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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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South Texas Project
Units 3 and 4
Docket Nos. 52-012 and 52-013
Response to Request for Additional Information

Reference: Letter: Rocky D. Foster to Scott Head "Request for Additional Information Letter No. 417 Related to SRP Section 01.05 for the Nuclear Innovation North America LLC Combined License Application(ML121230021) dated May 2, 2012.

Attached are the responses to the NRC staff questions included in Request for Additional Information (RAI) letter referenced above. The attachment provides the responses to the following RAI questions. This submittal completes the response to the referenced letter.

01-05-1 01-05-2 01-05-3 01-05-4

Changes to the COLA are shown in shaded text.

There are no commitments in this letter.

If you have any questions, please contact me at (361) 972-7136 or Bill Mookhoek at (361) 972-7274.

DO91
HRO

STI 33552915

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 6/25/12



Scott Head
Manager, Regulatory Affairs
South Texas Project Units 3 & 4

rhs

Attachments:

1. RAI 01.05-1 Response
2. RAI 01.05-2 Response
3. RAI 01.05-3 Response
4. RAI 01.05-4 Response
5. FSAR Appendix 1E

cc: w/o attachment except*
(paper copy)

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RAI 01.05-1**QUESTION:**

This request for additional information (RAI) specifically addresses Recommendation 2.1, of the Fukushima Near-Term Task Force recommendations contained in SECY-12-0025 as it pertains to the seismic hazard evaluation. This recommendation specifies the use of NUREG-2115, "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," (CEUS-SSC) in a site probabilistic seismic hazard analysis (PSHA). Consistent with Recommendation 2.1, as well as the need to consider the latest available information in the (PSHA) for South Texas Project Units 3 and 4 planned reactor site, the NRC staff requests that STP:

- a) Evaluate the potential impacts of the newly released CEUS-SSC model, with potential local and regional refinements as identified in the CEUS-SSC model, on the seismic hazard curves and the site-specific ground motion response spectra (GMRS)/foundation input response spectra (FIRS). For re-calculation of the PSHA, please follow either the cumulative absolute velocity (CAV) filter or minimum magnitude specifications outlined in Attachment 1 to Seismic Enclosure 1 of the March 12, 2012 letter "Request for information pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendations 2.1.2.3, and 9.3, of the near-term task force review of insights from the Fukushima Dai-Ichi accident." (ML12053A340).
- b) Modify the site-specific GMRS and FIRS if you determine changes are necessary given the evaluation performed in part a) above.

In order to minimize delays to the current licensing schedule, the staff requests that you respond within 60-days of receipt of this RAI or provide a schedule for your response within 30-days.

RESPONSE:

- a) An evaluation of the potential impact of the newly released CEUS SSC model (NUREG-2115, NRC, 2012a) on the characterization of seismic hazard curves and the site-specific ground motion response spectra (GMRS) shows that hazard curves and spectra developed from the CEUS SSC model are not significantly different from the hazard curves and spectra developed for STP 3 & 4 Combined License Application (COLA) (Part 2, Tier 2, FSAR Figure 2.5S.2-52) and confirms the original information in the STP 3 & 4 COLA.

The basis for this conclusion is a comparison of the STP 3 & 4 site GMRS with spectral accelerations developed from the Houston, Texas, demonstration site analysis provided in NUREG-2115, Chapter Eight.

For this comparison, values for the STP 3 & 4 site GMRS are taken directly from FSAR Table 2.5S.2-21 of the STP 3 & 4 COLA for 38 frequencies ranging from 100 Hz to 0.1 Hz. These values include site-specific amplification factors going from rock to free ground surface soil responses at the GMRS horizon.

Values for CEUS SSC rock 10^{-4} and 10^{-5} mean annual frequencies of exceedance (MAFEs) for the Houston demonstration site can be measured from the curves shown in NUREG-2115 (NRC, 2012a) Figures 8.2-3d, 8.2-3e, and 8.2-3f for 10 Hz, 1 Hz and PGA (taken as equivalent to 100 Hz response motions) spectral accelerations, respectively. These values can also be interpolated from a suite of mean spectral accelerations given in Table 8.2.3-1, again for 10 Hz, 1 Hz and PGA.

One additional CEUS SSC-based pair of 10^{-4} and 10^{-5} MAFE Houston rock values at 30 Hz was estimated by using the ratio of 100 Hz to 30 Hz rock motions from the STP 3 & 4 FSAR and applying this ratio to the CEUS SSC PGA value. The bases for the assumption that the ratio of 100 Hz to 30 Hz spectral acceleration developed for the STP site would be closely approximated for the Houston demonstration site is that this ratio is stable for a wide range of critical magnitudes and distances, that both the CEUS SSC and STP 3 & 4 COLA models use the same ground motion prediction equations (GMPEs), that the rock probabilistic seismic hazard analysis (PSHA) curves for the three frequencies for which comparison is directly comparable are very similar, and that the hazard in the Houston-STP region varies slowly with exact location.

The bases for the assumption that earthquake hazard in the Houston-STP region varies slowly is that the historical seismicity pattern on which recurrence parameters are based is similar for both the CEUS SSC and STP 3 & 4 COLA studies, that earthquakes in the region are sparse and do not cause significant differentiation in recurrence parameters within the region, and that the rock 10^{-4} and 10^{-5} MAFE results developed from USGS 2008 National Hazard Map Gridded data (<http://earthquake.usgs.gov/hazards/products/conterminous/2008/data/>) are very similar for the Houston and STP 3 & 4 locations (Petersen et al. 2008).

