for an ELAP event at STP, Units 3 and 4.

In its response to RAI 01.05-27, dated March 19, 2014 (ML14084A371), the applicant clarified that the only portable equipment used in STP, Units 3 and 4, as part of the mitigation strategies for an ELAP event are two 120V pumps used to transfer fuel oil to the ACIWA fuel tank from the EDG fuel oil storage tanks. The applicant stated that these pumps will be commercial grade components procured to perform the function of transferring fuel from an EDG fuel oil storage tank to the FPS fuel oil storage tank during Phase 3, based on the as-built design. The applicant indicated that these pumps will be procured in accordance with Sections 4 and 7 of the "STP 3 & 4 Quality Assurance Program Description." The applicant indicated that detailed procedures regarding the determination of the critical characteristics that ensure a commercial grade item is suitable for its intended use, a technical evaluation of the item, receipt standards, and a quality evaluation of the item will be developed during implementation of Operational Programs as discussed in FSAR Section 13.4S, "Operational Program Implementation," and stipulated in Section 7, "Control of Purchased Material, Equipment, and Services"; and in Section 7.2, "NQA-1-1994 Commitment," of "STP 3 & 4 Quality Assurance Program Description."

The staff reviewed the FSAR chapters and sections referenced in the applicant's response to RAI 01.05-27. For example, Section 7.2 in "STP 3 & 4 Quality Assurance Program Description" states:

> For commercial grade items, special quality verification requirements are established and described in STPNOC documents to provide the necessary assurance an item will perform satisfactorily in service. The STPNOC documents address determining the critical characteristics that ensure an item is suitable for its intended use, technical evaluation of the item, receipt requirements, and quality evaluation of the item.

The staff determined that the provisions specified for procurement of the portable pumps to be used as part of the mitigation strategies at STP, Units 3 and 4, in response to an ELAP event, as well as the planned maintenance and testing activities, are adequate because they address the equipment characteristics necessary to ensure the equipment can function as relied upon in the mitigation strategies. Accordingly, the staff finds the use of these pumps for this event to be acceptable. Further, the commercial grade dedication process has been determined to provide reasonable assurance of the capability of SSCs to perform safety-related functions at operating nuclear power plants. Therefore, the staff determined that the application of these commercial grade provisions specified in Section 7.2 of the QA Program Description was acceptable in demonstrating the capability of the portable pumps to be used as part of the mitigation strategies at STP, Units 3 and 4, in response to an ELAP event. With respect to portable equipment, the "FLEX Integrated Plan" references NEI 12-06, which states in Section 11.5 that portable equipment that directly performs a FLEX mitigation strategy for the core cooling, containment, or SFP cooling capabilities should be subject to maintenance and testing (including surveillance and inspections) to verify their proper function. The "FLEX Integrated Plan" states that STP, Units 3 and 4, will implement an administrative program, where the equipment used in these strategies will be controlled with respect to configuration control, maintenance, and testing. The applicant included this administrative program in its proposed license condition discussed later in this SER section. Therefore, RAI 01.05-27, is resolved and closed.

In RAI 01.05-28, the staff requested the applicant to describe the operational programs referenced in FSAR Appendix 1E and in the "FLEX Integrated Plan" that will provide assurance of the functional capability of the pumps, valves, and dynamic restraints used in the mitigation strategies for an ELAP event at STP, Units 3 and 4. FSAR Appendix 1E, Section 1E.2.4 references the operational programs in FSAR Section 13.4S. The NRC also requested the applicant to indicate whether the planned operational program for mitigating beyond-design-basis external events will be specified in FSAR Tier 2, Table 13.4S-1, “Operational Programs Required by NRC Regulation and Program Implementation,” with a milestone for implementation.

In its response to RAI 01.05-28, dated March 19, 2014 (ML14084A371), the applicant stated that specific operational programs listed in FSAR Section 13.4S will provide assurance of the functional capability of the pumps, valves, and dynamic restraints used in mitigation strategies for an ELAP event at STP, Units 3 and 4. The specific operational programs referenced by the applicant include the Fire Protection Program; Non-Licensed Plant Staff Training Program; Reactor Operator Training Program; Reactor Operator Requalification Program; Quality Assurance Program – Operation, Maintenance Rule, Motor-Operated Valve Testing Program; and the Initial Test Program. As a result, the applicant determined that an additional operational program for pumps, valves, and dynamic restraints used in mitigation strategies for an ELAP event did not need to be developed. The staff notes that other operational programs listed in FSAR Section 13.4S (such as Preservice Testing and Inservice Testing Programs) include activities related to pumps, valves, and dynamic restraints used in mitigation strategies for an ELAP event at STP, Units 3 and 4. The license condition described in Section 22.2.5, of this SER will require integration of the provisions relating to mitigation strategies with similar items in operational programs required in the design basis. Accordingly, the staff finds the applicant’s plan to provide assurance of the functional capability of the pumps, valves, and dynamic restraints used in mitigation strategies for an ELAP event at STP, Units 3 and 4, through the operational programs listed in FSAR Section 13.4S (including those identified in the RAI response and the other programs listed in FSAR Section 13.4S) to be acceptable. Therefore, RAI 01.05-28 is resolved and closed.

22.2.4.4 *Multi-Unit Event at STP Site*

The events at Fukushima demonstrate that beyond-design-basis external events may adversely affect multiple units at a single site.