Both the slow variation of earthquake recurrence parameters with location within the Houston-STP 3 & 4 region (as shown in both the USGS (2008) and NUREG-2115 (2012a) studies) and the detailed STP region-specific evaluation of potential earthquake sources indicate that there are no identified potential local and regional refinements needed to the CEUS SSC model in the Houston-STP 3 & 4 region.

NUREG-2115 uses a minimum moment magnitude of 5.0 in the calculation of the Houston demonstration site spectral acceleration values satisfying the minimum magnitude specifications outlined in Attachment 1 to Seismic Enclosure 1 of the March 12, 2012 letter, "Request for information pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendations 2.1, 2.3, and 9.3, of the near-term task force review of insights from the Fukushima Dai-Ichi accident," (NRC 2012b).

Houston rock spectral acceleration values from the CEUS SSC report must be modified to consider subsurface material properties like those found at the STP site in order to make an in-kind comparison of the GMRS. The STP 3 & 4 site is underlain by soils to depths of many hundreds of feet and is, therefore, characterized as a deep soil site. The STP 3 & 4 COLA calculates site-specific factors for the amplification of the base hard rock motion by the

subsurface soil column. These factors are 1.2 for 10 Hz, 3.0 for 1 Hz, and 1.7 for 100 Hz (PGA). The CEUS SSC study develops site amplification factors for a generic "deep soil" site and so characterizes Houston.

Comparison of the generic deep soil subsurface material properties and amplification factors from the CEUS SSC report with the subsurface material properties and amplification factors from the STP 3 & 4 COLA at frequencies of 10 Hz, 1 Hz, and 100 Hz (PGA) shows that the CEUS SSC amplification factors are somewhat lower (approximately 0.9 for 10 Hz, 2.6 for 1 Hz, and 1.4 for 100 Hz [CEUS SSC report Figure 8.1-5]). Because the hard rock motion amplitudes for Houston and the STP 3 & 4 site are similar, and the STP 3 & 4 COLA amplification factors are site specific, the STP 3 & 4 COLA amplification factors are used to estimate the CEUS SSC anticipated free-field ground surface seismic hazard curves.

Using the procedure recommended in RG 1.208 (NRC 2007) (as defined in STP 3 & 4 COLA Section 2.5S.2.6) to develop GMRS spectra from 10^{-4} and 10^{-5} MAFE hazard curves, a hard rock "GMRS" was developed from the three CEUS SSC Houston rock curves and the additional 30 Hz point scaled from a STP 3 & 4 COLA hard rock "GMRS" 100 Hz-to-30 Hz ratio. Again, these values were then assumed to be closely approximate to (and likely conservative for) a CEUS SSC equivalent rock GMRS at the STP 3 & 4 site.

Next, these four points were scaled by the STP FSAR site-specific amplification factors to get approximate CEUS SSC STP 3 & 4 site GMRS points. These points are shown in Figure 1 along with the STP 3 & 4 COLA GMRS spectrum. The figure shows that the estimated CEUS SSC STP 3 & 4 GMRS points developed in this way are very close to, and not significantly above, the STP 3 & 4 COLA points.

Lastly, also shown in Figure 1 is the STP 3 & 4 site-specific SSE. This design spectrum envelops both of the other spectra.

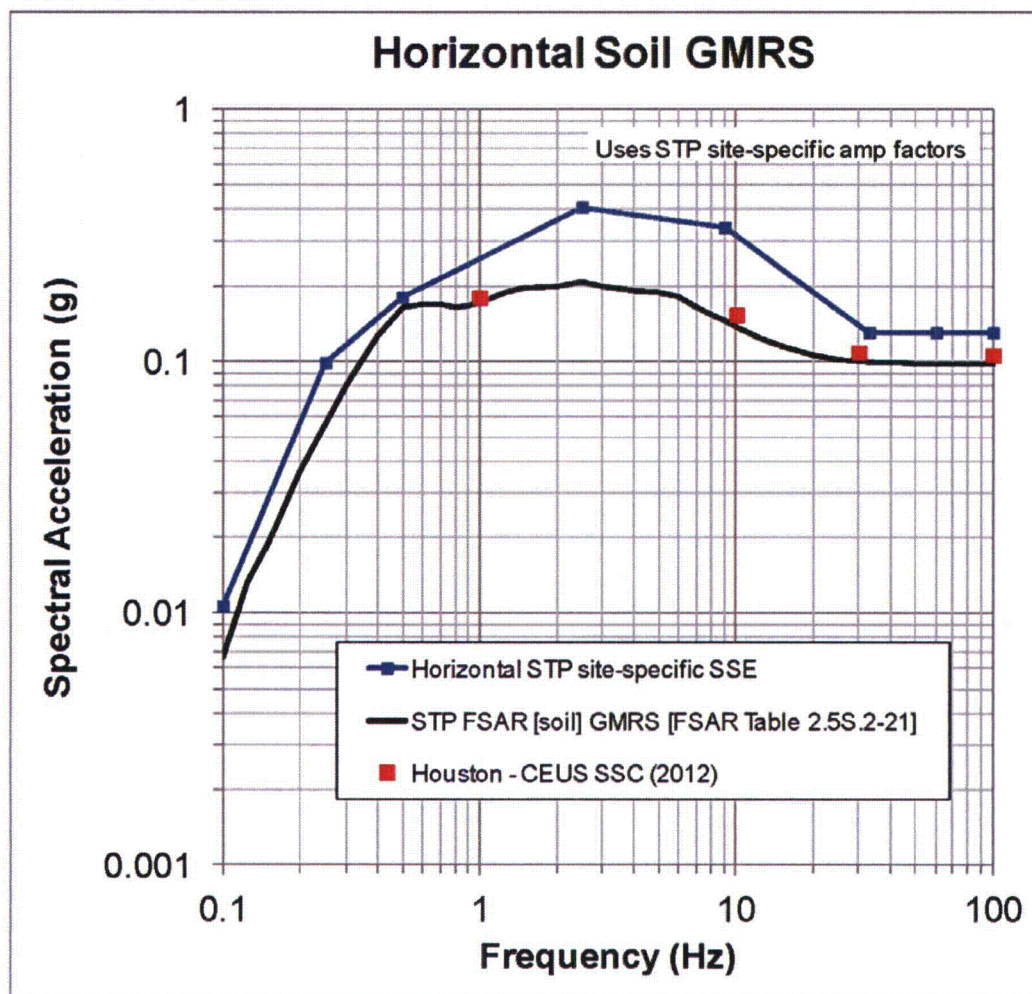


Figure 1. Comparison of the horizontal STP site-specific SSE, the STP 3 & 4 COLA GMRS (from FSAR Table 2.5S.2-21), and the 2012 Houston CEUS SSC GMRS.

Therefore, on the basis of this evaluation, it is concluded that implementation of the CEUS SSC model for the STP 3 & 4 site would be expected to result in similar or slightly higher GMRS motions than predicted in the FSAR, but that this difference is not significant because it is within reasonably expected uncertainty in the characterization of seismic hazard. In addition, and as shown in Figure 1, both the existing STP 3 & 4 COLA results and the estimated CEUS SSC results for the STP sites are enveloped by the SSE design spectrum.

- b) Because the STP 3 & 4 COLA results and the estimated CEUS SSC results for the STP site are not significantly different, this investigation found no reason to revise the seismic design basis in the STP 3 & 4 COLA based on an implementation of the CEUS SSC (NUREG-2115) model.

No COLA revision is required as a result of this RAI response.

REFERENCES

1. Nuclear Regulatory Commission (2007). "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," Regulatory Guide 1.208, U.S. Nuclear Regulatory Commission, Washington, D. C.
2. Nuclear Regulatory Commission (2012a). "Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," NUREG-2115, U.S. Nuclear Regulatory Commission, Washington, D. C.
3. Nuclear Regulatory Commission (2012b). "Request for information pursuant to Title 10 of the Code of Federal Regulations 50.54(f) regarding Recommendations 2.1, 2.3, and 9.3, of the near-term task force review of insights from the Fukushima Dai-Ichi accident." (ML12053A340)
4. Petersen, M.D., Frankel, A.D., Harmsen, S.C., Mueller, C.S., Haller, K.M., Wheeler, R.L., Wesson, R.L., Oliver, Y.Z., Boyd, S., Perkins, D.M., Luco, N., Field, E.H., Wills, C.J., and Rukstales, K.S. (2008). "Documentation for the 2008 Update of the United States National Seismic Hazard Maps," USGS Open-File Report 2008-1128, 128 pp.

01.05-2 SFP Instruments

By letter dated April 25, 2012, the NRC staff informed you that the NRC staff has been directed by the Commission to implement the Fukushima Near-Term Task Force recommendations contained 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" dated February 17, 2012. This request for additional information (RAI) specifically addresses Recommendation 7.1, "Reliable Spent Fuel Pool Instrumentation." The NRC staff requests that you 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.

Response

STP 3 & 4 has created a new FSAR appendix (1E) to address the NRC's post-Fukushima recommendations (see Attachment 5 to this letter). This appendix will be included in the next revision of the STP 3 & 4 COLA. The shaded text in the RAI response below will be included in Appendix 1E, Section 1E.2.6, "Spent Fuel Pool (SFP) Instrumentation (7.1)".

The certified ABWR design includes reliable level and temperature monitors in the SFP that provide indication via the process computer and annunciate in the Main Control Room (MCR). The instruments are powered by battery-backed non-Class 1E vital 120 VAC, normally powered by the Plant Investment Protection (PIP) buses, which are backed-up by the Combustion Turbine Generator (CTG) as described in FSAR Subsection 7.7.1.10.

STP 3 & 4 will enhance the spent fuel pool instrumentation to ensure that it provides a reliable indication of the water level in the spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

1. The spent fuel pool level instrumentation will include the following design features:

1.1 Instruments: The instrumentation will consist of two permanent, fixed instrument channels with level indication from the top of the fuel racks to the top of the spent fuel pool. Level instrumentation will include high and low water level alarms that annunciate in the Control Room.

1.2 Arrangement: The spent fuel pool 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 spent fuel pool. This protection will be provided by maintaining instrument channel separation within the

spent fuel pool area, and will utilize inherent shielding from missiles provided by existing corners in the spent fuel pool structure.

1.3 Mounting: Installed instrument channel equipment within the spent fuel pool will be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.

1.4 Qualification: The instrument channels will be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. This reliability will be established through use of an augmented quality assurance process (e.g., a process similar to that applied to the site fire protection program).

1.5 Independence: The instrument channels will be physically and electrically independent of each other.

1.6 Power supplies: The permanently installed instrumentation channels will be powered by separate Non-Class 1E Vital power supplies. The instrumentation channels will also provide for power connections from alternate sources independent of the plant ac and dc power distribution systems. The independent alternate sources used for instrument channel power will have sufficient capacity to maintain the level indication function for at least 72 hours.