On Page 3, “Systems Credited for FLEX,” of the “FLEX Integrated Plan,” the applicant states that the ACIWA and the FWSTs are the only system or components shared between STP, Units 3 and 4. The applicant describes the SSCs shared by STP, Units 3 and 4, for mitigation strategies as follows: All other mitigation systems (e.g., RCIC and COPS) discussed in the “FLEX Integrated Plan” are unit-specific. Both units would share one ACIWA system and the two FWSTs. Each FWST contains a minimum usable volume of 300,000 gallons (1,135.5 m^3) (see FSAR Subsection 9.5.1.3.5), which is sufficient for each unit through Phase 1. After 72 hours in Phase 3, the water in the FWSTs is calculated to be depleted, and operator action will be necessary to shift the ACIWA suction to the volume of water in the UHS basin.

The NRC staff’s evaluations of RCIC and COPS as relied upon for mitigation strategies are in Subsection 22.2.4.2.1 and Subsection 22.2.4.2.2 of this SER, respectively. The staff’s evaluation of ACIWA system, the shared pump, FWSTs, and UHS is in Subsection 22.2.4.2.5 of this SER. As described in these three sections of the SER, the staff finds the capabilities of the above equipment to be acceptable.

FSAR Section 1E.2.4 addresses additional onsite and offsite portable equipment that will be available for use at either unit. Sharing portable equipment provides additional defense-in-depth. The NRC staff’s evaluation of the adequacy of this portable equipment is set forth in Subsection 22.2.4.2.8 of this SER, and the staff finds the portable equipment to be acceptable.

As described in that SER section, multi-unit considerations do not change this finding because the applicant proposed a complement of portable equipment sufficient to execute mitigation strategies for an event affecting all units on the site, plus at least one additional set of equipment, in accordance with the guidance in NEI 12-06. Accordingly, the staff concluded that sufficient equipment capability exists for a multi-unit ELAP event.

In addition, the applicant states in FSAR Section 1E.2.4 that the detailed procedures and training to support the FLEX Program will be developed during the implementation of operational programs as described in FSAR Section 13.4S, in cooperation with STP, Units 1 and 2, as a four-unit site.

The staff noted that the procedures and training are not currently in FSAR Section 13.4S, and the adequacy of the procedures and training cannot now be reviewed. Therefore, a license condition will be imposed on procedures, training, and programmatic controls to address the multi-unit concerns, as further discussed below.

22.2.4.5 *Programmatic Controls and Procedures*

Page 19 of the "FLEX Integrated Plan" indicates the following:

> STP 3&4 will implement an administrative program whereby the equipment used in these strategies will be controlled with respect to configuration control, maintenance and testing. Preventative Maintenance (PM) activities and inventories will be established for required components and testing procedures will be developed and frequencies established based on type of equipment and considerations made within EPRI guidelines. These programs, maintenance requirements, and procedures, which are part of the STP 3&4 operational programs, will be in place 180 days prior to initial fuel load on Unit 3. It is expected that an NRC inspection of STP 3&4 operational programs will occur prior to fuel load.
>
> The Systematic Approach to Training (SAT) will be used to evaluate required training for station personnel based upon plant equipment and procedures that result from implementation of the strategies described in this report.
>
> This training will be developed and completed 180 days prior to the initial fuel load of Unit 3.

NEI 12-06, Chapter 11, "Programmatic Controls," summarizes the programmatic controls to be considered for the implementation of the plant-specific FLEX strategies. Chapter11 discusses quality attributes, equipment design, equipment storage, procedure guidance, maintenance and testing, training, staffing, and configuration controls. The staff reviewed the programmatic controls for the "FLEX Integrated Plan" listed above and determined that the description of the program lacked specificity. Furthermore, to the extent Order EA-12-049 addresses programmatic controls, binding requirements are necessary to ensure that the programmatic controls employed provide adequate protection of public health. In RAI 01.05-23, the staff requested the applicant to describe the programmatic controls. In its response to RAI 01.05-23, dated November 19, 2013 (ML13326A300), the applicant included a proposed license condition. In its response to RAI 01.05-28 dated March 19, 2014 (ML14084A371), the applicant supplemented this license condition. The applicant proposed the following updated license condition:

Prior to initial fuel load, the following requirements will be addressed using the guidance contained in JLD-ISG-2012-01, Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, Revision 0 (August 29, 2012):

a. An administrative program will be developed for configuration control, maintenance and testing of the equipment used in the strategies to mitigate beyond-design-basis external events, as described in Final Safety Analysis Report (FSAR) Appendix 1E, Section 1E.2.4. This program will establish requirements for Preventative Maintenance (PM) activities and inventories and will include testing procedures and frequencies.

b. Guidance and strategies to maintain or restore core cooling, containment and spent fuel pool cooling capabilities following a beyond-design-basis external event will be developed, implemented, and maintained. The guidance will include appropriate interfaces between the various accident mitigation procedures and guidelines so that the overall strategies are coherent and comprehensive.

c. A Training Program will be developed using the Systematic Approach to Training (SAT) to evaluate required training for station personnel based upon plant equipment and procedures that result from implementation of the strategies.

d. The basic configuration of the system design requirements to support FLEX implementation as stated in FSAR, Appendix 1E, Section 1E.2.4 are installed, and the documented results of inspections confirm that the system design requirements have been implemented as discussed in FSAR, Appendix 1E, Section 1E.2.4.

e. Analyses will be performed to verify that the RCIC room habitability and equipment functionality are acceptable under extended loss of AC power conditions for completion of mitigating strategy functions during Phase 1 as discussed in FSAR Appendix 1E.

   The habitability analysis of the heat-up of the RCIC room during an ELAP will confirm that the temperature will not prevent completion of the intermittent required operator actions. The acceptance criteria used for this analysis will utilize the guidance contained in NUREG/CR-6146, “Local Control Stations: Human Engineering Issues and Insights," Table D-2.

   The equipment functionality analysis will also confirm that the RCIC room temperature will not exceed the maximum temperature at which the equipment can meet the FLEX Integrated Plan requirements using the environmental qualification of the equipment as the acceptance criteria (Reference DCD Chapter 3, Appendix 3I).

f. Analyses will be performed to verify that the Remote Shutdown System (RSS) room habitability and equipment functionality are acceptable under extended loss of AC power conditions for completion of mitigating strategy functions during Phase 1 and that the Control Room (CR) habitability and equipment functionality are acceptable during Phase 3 as discussed in FSAR Appendix 1E.

The habitability analysis of the heat-up of the RSS room in Phase 1 and the CR in Phase 3 during an ELAP will confirm that the temperature will not prevent completion of the required operator actions. The acceptance criteria used for this analysis will utilize the guidance contained in NUREG/CR-6146, “Local Control Stations: Human Engineering Issues and Insights,” Table D-2.

The equipment functionality analysis will also confirm that the RSS room temperature and CR room temperature will not exceed the maximum temperature at which the equipment can meet the FLEX Integrated Plan requirements using the environmental qualification of the equipment as the acceptance criteria (Reference DCD Chapter 3, Appendix 3I).

g. The administrative program, guidance and strategies, training program, and procedures developed for the implementation of the mitigating strategies will include provisions to respond to multi-unit events.

h. The procedures developed to implement the mitigating strategies will address implementation in all plant operating modes, including shutdown and refueling.

An overall integrated plan will be developed 180 days prior to initial fuel load, including a description of how compliance with the requirements described in this license condition will be achieved.

The staff reviewed the proposed license condition and changed it as set forth below in Section 22.2.5 of this SER. Among other things, the license condition as modified by the staff would require the applicant to complete the development of the overall integrated plan (including procedures, etc.) at least 18 months before the scheduled completion of the last Inspection, Tests, Analyses, and Acceptance Criteria (ITAAC) on the schedule required by 10 CFR 52.99(a). The staff has reformatted the license condition, although if the NRC grants the application, the format will change to conform to the organization of the COL. The staff has largely retained the substance of the proposed condition, except in regard to requirements for verification of the adequacy of installed SSCs to the extent credited in the design basis. To the extent the applicant relies on SSCs credited in the design basis for mitigation functions, existing ITAAC set forth the required inspections, tests, and analyses necessary to verify that those SSCs will function as credited and relied upon. For these installed SSCs, the license condition requires additional verification only to the extent that the applicant’s “FLEX Integrated Plan” relies on them to perform mitigation functions not credited in the design basis. The staff concludes that with the above changes, this issue is resolved. Therefore, RAI 01.05-23 is resolved and closed. The staff may inspect the documents (e.g., guidance, procedures, training program, administrative program for configuration control, maintenance, and testing) described above before the initial fuel load to confirm that the license condition is satisfied.

22.2.5 Post Combined License Activities

The staff proposes to impose a license condition requiring the applicant to develop and implement plant procedures; develop, implement and maintain guidance and strategies; install the FLEX equipment; develop a training program; and establish administrative controls for configuration control, maintenance, and testing. The license condition further requires the development of an overall plan to be completed at least 18 months before the latest date set forth in the schedule for completing the inspections, tests, and analyses in the ITAAC submitted in accordance with 10 CFR 52.99(a). However, the NRC will not determine the final form of the license condition until the NRC prepares a COL. The details of the license condition are as follows:

License Condition 22.2-1

a. The licensee shall complete development of an overall integrated plan of strategies to mitigate a beyond-design-basis external event at least one year 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.

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 ultimate heat sink and (ii) providing for adequate capacity to perform the functions upon which the guidance and strategies rely for all units on the STP 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, as described in Final Safety Analysis Report (FSAR) Appendix 1E, Section 1E.2.4.

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. Before fuel load, the licensee shall analyze:

   1. The habitability of the RCIC room, RSS room and the main control room in regard to heat-up during a loss of all ac power and loss of normal access to the UHS to confirm that the RCIC, RSS and main control room temperature will not prevent the completion of the intermittent operator actions upon which the guidance and strategies required by this condition rely, in accordance with the acceptance criteria in Table D-2 of NUREG/CR–6146, “Local Control Stations: Human Engineering Issues and Insights”; and

   2. The RCIC and RSS room temperatures in regard to heat-up during a loss of all ac power and loss of normal access to the UHS to confirm that the RCIC and RSS room temperature will not exceed the maximum temperature at which the equipment located in these rooms can perform the functions on which the guidance and strategies required by this section rely, in accordance with the environmental conditions for which the equipment is qualified as described in FSAR Chapter 3, Appendix 3I.

g. Before fuel load, the licensee shall update the design calculation for Class 1E battery discharge to reflect ‘as-built’ plant design information to verify that the Class 1E batteries function as relied upon to support Phase 1 of the mitigation guidance and strategies required by this condition, as described in FSAR Appendix 1E.

h. Before fuel load, the licensee shall complete a successful integrated system validation of the extended loss of all ac power and loss of normal access to the UHS timeline in accordance with guidance in NUREG-0711, “Human Factor Engineering Program Review Model,” Section 11.4.3, “Integrated System Validation,” to verify that operator actions will be completed consistent with the start of the battery duty cycle, as described in the FLEX Integrated Plan, Revision 2.

i. 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 of this license continue to be satisfied and the licensee documents the evaluation in an auditable form.

22.2.6 Conclusion

The NRC staff evaluated the applicant’s “FLEX Integrated Plan” and the RAI responses to the NRC staff’s requests for information addressing mitigation strategies for beyond-design-basis external events at STP, Units 3 and 4. For the reasons set forth above, the staff concludes that the ABWR design described in the ABWR DCD and supplemented by the STP COLA has the capability to restore or maintain core cooling, containment, and SFP cooling capabilities in the event of an ELAP and LUHS in all modes of operation. This conclusion is based on using the SSCs to perform these functions, as described in the ABWR DCD and in FSAR Appendix 1E, supplemented by additional information and clarifications in the RAI responses.