1.7 Accuracy: The instrument channels will maintain their designed accuracy following a power interruption or change in power source without recalibration.

1.8 Testing: The instrument channel design will provide for routine testing and calibration.

1.9 Display: Trained personnel will be able to monitor the spent fuel pool water level from the control room, alternate shutdown panel, or other appropriate and accessible location. The display will provide on-demand or continuous indication of spent fuel pool water level.

2. The spent fuel pool instrumentation will be maintained available and reliable through appropriate development and implementation of the following programs:

2.1 Training: Personnel will be trained in the use and the provision of alternate power to the instrument channels.

2.2 Procedures: Procedures shall be established and maintained for the testing, calibration, and use of the spent fuel pool level instrument channels.

2.3 Testing and Calibration: Processes will be established and maintained for scheduling and implementing necessary testing and calibration of spent fuel pool level instrument channels to maintain the instrument channels at the design accuracy.

01.05-3 EP

By letter dated April 25, 2012, the NRC staff informed you that the NRC staff has been directed by the Commission to implement the Fukushima Near-Term Task Force recommendations contained 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" dated February 17, 2012. This request for additional information (RAI) specifically addresses Recommendation 9.3, "provisions for enhancing emergency preparedness." The NRC staff requests that you address each of the provisions for enhancing emergency preparedness as described in Enclosure 7 of SECY-12-0025, including any proposals for changes to your current application.

Response:

STP 3 & 4 has created a new FSAR appendix (1E) to address the NRC's post-Fukushima recommendations (see Attachment 5). This appendix will be included in the next revision of the STP 3 & 4 COLA. The shaded text in the RAI response below will be included in Appendix 1E, Section 1E.2.8, "Enhanced Emergency Plan Staffing and Communication (9.3)".

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 (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, Item 10 lists the ITAAC applicable to the Emergency Plan and its implementing procedures.

01.05-4 External Events

By letter dated April 25, 2012, the NRC staff informed you that the NRC staff has been directed by the Commission to implement the Fukushima Near-Term Task Force recommendations contained 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" dated February 17, 2012. This request for additional information (RAI) specifically addresses Recommendation 4.2, "Protection for equipment from external events." The NRC staff requests that you address each of the provisions for the protection for equipment from external events as described in Enclosure 4 of SECY-12-0025, including any proposals for changes to your current application.

Response:

STP 3 & 4 has created a new FSAR appendix (1E) to address the NRC's post-Fukushima recommendations (see Attachment 5 to this letter). This appendix will be included in the next revision of the STP 3 & 4 COLA. The shaded text in the RAI response below will be included in Appendix 1E, Section 1.E.2.4, "Mitigating Strategies for Beyond Design Basis Events (4.2)".

STP 3 & 4 incorporates three staged AC independent portable pumping systems:

- Two pumps (a fire truck and a trailer mounted portable pump) shared between STP 3 & 4 provide core, SFP, and containment cooling water to the RHR system via the ACIWA system (Subsection 5.4.7.1.1.10).
 - 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 (Subsection 19.4.6).
- One trailer mounted pump shared between STP 1, 2, 3, & 4 provides water in the event of the loss of large areas of the plant (Part 11, Subsection 5.1.2).
 - This trailer mounted pump is protected primarily by distance.

STP 3 & 4 is monitoring the development of the industry FLEX program (Reference 1E-3) and will implement applicable portions of the program. This industry program is developing diverse and flexible mitigation strategies to address extended loss of power and loss of ultimate heat sink that will increase the defense-in-depth for beyond design basis scenarios. This includes procurement of additional onsite portable equipment that will be stored at diverse locations and be capable of being used to assist in mitigating beyond design basis events. Equipment to be procured 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)

- Four handheld satellite phones (two/unit)
- Various hoses, fittings, cables, and jumpers necessary to connect the above equipment

STP 3 & 4 will develop, implement, and maintain guidance and strategies to restore core, containment, and SFP cooling following a beyond Design Basis Event involving one or both STP 3 & 4 units. The strategies will be capable of mitigating a simultaneous loss of all alternating current (ac) power (including both CTGs) and loss of normal access to the ultimate heat sink, and will be implementable in all operating modes. The equipment required to mitigate the beyond Design Basis Event will be adequately protected from external events.

The guidance will utilize a three-phase approach. Phase I will use installed equipment and resources to maintain or restore core, containment and spent fuel pool (SFP) cooling capabilities. The second or transition phase will provide sufficient portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from off site. The final phase III will obtain sufficient offsite resources to sustain those functions indefinitely.

The detailed procedures and training developed during implementation of operational programs as described in Section 13.4S will be developed in cooperation with Units 1 and 2 as a 4-unit site and will address all of the NRC requirements discussed above.

1E Response to NRC Post-Fukushima Recommendations

1E.1 Introduction

In response to the accident at the Fukushima Daiichi Nuclear Power Plant caused by the March 2011, Magnitude 9 Tohoku earthquake and subsequent tsunami, the Nuclear Regulatory Commission established a Near-Term Task Force (NTTF) to review NRC processes and regulations to determine if improvements to its regulatory system were needed. The NTTF developed a set of recommendations intended to clarify and strengthen the regulatory framework for protection against natural phenomena. These recommendations were issued in SECY-11-0093 (Reference 1E-9).