As part of the response capability assessment, the applicant included necessary supporting functions such as instrumentation, the ventilation of equipment, and emergency lighting.

The time frame for which supplemental equipment would be needed for the installed SSCs is sufficiently long, so that the licensee can rely on the regional response center for support. The time frame is based on load shedding of the batteries and the use of the RCIC system. In addition, the installed ACIWA system and available onsite portable resources can also perform necessary functions.

The staff further concludes that the installed equipment to be used is robust, as described above. In addition, as described in detail above, the staff finds that the onsite equipment that can be used, if needed, is reasonably protected because it is appropriately stored and designed.

Finally, the COL applicant has committed to make arrangements for obtaining offsite resources to continue the required functions after the installed capacity is exhausted.

Therefore, the staff concludes that there is reasonable assurance that the application meets the underlying purpose of NRC Order EA-12-049, if the capabilities outlined in the application are implemented as described.

22.3 Reliable Spent Fuel Pool Instrumentation (Recommendation 7.1)

22.3.1 Introduction

On March 11, 2011, a magnitude 9.0 earthquake struck off the coast of the Japanese island of Honshu. The earthquake resulted in a large tsunami that is estimated to have exceeded 14 meters (45 feet) in height, which inundated the Fukushima Dai-ichi Nuclear Power Plant site. The tsunami caused extensive damage to site facilities and resulted in a complete loss of all ac electrical power at five of the six units on the site. In responding to and managing this damage, the plant operators lacked, among other things, reliable instrumentation to determine the water level in the SFPs on the site. The absence of this information, combined with the operators' inability to visually observe the SFPs because of the conditions in the plant, raised concerns that at least one pool may have boiled dry—resulting in fuel damage—and highlighted the need for reliable SFP instrumentation.

The current SFP water level instrumentation at U.S. nuclear power plants typically has a narrow range and is therefore only capable of monitoring normal and a narrow range of 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 SFP instrumentation to provide emergency responders with reliable information on the condition of SFPs. A reliable and available indicator is essential to ensure that plant personnel can effectively prioritize emergency actions.

The Commission paper SECY-12-0025 (ML12039A111) states that for DCs and combined license applications submitted under 10 CFR Part 52 that are currently under active staff review, the staff plans to ensure that the Fukushima actions approved by the Commission are addressed before certification or licensing. The staff will request all COL applicants to provide the information required by the orders and request for information letters described in this SECY paper through the review process.

In addition, JLD-ISG-2012-03 Revision 0 (ML12221A339), "Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation," endorses with exceptions and clarifications the methodologies described in the Nuclear Energy Institute (NEI) industry guidance document NEI 12-02 (ML122400399) Revision 1, "Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," and provides an acceptable approach for satisfying the applicable requirements.

22.3.2 Summary of Application

By letter dated April 25, 2012 (ML121080046), the staff informed the applicant that the staff has been directed by the Commission to implement the Fukushima NTTF recommendations in SECY-12-0025. This request for additional information (RAI) specifically addresses Recommendation 7.1, "Reliable Spent Fuel Pool Instrumentation." The staff requested the applicant to address each of the provisions for monitoring key SFP parameters as described in the March 12, 2012, Order EA-12-051 (ML12054A679), including any proposals for changes to the current application.

The applicant responded to the staff's RAI in letters dated June 25, 2012, and December 6, 2012 (ML121850710 and ML12346A445, respectively), and March 14, 2013 (ML13079A343). As part of the RAI response, the applicant proposed to enhance the SFP level instrumentation to ensure that it provides a reliable indication of the water level in the spent fuel storage pool and that the instrumentation will indicate the SFP level as described in EA-12-051. The applicant's response describes site-specific inspections, tests, analyses, and acceptance criteria (ITAAC) to verify the proper installation of the instruments and that these instruments have all of the design functions described in the STP, Units 3 and 4, COL FSAR, Revision 11, Appendix 1E, Section 1E.2.6.

These changes are incorporated in Revision 12 of the STP COL FSAR.

22.3.3 Regulatory Basis and Guidance

The relevant requirements for reliable SFP instrumentation are established or described in the following:

- SRM-SECY-12-0025, which approves the issuance of orders for reliable SFP instrumentation under an administrative exemption to the Backfit Rule, 10 CFR 50.109, and the issue finality requirements in 10 CFR 52.63 and 10 CFR Part 52, Appendix D, Paragraph VIII.

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

The relevant guidance for reliable SFP instrumentation is set forth in JLD-ISG-2012-03, which endorses NEI 12-02 Revision 1, with exceptions and clarifications.

22.3.4 Technical Evaluation

In light of SECY-12-0025, the staff issued RAI letter 417 that requested additional information in relation to the lessons learned from Great Tohoku Earthquake and Tsunami. In RAI 01.05-2, the staff requested the applicant to:

> address each of the provisions for monitoring key spent fuel pool parameters as described in the March 12, 2012 Order, EA-12-051 (ML12054A679), including any proposals for changes to your current application.

Commission Order EA-12-051, describes the key parameters used to determine whether level instrument is considered reliable. NEI 12-02, Revision 1 provides an acceptable approach for satisfying the applicable requirements. The staff evaluated the applicant's response to RAI 01.05-2, and determined that additional information was needed. The staff issued RAI 01.05-6 and RAI 01.05-7, in which the staff requested the applicant to provide further clarification of the level instrument design criteria and related programmatic aspects. In its response to RAIs 01.05-6 and 01.05-7 dated March 14, 2013 (ML13079A343), the applicant provided additional design information, indicates that it will follow the guidance of NEI 12-02, and concludes, therefore, that the SFP level instrument is designed to be reliable. The staff's evaluation of these responses is discussed below.

Instruments:

Commission Order EA-12-051, Attachment 2, Section 1.1, states that the SFP level instrumentation shall consist of a permanent, fixed primary instrument channel and a backup instrument channel. The backup instrument channel may be fixed or portable. Portable instruments shall have capabilities that enhance the ability of trained personnel to monitor the SFP water level under conditions such as partial structural damage, high radiation levels, or high heat and humidity.

In its response to RAI 01.05-6, dated March 14, 2013, the applicant stated that the instrumentation will consist of two permanent and fixed instrument channels with the level indication from the top of the fuel racks to above the normal operating level of the SFP. Level instrumentation will include high and low water level alarms that indicate and annunciate in the control room, in addition to level indications independent of the process computer at the remote shutdown system panel or other appropriate and accessible locations. The level channels will be functional in all plant operating modes. The applicant's RAI response also provides marked-up FSAR changes.

The staff evaluated the applicant's proposed instrument description and the FSAR changes and determined that crediting two permanently installed instruments as primary and backup channels complies with the design features identified in the guidance in JLD-ISG-2012-03 and called for in Commission Order EA-12-051. The staff confirmed that the proposed FSAR changes were incorporated into Revision 9 of the FSAR. Therefore, this issue in RAI 01.05-6 is resolved and closed.

Arrangement:

Commission Order EA-12-051, Attachment 2, Section 1.1, stated that the SFP 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 SFP.

This protection may be provided by locating the safety-related instruments to maintain instrument channel separation within the SFP area, and to utilize inherent shielding from missiles provided by existing recesses and corners in the SFP structure.

In its response to RAI 01.05-6, dated March 14, 2013 (ML13079A343), the applicant stated that the SFP level instrument channels will 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 SFP. This protection will be provided by maintaining instrument channel separation within the SFP area and will utilize inherent shielding from missiles provided by existing corners in the SFP structure. The channel separation guidance in NEI 12-02, Revision 1 Section 3.2, will be considered in determining sensor locations. The applicant's response also provided marked-up FSAR changes.

The staff evaluated the applicant's proposed instrument location description and the proposed changes to FSAR Appendix 1E, Section 1E.2.6. The staff determined that the applicant's proposal to follow the channel separation guidance in NEI 12-02, Revision 1, Section 3.2, and to place the instrumentation in the corners of the SFP satisfies EA-12-051 in regard to protection of the instrumentation from missiles. The staff thus concluded that these features conform to the guidance in JLD-ISG-2012-03 and comply with Commission Order EA-12-051 in regard to missile protection. The staff confirmed that the proposed FSAR changes were incorporated into Revision 9 of the FSAR. Therefore, this issue in RAI 01.05-6 is resolved and closed.

Mounting:

Commission Order EA-12-051, Attachment 2, Section 1.3, states that the installed instrument channel equipment within the SFP shall be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the SFP structure.

In its response to RAI 01.05-6, dated March 14, 2013 (ML13079A343), the applicant stated that the seismic design of the instrument mounting will be consistent with the SFP seismic design. In addition, an evaluation of other hardware stored in the SFP will be conducted to ensure that it will not create adverse interactions with the fixed instrument locations.

The staff evaluated the RAI response and the proposed instrument mounting description in the FSAR. The staff determined that designing the instrument mounting with the same seismic classification as the SFP liner (Seismic Category I) will ensure that the mounting will retain its configuration following a seismic event. The applicant also proposes to evaluate other equipment in the vicinity of the SFP to verify that a failure of the equipment will not create adverse interactions with the instruments. Accordingly, the staff concludes that these features conform to the guidance in JLD-ISG-2012-03 and comply with Commission Order EA-12-051 in regard to the seismic design of the mountings. The staff confirmed that the proposed FSAR changes were incorporated into Revision 9 of the FSAR. Therefore, this issue in RAI 01.05-6 is resolved and closed.

Qualification:

Commission Order EA-12-051, Attachment 2, Section 1.4, states, in part, that the primary and backup instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the SFP water at saturation conditions for an extended period.

In its response to RAI 01.05-6, dated April 23, 2013 (ML13120A163), the applicant stated that the instrument channels will be reliable at temperature, humidity, and radiation levels consistent with normal operation; an accident event; and post-event conditions. In addition, the applicant states that these instrument channels will be included in the Design Reliability Assurance Program (D-RAP).

The staff reviewed the applicant's response and the proposed FSAR changes. Because the SFP instrument channels will be included in the D-RAP, the staff concludes that the instruments will be designed to remain operational during all anticipated conditions, and these capabilities will be demonstrated in accordance with the guidance in JLD-ISG-2012-03. Accordingly, the staff determined that these features comply with Commission Order EA-12-051. The staff confirmed that the proposed FSAR changes were incorporated into Revision 9 of the FSAR. Therefore, this issue in RAI 01.05-6 is resolved and closed.

Independence:

Commission Order EA-12-051, Attachment 2, Section 1.5, states that the primary instrument channel shall be independent of the backup instrument channel.

The applicant's response to RAI 01.05-6, and the proposed FSAR changes clarify that the instrument channels are physically and electronically independent.

The staff reviewed the applicant's response along with the proposed FSAR changes. The staff verified that the physical separation of the primary and backup channels will be sufficient to establish physical independence, and that the channels are not electronically connected. Accordingly, the staff concludes that this feature conforms to the guidance in JLD-ISG-2012-03 and complies with Commission Order EA-12-051 in regard to independence. The staff confirmed that the proposed FSAR changes were incorporated into Revision 9 of the FSAR. Therefore, this issue in RAI 01.05-6 is resolved and closed.