SECY-11-0124 and SECY-11-0137 provided the NRC Commissioners with the Staff's recommendations, including prioritization for implementation. Subsequently, SECY-12-0025 was issued describing proposed orders to be issued to licensees and a draft request for information pursuant to 10 CFR 50.54(f). SECY-12-0025 stated that combined license plants under review would address the three orders and the request for information through the review process. On March 9, 2012, the Commission issued a Staff Requirements Memorandum (SRM) for SECY-12-0025 (Reference 1E-4) approving issuance of the orders and request for information with some modifications.

This appendix addresses the Tier 1 recommendations and Orders contained in SECY-12-0025, the Tier 2 recommendations contained in SECY-11-0137, and the modifications documented in the SRM consistent with the as issued orders (EA-12-049, 050 and 51) and request for information dated March 12, 2012. The NRC Recommendation in each of the following subsections is a summary of the recommendation from the NRC documents. The response to each recommendation discusses how STP 3 & 4 addresses the recommendation. The numbers in parentheses of subsection headings correspond to the NTTF recommendation number.

1E.2 Tier 1 NRC Recommendation/Responses

1E.2.1 Seismic and Flooding Reevaluations (2.1)

1E.2.1.1 Seismic

NRC Recommendation

Perform a reevaluation of the seismic hazards using present-day NRC requirements and guidance to develop a Ground Motion Response Spectrum (GMRS). The new consensus seismic source models from the Central and Eastern United States Seismic Source Characterization (CEUS), NUREG-2115, may be used to characterize the hazards.

Response

Present-day regulatory guidance and methodologies, including the approach described in Regulatory Guide (RG) 1.208, "A Performance Based Approach to Define the Site-Specific Earthquake Ground Motion," were used to evaluate seismic hazards for the STP 3 & 4 site as discussed in Chapters 2 and 3. The evaluation conducted in conformance with RG 1.208 is discussed in Subsection 2.5S.2.6.

STP 3 & 4 reviewed the updated information provided in the CEUS and confirmed that it does not identify any new hazards that are not adequately considered in Chapters 2 and 3, and is not materially different than the GMRS discussed in 2.5S.2.6.

1E.2.1.2 Flooding**NRC Recommendation**

Perform a reevaluation of all appropriate external flooding sources, including the effects from local intense precipitation on the site, probable maximum flood (PMF) on streams and rivers, storm surges, seiches, tsunami, and dam failures. It is requested that the reevaluation apply present-day regulatory guidance and methodologies being used for ESP and COL reviews including current techniques, software, and methods used in present-day standard engineering practice to develop the flood hazard.

The recommendation also noted that flooding risks are of concern because the safety consequences of a flooding event may increase sharply with a small increase in flooding level.

Response

Present-day regulatory guidance and methodologies were used to evaluate flooding hazards relative to the STP 3 & 4 site as discussed in Section 2.4S.

Scenarios evaluated include:

- Dam Break Analysis (2.4S.4)
- Main Cooling Reservoir (MCR) Embankment Breach Analysis (2.4S.4.2.2)
- Probable Maximum Flood on Streams and Rivers (2.4S.3)
- Local Probable Maximum Precipitation (2.4S.2.3)
- Probable Maximum Surge and Seiche (2.4S.5)
- Probable Maximum Tsunami (2.4S.6)
- Ice Induced Flooding (2.4S.7)
- Channel Diversions (2.4S.9)
- Low Water Considerations (2.4S.11)

Conservatism in the STP 3 & 4 analyses of possible flooding resulting from these events and the plant design minimize the likelihood of even a small increase in flooding level. The postulated MCR embankment breach has been determined to be the design basis flood (DBF) for STP 3 & 4. Very conservative assumptions regarding both the

maximum breach size and the speed at which the breach occurs make it highly improbable that the predicted flood level could be exceeded during an actual MCR breach. Additionally, an MCR breach is highly improbable because:

- Overtopping of the embankment is not possible due to very large freeboard:
 - MCR minimum embankment height is 65.8 feet MSL;
 - MCR operating level is less than 49 feet MSL;
 - Maximum MCR level during a concurrent Probable Maximum Precipitation (PMP) event is 52.6 feet MSL.
- The potential for a seismic induced MCR embankment failure is negligible because of both the embankment design and the low potential for significant seismic activity in the site vicinity.
- An MCR embankment failure at any point except a very small portion of the 12.4 mile embankment perimeter has no impact on site structures.

Although the above discussion demonstrates the improbability of a flood exceeding the design basis flood levels, STP 3 & 4 also performed an analysis to determine at what flood level (Cliff Edge) the ability to cool the core would be lost. Although unachievable in any realistic scenario, this level demonstrates the margin beyond design that is built into STP 3 & 4. The flood level that the EDGs would be lost, and therefore, the ability to cool the core would be lost, was determined to be 51 feet.

MCR embankment breach analysis is described in Subsection 2.4S.4.2.2.

1E.2.2 Seismic and Flooding Walkdowns (2.3)

NRC Recommendations

Perform seismic walkdowns in order to identify and address plant specific degraded, non-conforming, or unanalyzed conditions and verify the adequacy of strategies, monitoring, and maintenance programs such that the nuclear power plant can respond to external events. The walkdown will verify current plant configuration with the current licensing basis, verify the adequacy of current strategies, maintenance plans, and identify degraded, non-conforming, or unanalyzed conditions. The walkdown procedure should be developed and submitted to the NRC. The procedure may incorporate current plant procedures, if appropriate. Prior to the walkdown, licensees should develop acceptance criteria, collect appropriate data, and assemble a team with relevant technical skills

The NRC also requests that each addressee confirm that they will use the industry developed, NRC endorsed, flood walkdown procedures or provide a description of plant-specific walkdown procedures.