Power Sources:

Commission Order EA-12-051, Attachment 2, Section 1.6, states, in part, that permanently installed instrumentation channels shall each be powered by a separate power supply. Permanently installed and portable instrumentation channels shall provide for power connections from sources independent of the plant ac and dc power distribution systems, such as portable generators or replaceable batteries.

The applicant's response to RAI 01.05-6, indicates that the level instrument channels will be powered by Class 1E batteries. These batteries are designed to provide power for more than 76 hours post-event utilizing deep load shedding and division cross-connection strategies. Flex equipment (480 V ac portable generators) is credited to recharge the batteries and power other ESF loads

The staff reviewed the applicant's response and the proposed FSAR changes. The staff identified that the level instrument channels are powered by separate Class 1E batteries capable of powering the instruments for more than 76 hours. The applicant credits the use of a 480 V ac portable generator to recharge the batteries and to power the instrumentation 36 hours after event initiation. The staff's evaluation of the FLEX equipment capability and availability is in Section 22.2 of this SER.

The 480 V ac FLEX DG is credited with the capability to recharge the Class 1E batteries and to power other ESF loads. The applicant's response states that since the Class 1E batteries can power the instruments continuously, there is no need for a backup power supply. The staff noted that this statement is true only if the 480 V ac FLEX DG is available. The staff has determined that the 480 V ac FLEX DG is independent of the plant ac and dc power distribution systems since the FLEX DG can power the instrumentation directly. Accordingly, the design conforms with NRC Order EA 12-051 provisions for power connections to the instrument channels. Based on the evaluation of the applicant's response and the proposed FSAR changes, the staff concludes that these design features conform to the guidance in JLD-ISG-2012-03 and comply with Commission Order EA-12-051 in regard to power supplies. The staff confirmed that the proposed FSAR changes were incorporated into Revision 9 of the FSAR. Therefore, this issue in RAI 01.05-6 is resolved and closed.

Accuracy:

Commission Order EA-12-051, Attachment 2, Section 1.7, states that the instrument channels shall maintain their designed accuracy following a power interruption or change in power source without recalibration.

The applicant's response to RAI 01.05-6, stated that the instrument channels will maintain their designed accuracy following a power interruption or a change in power source without recalibration. The applicant also stated that considerations in determining required instrument accuracy should include SFP conditions (e.g., saturated water or steam conditions). Instrument accuracy will also be sufficient to allow trained personnel to determine when the actual level reaches the specified lower level of each indicating range (Levels 1, 2, and 3) without a conflicting or ambiguous indication.

The staff reviewed the applicant's response and the proposed FSAR changes. The accuracy of the instruments as installed will be verified through testing, which will be confirmed through ITAAC 3.0-28, as discussed below. Accordingly, the staff concluded that these design features comply with Commission Order EA-12-051 and the guidance in JLD-ISG-2012-03. The staff confirmed that the proposed FSAR changes were incorporated into Revision 9 of the FSAR. Therefore, this issue in RAI 01.05-6 is resolved and closed.

Testing:

Commission Order EA-12-051, Attachment 2, Section 1.8, states that the instrument channel design shall provide for routine testing and calibration.

The applicant's response to RAI 01.05-6, stated that the instrument channel design will provide for routine testing and calibration, which can be accomplished in situ.

The staff reviewed the applicant's system description and the proposed FSAR changes. The staff identified that the permanently installed instrument channels are normally used to monitor the SFP level and will be subject to routine testing and calibration in accordance with plant procedures. Accordingly, the staff concludes that these design features comply with Commission Order EA-12-051 and the guidance in JLD-ISG-2012-03. The staff confirmed that the proposed FSAR changes were incorporated into Revision 9 of the FSAR. Therefore, this issue in RAI 01.05-6 is resolved and closed.

Display:

Commission Order EA-12-051, Attachment 2, Section 1.9 states that trained personnel shall be able to monitor the SFP water level from the control room, the alternate shutdown panel, or another appropriate and accessible location. The display shall provide on-demand or continuous indication of SFP water level.

The applicant's response to RAI 01.05-6, stated that trained personnel will be able to monitor the SFP water level from the control room, the remote shutdown system panel, or other appropriate and accessible locations. The display will provide on-demand or continuous indications of the SFP water level.

The staff reviewed the applicant's system description and the proposed FSAR changes. The location of the level indication display, as installed, will be confirmed through ITAAC 3.0-28, as discussed below. Accordingly, the staff concluded that these design features comply with Commission Order EA-12-051 and the guidance in JLD-ISG-2012-03. The staff confirmed that the proposed FSAR changes were incorporated into Revision 9 of the FSAR. Therefore, this issue in RAI 01.05-6 is resolved and closed.

Programs:

Commission Order EA-12-051, Attachment 2, Section 2, states that the SFP instrumentation shall be maintained available and reliable through appropriate development and implementation of a training program, procedures, and a testing and calibration program. Personnel shall be trained in the use of the primary and backup instrument channels, provision of alternate power to each channel and testing and calibration of each channel. Procedures shall be established and maintained for the testing, calibration, and use of the primary and backup spent SFP instrument channels. Processes shall be established and maintained for scheduling and implementing testing and calibration of the primary and backup SFP level instrument channels sufficient to maintain them at the design accuracy.

In its response to RAI 01.05-7, dated April 23, 2013 (ML131120A163), the applicant stated that FSAR Appendix 1E, Section 1E.2.6, will be revised to identify that the SFP instrumentation will be maintained to be available and reliable through the appropriate development and implementation of the following programs:

- Training: Personnel will be trained to perform job-specific functions necessary for their assigned tasks (maintenance, calibration, surveillance, etc.).
- Procedures: Procedures shall be established and maintained for testing, calibrating, and using the spent fuel pool level instrument channels. These procedures will also address any known potentially abnormal response issues associated with the instrumentation.
- Testing and Calibration: Processes will be established and maintained for scheduling and implementing necessary testing and calibration of the spent fuel pool level instrument channels so as to maintain the instrument channels at the design accuracy. Additionally, the-out-of-service provisions in NEI 12-02, Revision 1 Section 4.3, will be implemented for the SFP level channels. The

spent fuel pool level instrument channels will be included in the Design Reliability Assurance Program (DRAP).