Response

This recommendation is not applicable since the STP 3 & 4 units have not yet been built. However, seismic and flooding plant walkdowns will be conducted after construction as documented in COL Information Item 19.9.5.

1E.2.3 Station Blackout (SBO) Rulemaking (4.1)**NRC Recommendation**

Strengthen the station blackout (SBO) mitigation capability at all operating and new reactors for design-basis and beyond-design-basis events. This includes (1) a minimum coping time of 8 hours for loss of all AC, (2) establishing the equipment, procedures, and training necessary to implement an extended coping time of 72 hours for core and spent fuel cooling and for reactor coolant system and primary containment integrity, and (3) pre-planning and pre-staging offsite resources to support uninterrupted core and spent fuel cooling, and RCS and primary containment integrity under conditions involving significant degradation of offsite transportation infrastructure associated with a significant natural disaster. This recommendation will be implemented by rulemaking.

SECY-12-0025 adds the requirement that the loss of the Ultimate Heat Sink (UHS) should be evaluated as part of the SBO evaluation.

Response

(1) The STP 3 & 4 design can withstand an SBO for an indefinite period of time using an alternate AC power source, the Combustion Turbine Generator (CTG), as described in Appendix 1C (Table 1C-2). Additionally, the STP 3 & 4 design can withstand a sustained loss of all AC power, including the loss of both CTGs, for 72 hours while maintaining core cooling, as described in 19E2.1.2.2.2.

The STP 3 & 4 design has a number of features that mitigate an SBO and extended loss of all AC power:

- The primary mitigation for an extended SBO is provided by a CTG, which is independent from the Emergency Diesel Generators (EDGs) and can be connected to the 4.16 KV Class 1E buses. This CTG has black start capability and can be available for use within 10 minutes. There is one CTG per unit, they can be cross-tied, and one CTG can supply the safety loads for both units. The CTGs are housed in International Building Code structures which are protected from the design basis flood and adverse weather conditions. (Tier 1 Section 2.12.11 and Tier 2 Appendix 1C).
- The batteries have a capacity of 8 hours. This capacity can be extended well beyond 8 hours if load shedding is performed (Section 8.3.2.1.3.1).
- The Reactor Core Isolation Cooling system can provide core cooling for at least 8 hours during SBO conditions without reliance on AC power (Appendix 19E.2).
- The Alternating Current-Independent Water Addition (ACIWA) system is a seismically qualified system with an external permanent diesel-driven pump and

water supply capable of providing water to the Residual Heat Removal (RHR) system for core and containment cooling without reliance on AC power (Tier 1 Figure 2.15.6).

- Seismically-qualified external connections on opposite sides of the Reactor Building can be used to provide makeup water and sprays to the Spent Fuel Pool (SFP) with the use of staged portable diesel driven water supply pumps as described in Part 11 (Mitigative Strategies Report) Sections 6.1 and 6.2.

(2) STP 3 & 4 has the installed equipment (e.g., ACIWA system) to implement an extended coping time in excess of 72 hours without reliance on AC power for core and spent fuel cooling and for reactor coolant system and primary containment integrity as documented in Subsection 19E.2.2.3. The 72 hours of core cooling can be provided without reliance on the UHS. Relevant procedures and training will be developed per the Operational Program Development Plan described in Sections 13.4S and 13.5.

(3) Pre-planning and pre-staging resources to support uninterrupted core and spent fuel pool cooling, and RCS and primary containment integrity under conditions involving significant degradation of the onsite facilities associated with large fires and explosions are documented in Part 11. Additionally, as discussed in the next section, STP will arrange for sufficient offsite resources to sustain core, containment, and spent fuel pool cooling indefinitely. These plans and resources will provide this capability under circumstances involving significant degradation of offsite transportation infrastructure associated with a significant natural disaster.

Detailed procedures and training associated with strengthening SBO mitigation capabilities will be developed during implementation of operational programs as described in Section 13.4S.

1.E.2.4 Mitigating Strategies for Beyond Design Basis Events (4.2)

NRC Recommendation

NRC issued an Order to power reactor licensees and holders of construction permits requiring a three-phase approach for mitigating beyond-design-basis external events. The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment and spent fuel pool (SFP) cooling capabilities. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from off site. The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely.

Response

STP 3 & 4 incorporates three staged AC independent portable pumping systems:

- Two pumps (a fire truck and a trailer mounted portable pump) shared between STP 3 & 4 provide core, SFP, and containment cooling water to the RHR system via the ACIWA system (Subsection 5.4.7.1.1.10).
 - 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 (Subsection 19.4.6)
- One trailer mounted pump shared between STP 1, 2, 3, & 4 provides water in the event of the loss of large areas of the plant (Part 11, Subsection 5.1.2).
 - This trailer mounted pump is protected primarily by distance.

STP 3 & 4 is monitoring the development of the industry FLEX program (Reference 1E-3) and will implement applicable portions of the program. This industry program is developing diverse and flexible mitigation strategies to address extended loss of power and loss of ultimate heat sink that will increase the defense-in-depth for beyond design basis scenarios. This includes procurement of additional onsite portable equipment that will be stored at diverse locations and be capable of being used to assist in mitigating beyond design basis events. Equipment to be procured 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)
- Four handheld satellite phones (two/unit)
- Various hoses, fittings, cables, and jumpers necessary to connect the above equipment

STP 3 & 4 will develop, implement, and maintain guidance and strategies to restore core cooling, containment, and SFP cooling following a beyond Design Basis Event (DBE) involving one or both STP 3 & 4 units. The strategies will be capable of mitigating a simultaneous loss of all alternating current (ac) power (including both CTGs) and loss of normal access to the ultimate heat sink, and will be implementable in all operating modes. The equipment required to mitigate the beyond DBE will be adequately protected from external events.

The guidance will utilize a three-phase approach. Phase I will use installed equipment and resources to maintain or restore core, containment and spent fuel pool (SFP) cooling capabilities. The second or transition phase will provide sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from off site. The final phase III will obtain sufficient offsite resources to sustain those functions indefinitely.

The detailed procedures and training developed during implementation of operational programs as described in Section 13.4S will be developed in cooperation with Units 1 and 2 as a 4-unit site and will address all of the NRC requirements discussed above.

1E.2.5 Reliable Hardened Vents (5.1)**NRC Recommendation**

NRC issued an Order to operating Boiling-Water Reactor (BWR) licensees with Mark I and Mark II containments requiring them to have a reliable hardened vent to remove decay heat and maintain control of containment pressure within acceptable limits following events that result in the loss of active containment heat removal capability or prolonged Station Blackout (SBO). The hardened vent system is required to be accessible and operable under a range of plant conditions, including a prolonged SBO and inadequate containment cooling.

Response

This recommendation does not apply since STP 3 & 4 does not have a Mark I or Mark II containment.

However, each STP 3 & 4 unit does have a passive, reliable hardened vent as part of the Containment Overpressure Protection System (COPS). COPS is Seismic Category I and is qualified for accident pressures. The vent paths for the units are not shared. This design is described in Subsection 6.2.5.2.6 and its use in conjunction with long term cooling without AC power to prevent fuel damage is demonstrated in Appendix 19E.2.2.

1E.2.6 Spent Fuel Pool (SFP) Instrumentation (7.1)**NRC Recommendation**

NRC issued an order to power reactor licensees and holders of construction permits requiring them to have a reliable indication of the water level in associated spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

Response

The certified ABWR design includes reliable level and temperature monitors in the SFP that provide indication via the process computer and annunciate in the Main Control Room (MCR). The instruments are powered by battery-backed non-Class 1E vital 120 VAC, normally powered by the Plant Investment Protection (PIP) buses, which are

backed-up by the Combustion Turbine Generator (CTG) as described in Subsection 7.7.1.10.

STP 3 & 4 will enhance the spent fuel pool instrumentation to ensure that it provides a reliable indication of the water level in the spent fuel storage pools capable of supporting identification of the following pool water level conditions by trained personnel: (1) level that is adequate to support operation of the normal fuel pool cooling system, (2) level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, and (3) level where fuel remains covered and actions to implement make-up water addition should no longer be deferred.

1. The spent fuel pool level instrumentation will include the following design features:

1.1 Instruments: The instrumentation will consist of two permanent, fixed instrument channels with level indication from the top of the fuel racks to the top of the spent fuel pool. Level instrumentation will include high and low water level alarms that annunciate in the Control Room.

1.2 Arrangement: The spent fuel pool 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 spent fuel pool. This protection will be provided by maintaining instrument channel separation within the spent fuel pool area, and will utilize inherent shielding from missiles provided by existing corners in the spent fuel pool structure.

1.3 Mounting: Installed instrument channel equipment within the spent fuel pool will be mounted to retain its design configuration during and following the maximum seismic ground motion considered in the design of the spent fuel pool structure.

1.4 Qualification: The instrument channels will be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. This reliability will be established through use of an augmented quality assurance process (e.g., a process similar to that applied to the site fire protection program).

1.5 Independence: The instrument channels will be physically and electrically independent of each other.

1.6 Power supplies: The permanently installed instrumentation channels will be powered by separate Non-Class 1E Vital power supplies. The instrumentation channels will also provide for power connections from alternate sources independent of the plant ac and dc power distribution systems. The independent alternate sources used for instrument channel power will have sufficient capacity to maintain the level indication function for at least 72 hours.

1.7 Accuracy: The instrument channels will maintain their designed accuracy following a power interruption or change in power source without recalibration.

1.8 Testing: The instrument channel design will provide for routine testing and calibration.

1.9 Display: Trained personnel will be able to monitor the spent fuel pool water level from the control room, alternate shutdown panel, or other appropriate and accessible location. The display will provide on-demand or continuous indication of spent fuel pool water level.

2. The spent fuel pool instrumentation will be maintained available and reliable through appropriate development and implementation of the following programs:

2.1 Training: Personnel will be trained in the use and the provision of alternate power to the instrument channels.

2.2 Procedures: Procedures shall be established and maintained for the testing, calibration, and use of the spent fuel pool level instrument channels.

2.3 Testing and Calibration: Processes will be established and maintained for scheduling and implementing necessary testing and calibration of spent fuel pool level instrument channels to maintain the instrument channels at the design accuracy.

1E.2.7 Emergency Procedures Rulemaking (8.0)

NRC Recommendation

Strengthen and integrate onsite emergency response capabilities such as emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and extensive damage mitigation guidelines (EDMGs). This includes modification of Technical Specifications to reference the approved EOP technical guidelines and providing more realistic, hands-on training on SAMGs and EDMGs.