The applicant's response also reflects the consideration that the activities intended to support the development of procedures and training for the enhanced SFP instrumentation are already covered by the existing commitment in Section 13.5 of the FSAR.

The staff evaluated the applicant's RAI response and the proposed FSAR changes. The staff finds that the program descriptions in the proposed FSAR Appendix 1E, Section 1E.2.6, conform to the guidance in JLD-ISG-2012-03, which calls for the development of procedures for testing and calibration of the primary and backup SFP level instrument channel, and therefore comply with Commission Order EA-12-051. The staff notes that procedures relating to the use and installation of FLEX equipment are evaluated in Section 22.2 of this SER. The staff has also determined that the existing commitments in FSAR Section 13.5 already covered the procedures described in FSAR Appendix 1E, Section 1E.2.6. Therefore, no new commitment is needed for the development of these procedures. Accordingly, the staff concluded that the applicant's response is acceptable. The staff confirmed that the proposed FSAR changes were incorporated into Revision 9 of the FSAR. Therefore, this issue in RAI 01.05-7 is resolved and closed.

ITAAC:

The applicant's response to RAI 01.05-6, proposed to create a new site-specific ITAAC in COL application Part 9, "Inspections, Tests, Analyses, Acceptance Criteria," Section 3.0 (Site-Specific ITAAC) Table 3.0-28, "Spent Fuel Pool Level Instrumentation," (as shown below), to ensure that the SFP level instrumentation will be designed and installed as described in FSAR Appendix 1E.

Spent Fuel Pool Level Instrumentation ITAAC

Design Requirement	Inspections, Tests, Analyses	Acceptance Criteria
The spent fuel pool level instrumentation channels are properly installed, in the correct locations, and meet all design features in FSAR Appendix 1E.	Inspections will be performed to verify that the spent fuel pool level instrument channels are properly installed, in the correct locations, and meet all design features in FSAR Appendix 1E.	The results of inspections and tests confirm that the spent fuel pool level instrument channels are properly installed, in the correct locations, and meet all design features in FSAR Appendix 1E, Subsection 2.6.

The staff reviewed the proposed site-specific ITAAC. The staff finds the proposed ITAAC acceptable because the acceptance criteria will confirm that the installed level instrumentation meets the design functions specified in FSAR Appendix 1E, Section 1E.2.6. The staff confirmed that the applicant incorporated the proposed revision into Part 9 of the COL application, Revision 11. Therefore, this issue in RAI 01.05-6 is resolved and closed.

22.3.5 Post Combined License Activities

The applicant identifies the following site-specific ITAAC:

- Site-specific ITAAC Table 3.0-28 ensures that the SFP level instrumentation will be designed and installed as described in FSAR Appendix 1E.

22.3.6 Conclusion

The staff evaluated the applicant's RAI responses to the staff's request for addressing Recommendation 7.1 in SECY-12-0025. The staff specifically requested the applicant to address each of the provisions for monitoring key SFP parameters—as described in the March 12, 2012, Order EA-12-051 (ML12054A679)—including any proposals for changes to the current application. For the reasons set forth above, the staff' concludes that the applicant has provided adequate information on the SFP water level instrumentation.

As set forth above, the staff has determined that these instruments are designed in accordance with the guidance in JLD-ISG-2012-03. The staff therefore considers these instruments to be reliable, able to withstand design-basis natural phenomena, and capable of monitoring key SFP level parameters as described in Commission Order EA-12-051.

22.4 Emergency Preparedness (Recommendation 9.3)

22.4.1 Introduction

The accident at Fukushima reinforced the need for an effective Emergency Plan (EP), with the objective of ensuring that the capability exists for a licensee (or COL applicant) to implement measures that mitigate the consequences of a radiological emergency and protect the public. The accident at Fukushima highlighted the need to determine the number of positions in the emergency organization needed to respond to a multi-unit event and fill those positions as part of the emergency plan. Additionally, there is a need to ensure that the communication equipment relied upon to coordinate the event response during a prolonged station blackout has adequate power.

22.4.2 Summary of Application

In Revision 12 of the FSAR, Part 2, Appendix 1E, Section 1E.2.8, "Enhanced Emergency Plan Staffing and Communication (9.3)," the applicant discusses how STP Units 3 and 4 addresses NTTF Recommendation 9.3. This information was developed in the applicant's RAI responses.

Specifically, on April 24, 2013 (ML13119A175), the applicant provided Revision 3 to its response to RAI 01.05-3 regarding NTTF Recommendation 9.3, which covers emergency preparedness communications and staffing. In regard to Application Part 2, Appendix 1E Section 1E.2.8, "Enhanced Emergency Plan Staffing and Communication (9.3)," the applicant's response states:

> The Emergency Plan for STP 3 and 4 will be part of a site-wide emergency plan for Units 1 through 4. NEI 12-01, Revision 0 (Guidelines for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities) will be used in assessing staff and communications capabilities necessary to respond to a beyond design basis multi-unit event. The results of the assessment will be addressed in the detailed Emergency Plan procedures developed during implementation of operational programs as described in FSAR Section 13.4S and in concert with STP Units 1 and 2. COLA Part 9, Table 4.0-1,

Items 5.3, 5.4, 10.3, and 10.4 lists the ITAACs applicable to the staffing and communications capability assessments.

In Revision 4 of its RAI response, the applicant removed the proposed ITAAC because the issues that were to be covered by the ITAAC will be addressed through license conditions.

22.4.3 Regulatory Basis

The relevant requirements for the EP regarding 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, “... and 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, and...”
- Appendix E, “Emergency Planning and Preparedness for Production and Utilization Facilities,” to 10 CFR Part 50, “Domestic Licensing 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.”

The relevant guidance for the EP regarding beyond-design-basis external events are established or described in the following:

- SRM-SECY-12-0025, approves the issuance of orders for beyond-design-basis external events, as necessary, for ensuring continued adequate protection. SECY-12-0025 states, in part, that the staff will also request all COL applicants to provide information required by the orders and information letters described in this paper, as applicable, through the review process.
- NUREG–0696, “Functional Criteria for Emergency Response Facilities,” 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.
- NEI 12-01, Revision 0 guidance (ML12125A412), which the staff found to be an acceptable method for licensees to employ when responding to the 10 CFR 50.54(f) letters regarding NTTF Recommendation 9.3 (ML12131A043).
- 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.

22.4.4 Technical Evaluation

By letter dated April 25, 2012 (ML121080046), the NRC informed the applicant that the staff would be sending RAIs related to the approved Fukushima NTTF recommendations in SECY-12-0025. In RAI 01.05-3, the staff requested the applicant to address Recommendation 9.3, “provisions for enhancing emergency preparedness.” The NRC issued this information request regarding the power supplies for communications systems and staffing to determine if additional regulatory action is warranted. This request was based upon NTTF Recommendation 9.3, which proposed that facility emergency plans provide for a means to power communications equipment needed to communicate onsite and offsite during an extended loss of ac power and staffing to fill all necessary positions to respond to a multi-unit event.

On April 24, 2013, the applicant provided its Revision 3 response to RAI 01.05-3 (ML13119A175), in which it addressed NTTF Recommendation 9.3 regarding emergency preparedness communications and staffing. In addition, the applicant proposed FSAR changes and committed to include these changes in the next routine revision of the COL application. In its response to RAI 01.05-3, dated April 24, 2013, the applicant commits to perform assessments of communications and staffing using NEI 12-01, Revision 0, two years before the scheduled initial fuel loading of STP, Unit 3. By letter, from the NRC to the NEI dated May 15, 2012 (ML12131A043), the NRC determined 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. Therefore, the staff determined the applicant’s commitment to perform assessments of the staffing and communications capabilities using NEI 12-01, Revision 0, is acceptable.

In the Revision 3 response, the applicant also proposed to include four new ITAAC Items (5.3, 5.4, 10.3, and 10.4) in Table 4.0-1, “Emergency Planning-Inspection, Test, Analysis, and Acceptance Criteria (EP-ITTAC),” of Part 9 of the COL application. However, the staff does not agree that these four new ITAAC items are necessary. The staff determined that these items should more appropriately be addressed as license conditions. The staff intends to make these items part of a new license condition discussed below in Section 22.4.5 of this SER.

Based on the above evaluation, the staff finds that the response to RAI-01.05.3 is acceptable except for the proposed ITAAC, which are not necessary. The items covered by these proposed ITAAC will instead be addressed through license conditions. The portion of the response that is acceptable conforms to the guidance in SECY-12-0025, SRM-SECY-12-0025, and NEI 12-01, Revision 0 regarding communications and staffing for NTTF Recommendation 9.3, NUREG–0654/FEMA-REP-1, and NUREG–0696; and meets the applicable requirements in 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50.

The applicant submitted Revision 4 to the response to RAI 01.05-3, dated October 1, 2014 (ML14282A242), to eliminate these ITAAC items in Part 9 of the COL application. The staff confirmed that the changes were incorporated into Revision 11, Part 9 of the COL application. Therefore, this issue in RAI 01.05-3 is resolved and closed.

22.4.5 Post Combined License Activities

As discussed earlier in the technical evaluation, the staff proposed including the following license condition related to emergency planning actions regarding communications and staffing:

License Condition (22.4-1)

> Communications:
>
> At least 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 all ac power. The communications capability assessment shall be performed in accordance with NEI-12-01, Revision 0, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities.".
>
> At least 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 complete implementation of corrective actions identified in the communications capability assessment described above, including any related emergency plan and implementing procedure changes and associated training.
>
> Staffing:
>
> At least 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 assessments of the onsite and augmented staffing capability for responding to a multi-unit event. The staffing assessments will be performed in accordance with NEI 12-01, Revision 0, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities."
>
> At least 180 days before the date scheduled for initial fuel loading 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:
>
> - Incorporation of corrective actions identified in the staffing assessments required by this condition; and
> - Identification of how the augmented staff will be notified given degraded communications capabilities.

22.4.6 Conclusion

The staff evaluated the applicant's response to the NRC's request for information addressing NTTF Recommendation 9.3, as discussed in SECY-12-0025. The staff specifically requested the applicant to address each of the provisions for enhancing emergency preparedness as described in SECY-12-0025, Enclosure 7, including any proposals for changes to the current application. These provisions included NTTF Recommendation 9.3, which proposed that facility emergency plans provide for a means to power communications equipment needed to communicate onsite and offsite during an extended loss of ac power and staffing to fill all positions necessary to respond to a multi-unit event. For the reasons set forth above, the staff concludes that the applicant has provided adequate information to address emergency preparedness communications and staffing in connection with NTTF Recommendation 9.3.

Based on the above, the staff determined that the license condition described above is necessary and appropriate to assure that the application conforms to the guidance provided in SECY-12-0025 and NEI 12-01, Revision 0, regarding communications and staffing for NTTF Recommendation 9.3; NUREG–0654/FEMA-REP-1; and NUREG–0696; and accordingly meets the applicable requirements in 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50.