Response

STP 3 & 4 procedure development will integrate the EOPs, SAMGs, and EDMGs by using the following guidance:

- Industry (BWROG) guidance as endorsed by applicable NRC regulatory guides consistent with the Task Force recommendation (SECY-11-0124).
- Plant Specific Technical Guidelines (PSTGs), EOPs and SAMGs development activities using as inputs the standard ABWR guidelines (Sections 13.5 and 1A.2) and generic industry guidance per NEI 91-04, Revision 1, Severe Accident Issue

Closure Guidelines, which includes the industry commitment to incorporate severe accident strategies into the overall accident management program.

- EDMGs development as described in NEI 06-12 (Mitigative Strategies Report).

Chapter 13 describes the procedure development plan.

The STP 3 & 4 Technical Specifications meet the requirement to reference the approved EOP Guidelines. (Technical Specifications 5.5.1.1.b)

Training development requirements in Section 13.2 and 13.4S will meet the applicable requirements for realistic hands-on training.

1E.2.8 Enhanced Emergency Plan Staffing and Communication (9.3)

NRC Recommendation

Assess current communications systems and equipment used during an emergency event assuming the potential onsite and offsite damage as a result of a large scale natural event resulting in a loss of all alternating current (AC) power. It is also requested that consideration be given to any enhancements that may be appropriate for the emergency plan with respect to communications requirements of 10 CFR 50.47, Appendix E to 10 CFR Part 50, and the guidance in NUREG-0696 in light of the assumptions stated above. Also consider the means necessary to power the new and existing communications equipment during a multi-unit event, with a loss of all AC power.

Assess current staffing levels and determine the appropriate staff to fill all necessary positions for responding to a multi-unit event during a beyond design basis natural event and determine if any enhancements are appropriate given the considerations of NTTF Recommendation 9.3.

Response

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 (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, Item 10 lists the ITAAC applicable to the Emergency Plan and implementing procedures.

1E.3 Tier 2 NRC Recommendations/Responses**1E.3.1 Other Natural External Hazards (2.1)****NRC Recommendation**

Reevaluate and upgrade as necessary the design basis of structures, systems, and components important to safety for protection against natural external hazards other than seismic and flooding.

Response

The hazards and natural phenomena potentially affecting the STP 3 & 4 site have been identified, screened and evaluated in accordance with the latest revisions of the Standard Review Plan. The review and conclusions are documented in Chapters 2 and 3 along with the appropriate design features necessary to mitigate the events. The natural events of particular interest at the STP site are hurricane wind and missiles. STP 3 & 4 meets the latest regulatory guidance document (RG 1.221) for hurricane winds and missiles, which is based on an extreme hurricane (Subsection 2.3S.1.3.3.2 and 3H.11).

1E.3.2 Safety-related AC electrical power for the SFP makeup system (7.2)**NRC Recommendation**

NRC to issue an order requiring safety related AC power for the SFP makeup system.

In accordance with SECY-11-0137, Recommendation 7.2 will be implemented by rulemaking to provide reliable SFP instrumentation and makeup capabilities

Response

The STP 3 & 4 design provides emergency makeup to the SFP using any of the three trains of the RHR system, which are powered by safety-related AC power (Tier 1, Subsection 2.4.1).

1E.3.3 Technical Specifications requirement for onsite emergency power (7.3)**NRC Recommendation**

NRC to issue an order to revise technical specifications to require that one train of onsite emergency power be operable for SFP makeup and instrumentation whenever spent fuel is in the SFP, regardless of the operational mode of the reactor.

In accordance with SECY 11-0137, Recommendation 7.3 will be implemented by rulemaking to provide reliable SFP instrumentation and makeup capabilities.

Response

The STP 3 & 4 Technical Specifications require at least one Emergency Diesel Generator and one Residual Heat Removal (RHR) pump to be operable in all modes. The safety related RHR system is backed by the emergency diesel generators and can also be powered by the Combustion Turbine Generator. The RHR system is capable of providing makeup to the SFP.

1E.3.4 Spent Fuel Pool Spray (7.4)**NRC Recommendation**

NRC to issue order requiring seismically qualified means to spray water into the spent fuel pools, including an easily accessible connection to supply the water (e.g., using a portable pump or pumper truck) at grade outside the building

In accordance with SECY 11-0137, Recommendation 7.4 will be implemented by rulemaking to provide reliable SFP instrumentation and makeup capabilities

Response

STP 3 & 4 has committed to install a diverse spent fuel pool makeup and spray system as described in Part 11 (Mitigative Strategies Report), Section 6.0 that meets the criteria specified in this recommendation

1E.4 References

1E-1 SECY-11-0137, "Prioritization of Recommended Actions to be taken in response to Fukushima Lessons Learned" October 3, 2011.

1E-2 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"

1E-3 NEI 12-XX, "FLEX Implementation Guide, Rev. A2" February 2012 (DRAFT)

1E-4 SRM for SECY 12-0025, "Staff Requirements-SECY-12-0025 Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Tohoku Earthquake and Tsunami" March 9, 2012

1E-5 EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond Design-Basis External Events," March 12, 2012

1E-6 EA-12-050, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents" March 12, 2012

1E-7 EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," March 12, 2012

1E-8 NUREG-2115, "Central and Eastern United States Seismic Source Characterization"

1E-9 SECY-11-0093, "The Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," July 12, 2011

1E-10 NEI 12-01, Guidelines for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities