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December 6, 2012
U7-C-NINA-NRC-120062

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 revised responses to NRC staff questions included in the Request for Additional Information (RAI) letter referenced above. The attachments provide the revised responses to the following RAI questions which supersede the previous responses.

01.05-1 01.05-2 01.05-3 01.05-4

Attachment 5 contains a copy of FSAR Appendix 1E with pending COLA changes shown in shaded text. The FSAR changes shown in shaded text in this submittal will be included in the next revision of the STP 3 & 4 COLA. This submittal completes the response to the referenced letter.

The commitment made in this letter is documented in Attachment 6.

If you have any questions, please contact me at (979) 316-3011 or Bill Mookhoek at (979) 316-3014.

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NRO

STI 33600445

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

Executed on 12/6/12

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

rhs

Attachments:

1. RAI 01.05-1 Revised Response
2. RAI 01.05-2 Revised Response
3. RAI 01.05-3 Revised Response
4. RAI 01.05-4 Revised Response
5. FSAR Appendix 1E
6. New Commitment

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

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RAI 01.05-1, Revision 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 spectral accelerations at 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 3 & 4 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 (McGuire et al., 2001), that both the CEUS SSC and STP 3 & 4 COLA models use the same ground motion prediction equations (GMPEs) for all seismic sources identified in the STP 3 & 4 FSAR model as contributing to 99% of the total hazard, 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 (e.g., Petersen et al., 2008).

There are several reasons to believe that earthquake hazard in the Houston-STP region varies slowly. First, the historical seismicity pattern on which recurrence parameters are based is similar for both the CEUS SSC and STP 3 & 4 COLA studies. Figures 1a and 1b show regional earthquakes as cataloged for these two studies. Differences in detail exist due to the slightly different periods of coverage and the different magnitude scales used in the two studies (body-wave magnitude, m_b, for EPRI-SOG and moment magnitude, M_w, for CEUS SSC), but the general pattern and, in particular, the general quiescence of the region surrounding both the Houston and STP sites (shown in the figures) are clear.

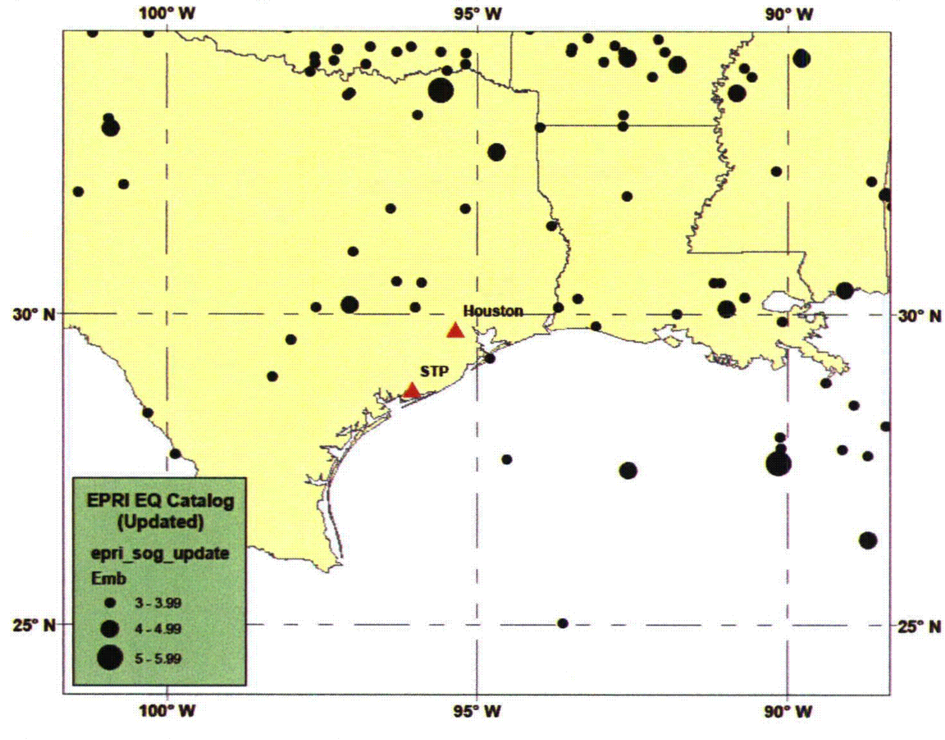

Figure 1a. STP-updated EPRI-SOG regional mainshock [independent] seismicity from the STP 3 & 4 FSAR. Magnitudes shown use the body-wave magnitude [m_b] scale.

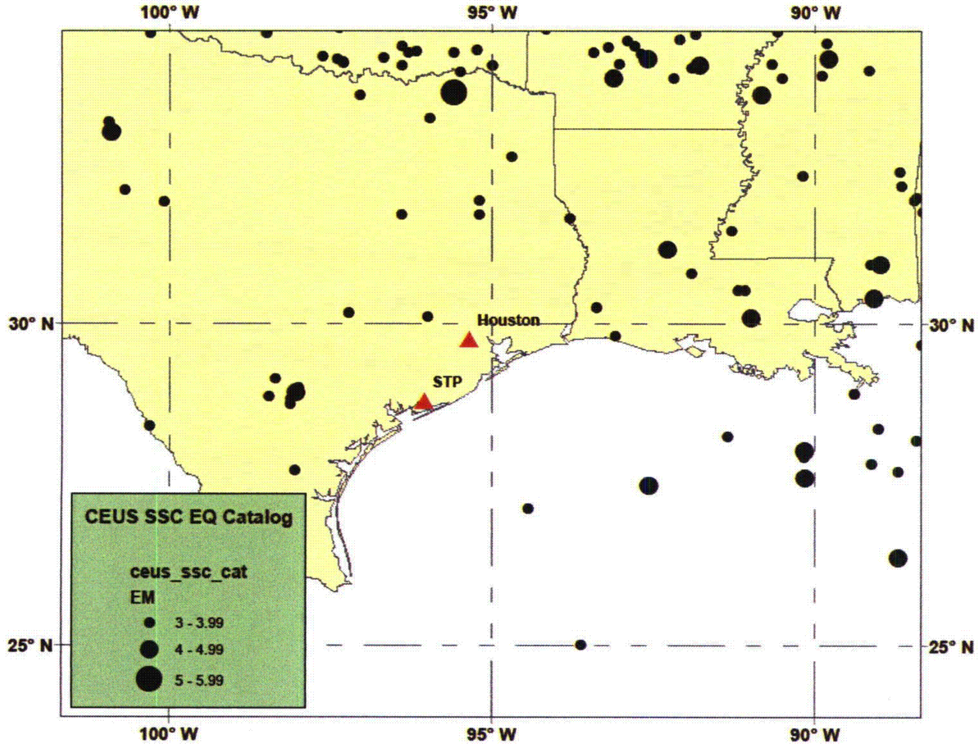

Figure 1b. Map of regional mainshock [independent] seismicity from the CEUS SSC seismicity catalog [NRC, 2012a]. Magnitudes shown use the moment magnitude [M_w] scale.

Second, the STP and Houston sites also share similar geologic and tectonic settings. For example, both sites are located in the Coastal Prairies section of the Gulf Coastal Plain Physiographic Province (STP Units 3 & 4 COLA FSAR Figure 2.5S.1-6), whose geologic and tectonic evolution is discussed in the COLA (FSAR Section 2.5S.1.1.4). Both sites are within a broad region characterized by relatively uniform northeast-southwest compression (FSAR Section 2.5S.1.1.4.2). This is reflected in the characterization of regional seismic source zones in both the STP 3 & 4 COLA (see Figure 2.5S.2-8) and the CEUS SSC report [e.g., see Figures 6.2-2 and 7.1-1 of the final CEUS SSC report (NRC, 2012a)].

Third, the earthquake activity parameter, "a", of the usual Gutenberg-Richter (1956) recurrence relation, $\log_{10}N = a - bM$ (where N is the cumulative number of earthquakes in a given time, M is magnitude, and a and b are curve fitting parameters), especially when smoothed over local or regional areas, would not be expected to show sharp geographical differences in an area of widely scattered small to moderate earthquakes and similar geologic and tectonic settings. Indeed this is borne out by the $\log_{10}N_5$ values - where N_5 is the averaged smooth annual cumulative rate of earthquakes for magnitude 5 and greater per equatorial degree - used to develop rock 10^{-4} and 10^{-5} mean annual frequency of exceedance (MAFE) results in the CEUS SSC and STP 3 & 4 COLA studies. This is shown in Figures 2a and 2b.

Both figures again show the Houston and STP locations and the $\log_{10}$ of interpolated N_5 values at these two sites using inverse-square weighting of the values at the four closest grid points. As with the seismicity maps there are differences in detail, mostly attributable to the different magnitude scales used for the two characterizations, but the geographic variation in $\log_{10}N_5$ is slow and the implied regional earthquake activity suggests that the activity around the STP 3 & 4 site is slightly lower than (that is, of a more negative value) or very similar to (that is, of the same value within the significant digits shown) that around the Houston site.

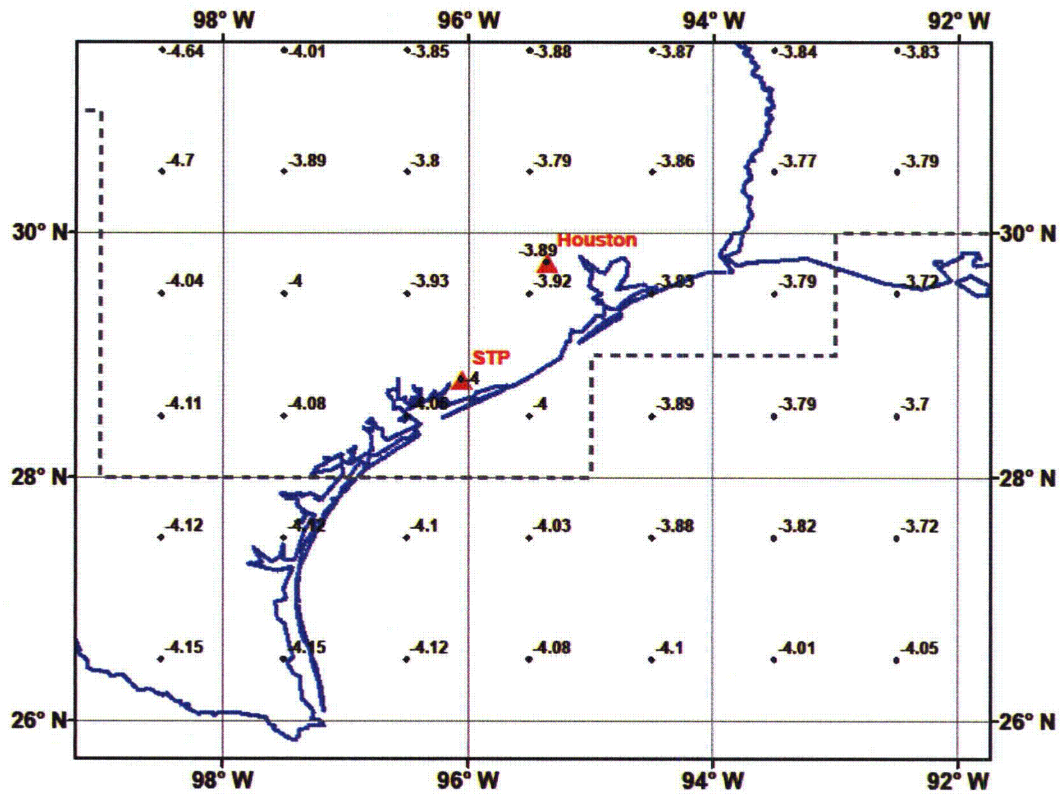

Figure 2a. Activity rates for the average of the STP-updated EPRI-SOG ESTs (Earth Science Teams), presented here by $\log_{10}N_5$, where N_5 is the averaged smooth cumulative rate of earthquakes for magnitude m_b 5 and greater per equatorial degree. Grid point spacing is 1.0 degree. The dashed line shows the southern boundary of the EPRI incompleteness regions.

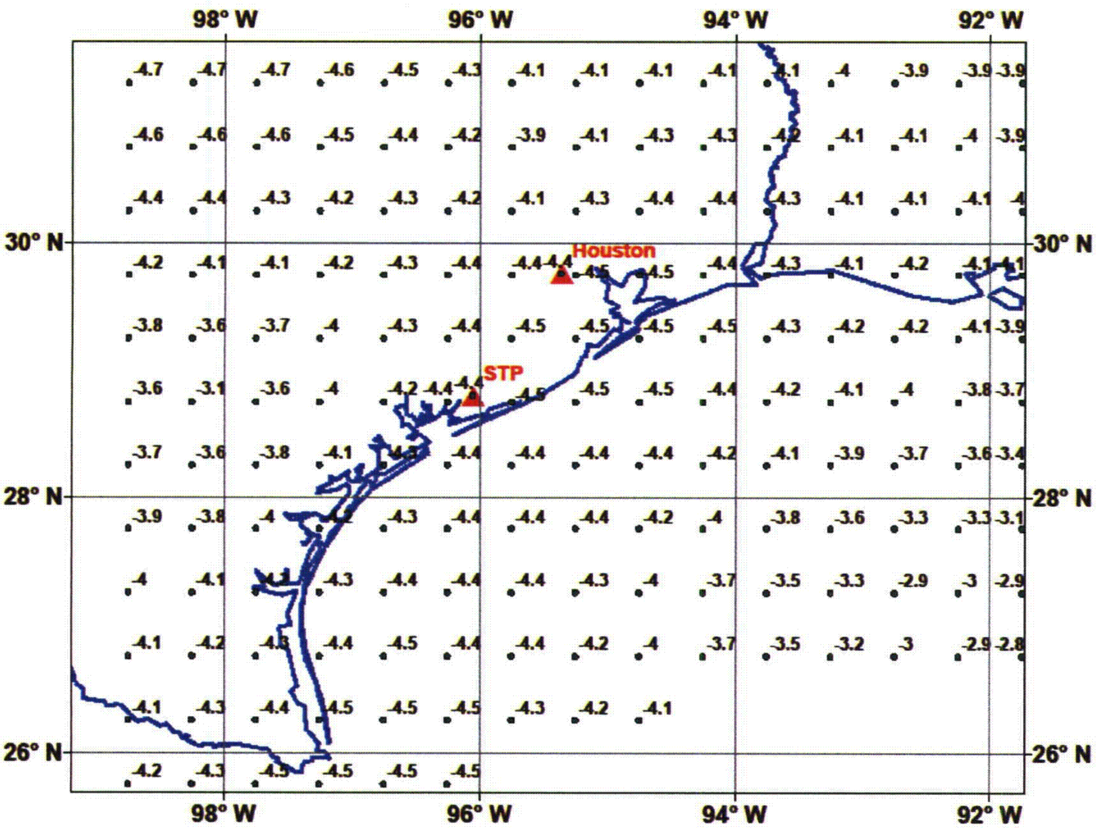

Figure 2b. Activity rates for the CEUS SSC presented here by $\log_{10}N_5$, where N_5 is the averaged smooth cumualative rate of earthquakes for magnitude M_w 5 and greater per equatorial degree. Grid point spacing is 0.5 degrees.

To emphasize this last point the numbers in Figures 2a and 2b have been normalized in Figures 3a and 3b by the Houston N_5 value. The numbers in these figures imply that N_5 varies from 20% less at STP than at Houston (the STP COLA Units 3 & 4 seismic source model) to 10% higher at STP than at Houston (the CEUS SSC seismic source model).

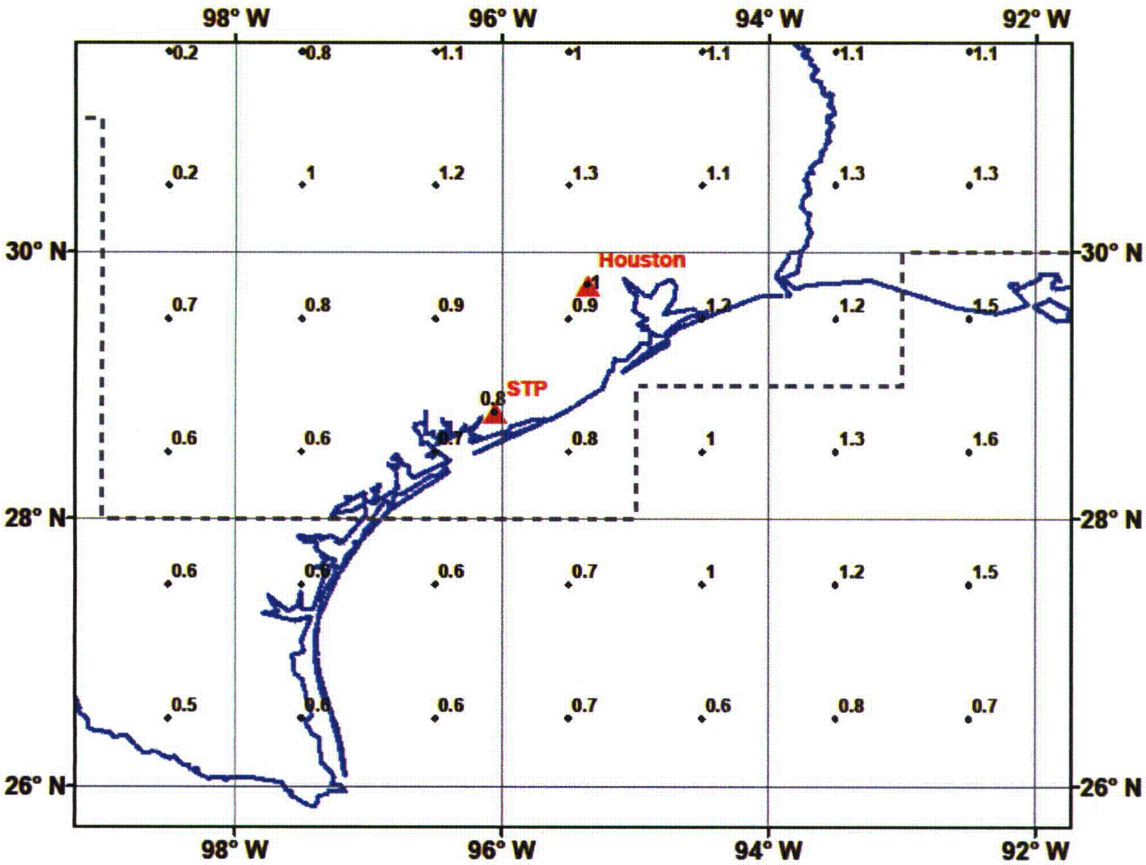

Figure 3a. Relative values of activity rates, $N_5/N_{5,Houston}$, for the STP-updated EPRI-SOG ESTs. The dashed line shows the southern boundary of the EPRI incompleteness regions.

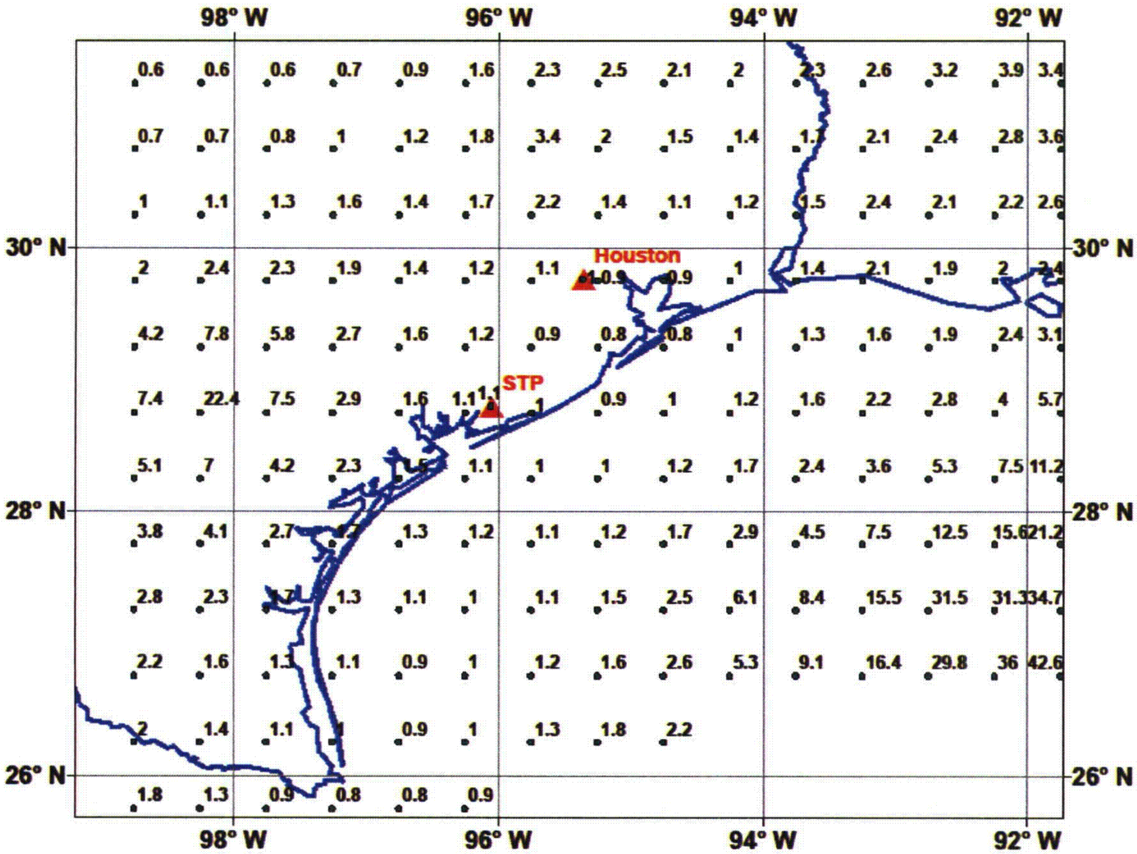

Figure 3b. Relative values of activity rates, $N_5/N_{5,Houston}$, for the CEUS SSC study.

Against this background several comparisons of earthquake hazard at the STP site and Houston may be made. These are shown in Figures 4a through 4c.

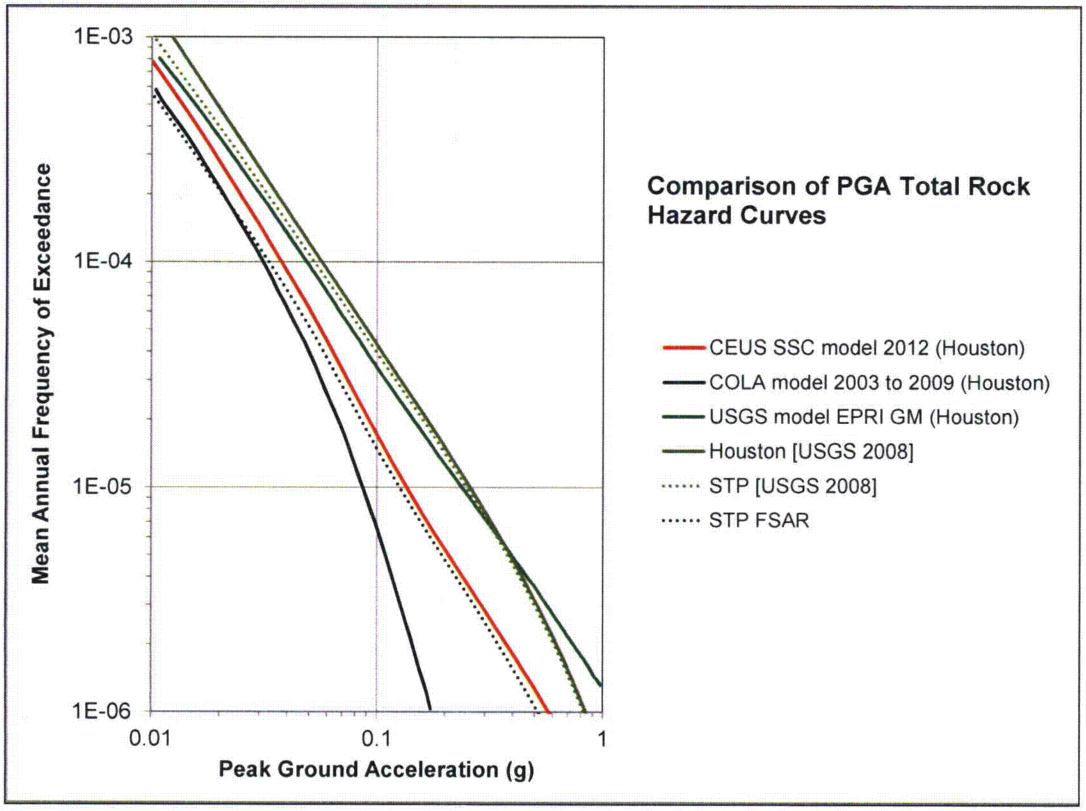

Figure 4a. PGA total rock hazard curves from CEUS SSC (Figure 8.2-31) the STP 3 & 4 COLA, and the USGS (2008). See text for description.

Three of the PGA total rock hazard curves that are shown in Figure 4a are taken directly from the CEUS SSC report. These are all for the Houston demonstration site but using three different seismic source models. The hazard curve for Houston using the full CEUS SSC model is shown as a solid red line. This is the curve that is believed to be a good approximation of what the implementation of the full CEUS SSC model would give at the STP site. Other hazard curves from the CEUS SSC report for Houston are called "USGS Model EPRI GM" (solid green line) and the "COLA model 2003 to 2009," (solid black line).

The basis for the USGS curve appears unambiguous. It marries the geographic and magnitude distributions of earthquakes from Petersen et al. (2008) with the ground motion prediction equations (GMPE) and updated equation uncertainties from the Electric Power Research Institute (EPRI, 2004 and 2006, respectively). The basis for the "COLA model 2003 to 2009" model is only very briefly described in the CEUS SSC report and is less unambiguous. This is discussed further below.

The remaining three hazard curves of Figure 4a are two curves developed from the 2008 USGS National Hazard Map Gridded data (Petersen et al., 2008, with digital earthquake recurrence parameter values associated with the implementation of this model cited throughout as USGS (2008) (downloaded from http://earthquake.usgs.gov/hazards/products/conterminous/2008)) for Houston (solid olive green line) and the STP site (dotted olive green line) and the hazard curve directly from the STP Units 3 & 4 COLA (see Figure 2.5S.2-18) (dotted black line).

A number of implications can be drawn from comparisons of these six hazard curves. First, the only intra-model [seismic source and GMPE] comparison – that of the USGS (2008) hazard curves for Houston and the STP site, shows that these curves are very close to one another, the STP curve falling slightly below the Houston curve for all PGAs. This agrees with arguments made above that, based on regional seismicity patterns and their similar geologic and tectonic setting, hazard varies only very slowly between Houston and the STP site, and that the hazard at STP would be expected to be very similar to that at Houston.

Differences between the solid green "USGS Model EPRI GM" and solid olive green USGS Houston curves reflect the effect of the different GMPEs used. Although larger than differences arising from the locations of Houston and STP, these differences are still much less than the differences among the three CEUS SSC report curves. Since each of these three CEUS SSC report curves use the same EPRI (2004, 2006) ground motion models, it can reasonably be assumed that the sources in each of these three models, contributing most significantly to the hazard at Houston, used the Gulf Coast GMPEs for local sources and Mid-continent GMPEs for more distant sources to the north, such as New Madrid - just as what was done in the STP Unit 3 & 4 COLA - and that the differences among the three CEUS SSC report curves most likely arise from a combination of differences in their Gutenberg-Richter recurrence parameters and/or maximum magnitude distribution of contributing seismic sources.

The comparison of these same six curves for spectral frequencies of 10 Hz and 1 Hz are shown in Figures 4b and 4c, respectively.

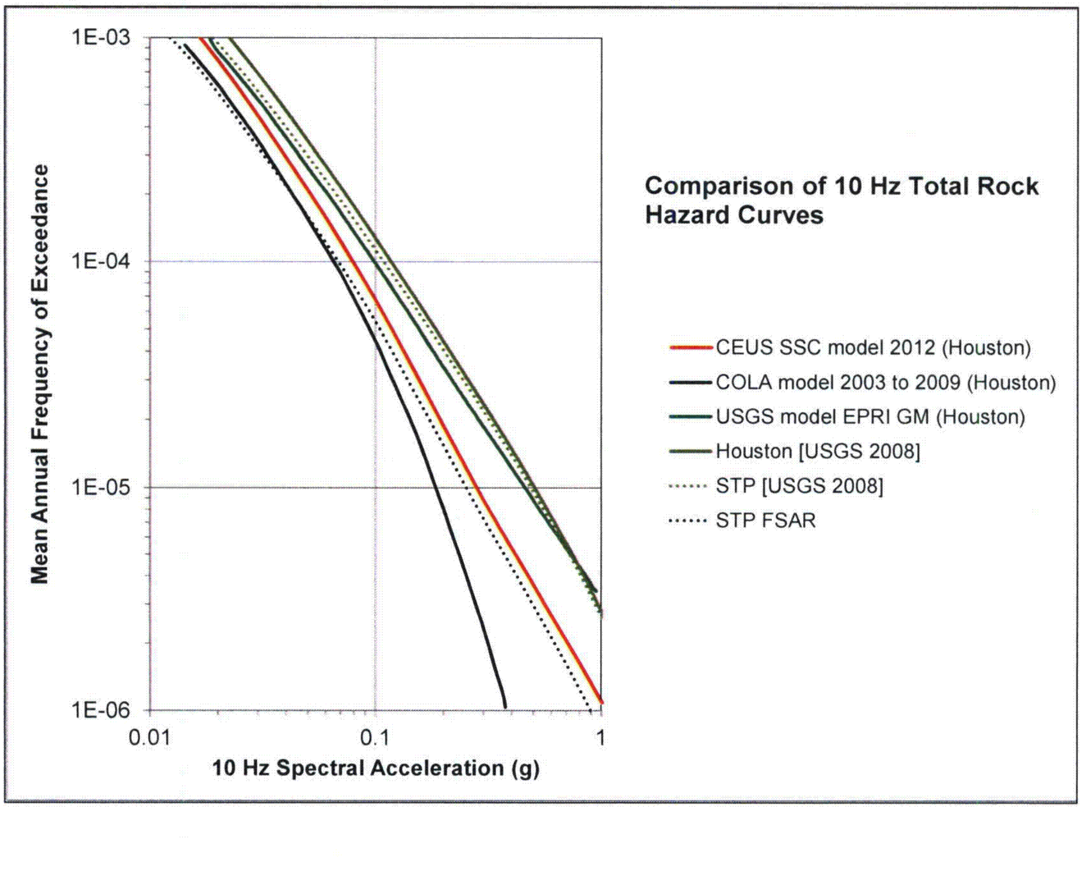

Figure 4b. 10 Hz spectral acceleration total rock hazard curves from CEUS SSC (Figure 8.2-3j) the STP 3 & 4 COLA, and the USGS (2008).

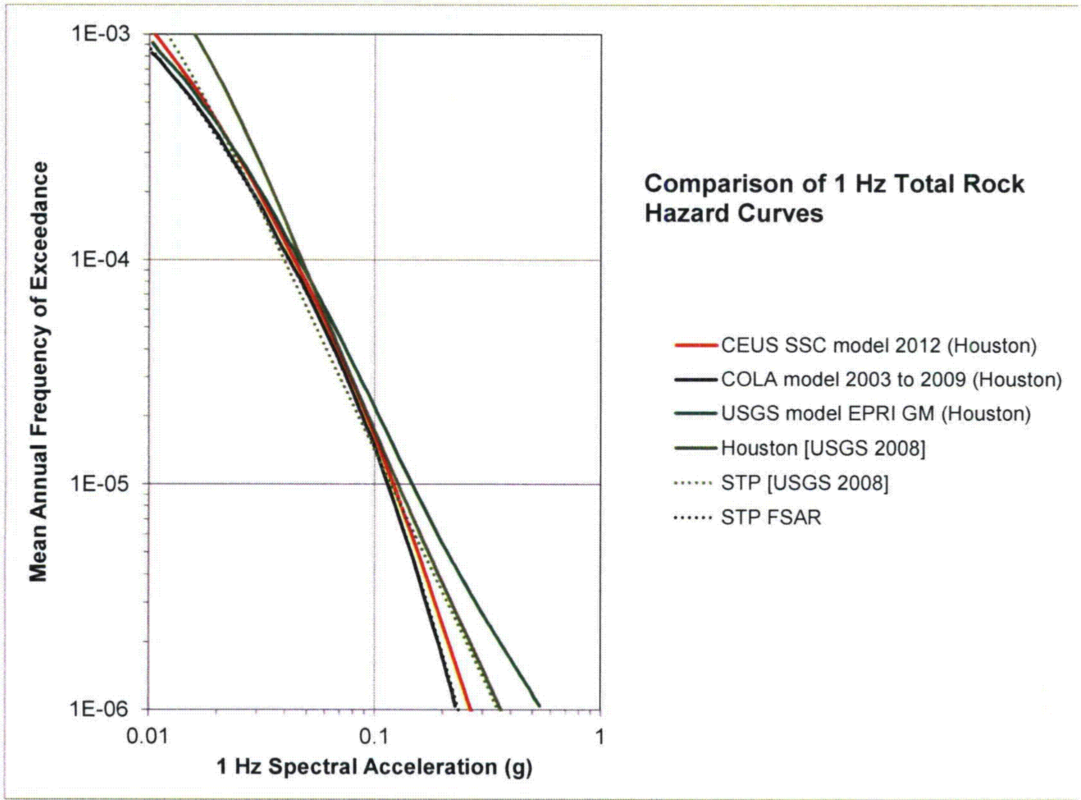

Figure 4c. 1 Hz spectral acceleration total rock hazard curves from CEUS SSC (Figure 8.2-3k) the STP 3 & 4 COLA, and the USGS (2008).

The STP Units 3 & 4 rock hazard curves [dotted black] match the shape and have lower amplitudes than the Houston demonstration site rock hazard curves using the CEUS SSC model [solid red] over the three frequencies for which a direct comparison is available and for the entire range of ground motion accelerations investigated. All curves appear to match each other better as the ground motion frequency decreases, especially in the 10^{-4} and 10^{-5} MAFE range of greatest interest for GMRS development.

Both the slow variation of earthquake recurrence parameters with location within the Houston-STP 3 & 4 region (as shown in Figures 3a and 3b) 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. That is, earthquake hazard would be expected to vary more slowly with specific location within this region than in a region divided by identified tectonic boundaries in the seismic source model.

Given the above observations, given that there is very little difference between Houston and STP site results in the only case where an intra-model comparison is available (the USGS, 2008 results), and given that the comparison between CEUS SSC Houston and STP Units 3 & 4 COLA results show similar ground motion hazard values for the STP Units 3 & 4 site as would be implied by the N_5 value distribution, we conclude that the CEUS SSC Houston demonstration site results provide a reasonable estimate for CEUS SSC model rock results at the STP site within the uncertainty in seismic source characterization among the several models examined.

A question remains if, in comparing the CEUS SSC "COLA model 2003 to 2009" hazard curves at Houston it is assumed that these hazards should match the hazard curves that would be generated at Houston from the seismic source model used for the STP Units 3 & 4 COLA.

As mentioned above there is very little description of the "COLA model 2003 to 2009" model in the CEUS SSC report. The report says only (NRC 2012a, p 8.2):

> "*Figures j-l:* Comparison of mean rock hazard from three source models for 10 Hz SA, 1 Hz SA, and PGA. This comparison shows total hazard for the current CEUS SSC source model and for two other source models, all using the EPRI (2004, 2006) ground motion model. One source model is the USGS model developed for the National Seismic Hazard Mapping Project (Petersen et al., 2008). The other is the "COLA" model that has been used for nuclear power plant licensing applications since 2003. This is the EPRI-SOG (EPRI, 1988) model updated with more recent characterizations of several seismic sources. The updated New Madrid fault source (NMFS) is based on the Clinton and Bellefonte applications, and the updated Charleston seismic zone is based on the Vogtle application. Also, maximum magnitude (M_{max}) values for some seismic sources near the Gulf of Mexico coastline were updated to reflect recent seismicity. Calculations of hazard for all three models use the EPRI (2004, 2006) ground-motion equations, so the differences in hazard presented here between the three models is attributable to differences in the source models themselves."

While the "updated EPRI-SOG" model used to calculate the rock "COLA model 2003 to 2009" hazard curves is similar to the EPRI-SOG update used for the STP Units 3 & 4 COLA (both use the same updated New Madrid source characterization noted and the update of Gulf of Mexico maximum magnitudes) there is one important difference. That is, the spatial distribution of "a" and "b" values of the original EPRI-SOG model was apparently not updated to recognize that in the original EPRI-SOG model local seismicity near STP and within the Gulf of Mexico had been under-represented or simply omitted as too near or south of the boundary of the EPRI-SOG incompleteness regions, needed to specify the "a" and "b" values – see Figure 5. These issues were treated extensively in the STP Units 3 & 4 COLA and remedied – see FSAR Sections 2.5S.2.1.5 and 2.5S.2.4.2.2.

A critical comparison of the seismic activity rates of the original EPRI-SOG (Figure 5) with those of the STP-updated activity rates (Figure 2a) shows for the original EPRI-SOG not only the obvious empty grid points of seismic activity values in the near offshore of the Gulf of Mexico, but in the immediate vicinity of both Houston and STP the $\log_{10}N_5$ values are slightly higher [less negative] for the updated EPRI-SOG characterization in Figure 2a. Therefore, for the updated EPRI-SOG model of the Gulf of Mexico seismic sources, not only were empty grid points of seismic activity filled in, but following re-evaluation of the smoothed activity rates, the grid points of activity rate for the updated EPRI-SOG model used at STP show slightly greater activity rates than in the original EPRI-SOG model.

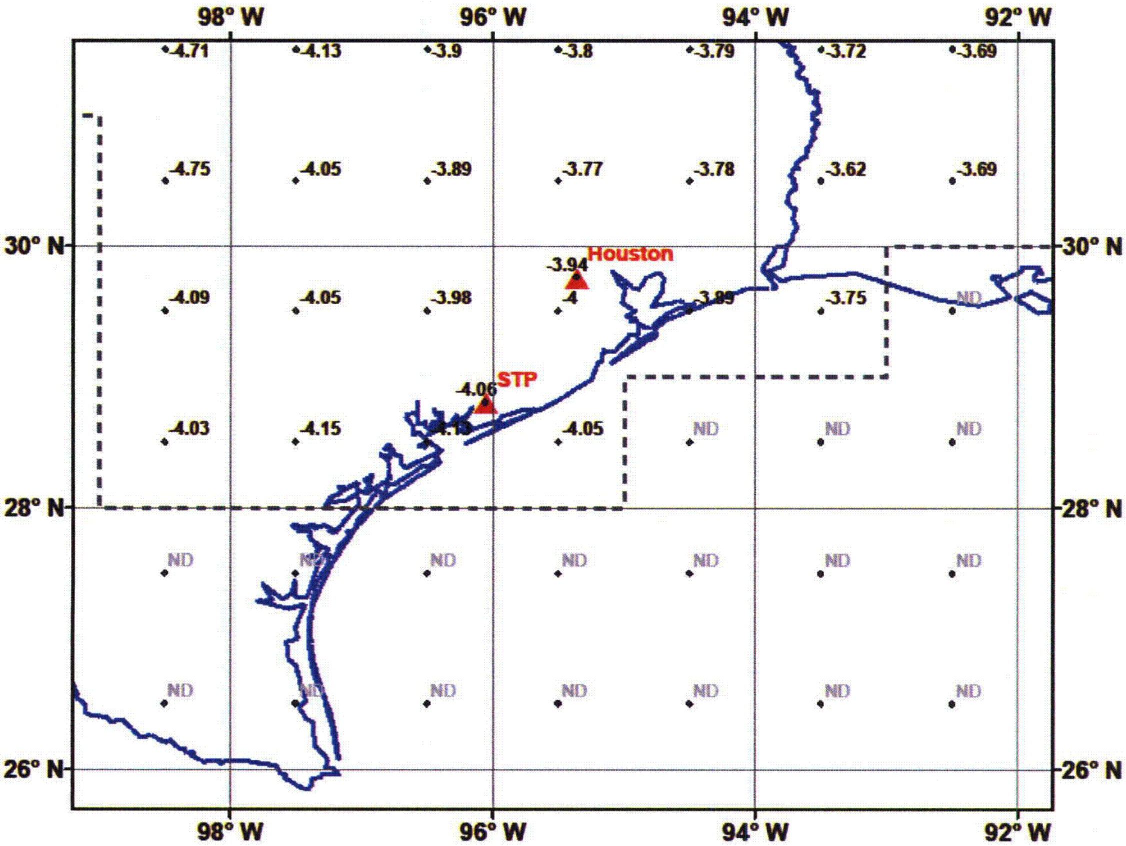

Figure 5. Activity rates for the average of the original EPRI-SOG ESTs, presented here by $\log_{10}N_5$, where N_5 is the averaged smooth cumulative rate of earthquake for magnitude m_b 5 and greater per equatorial degree. Grid point spacing is 1.0 degree. "ND" indicates no data at that grid point. The dashed line shows the southern boundary of the EPRI incompleteness regions.

Therefore, the curves characterized in the CEUS SSC report as "COLA model 2003 to 2009" hazard curves for Houston are not updated to the degree of the updated EPRI-SOG model used for the STP Units 3 & 4 COLA. This difference would be expected to, and apparently does, affect high frequencies (that are contributed by moderate nearby earthquakes in the model) much more than lower spectral frequencies (that are much more dominated by the large magnitude, distant New Madrid source that is the same in both updates).

The CEUS SSC PSHA calculations use a minimum moment magnitude, M_w, of 5.0 for 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 Units 3 & 4 site in order to make an inkind comparison of the GMRS. The STP 3 & 4 site is underlain by soils to depths of many thousands 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 – given here as the ratio of the STP 3 & 4 COLA soil GMRS [FSAR Table 2.5S.2-21] and the corresponding hard rock GMRS [derived from the hard rock UHRS values in FSAR Tables 2.5S.2-18 and -19] – 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 approximately 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-based site-specific GMRS.

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 used as close approximations to 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 Units 3 & 4 site GMRS points. These points are shown in Figure 6 along with the STP Units 3 & 4 COLA GMRS spectrum. The figure shows that the estimated CEUS SSC STP Units 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 6 is the STP 3 & 4 site-specific SSE. This design spectrum envelops both of the other spectra.

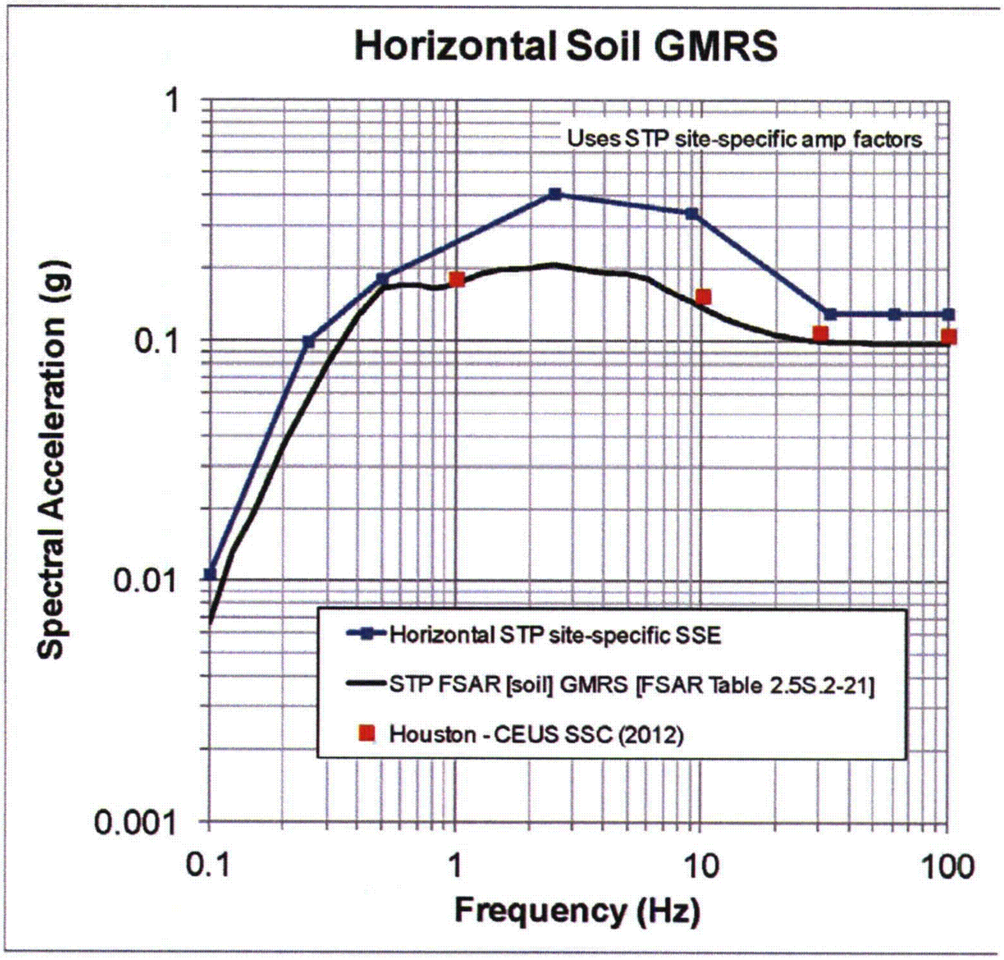

Figure 6. Comparison of the horizontal STP site-specific SSE, the STP 3 & 4 COLA GMRS (from FSAR Table 2.5S.2-21), and the 2012 Houston CEUS SSC GMRS.

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 6, 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.

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.

The text shown below will be included in FSAR Appendix 1E, Section 1E.2.2.1 in the next revision of the STP 3 & 4 COLA. (See Attachment 5).

Additionally, it was verified that both the existing STP 3 & 4 COLA results and the estimated CEUS SSC results for the STP sites are enveloped by the STP 3 & 4 SSE design spectrum.

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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 created a new FSAR Appendix (1E) to address the NRC's post-Fukushima recommendations (see Attachment 5 to this letter). This appendix was included in Revision 8 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)" in the next revision of the COLA.

> 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 DCD 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. Detailed design information regarding the enhancements discussed below will be submitted to the NRC 180 days prior to fuel load. The enhancements will be completed prior to fuel load.
>
> 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 quick and accessible 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~~ until offsite resource capability is reasonably assured. These power supplies will be stored in a Seismic Category I building and will be easily accessible for timely installation.

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 created a new FSAR appendix (1E) to address the NRC's post-Fukushima recommendations (see Attachment 5). This appendix was included in Revision 8 of the STP 3 & 4 COLA. The text in the RAI response below was 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, 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, Item 10 lists the ITAAC applicable to the Emergency Plan and its implementing procedures.

STP 3 & 4 makes the following commitment regarding this issue:

An assessment of staff and communications capabilities necessary to respond to a beyond design basis multi-unit event will be conducted in accordance with NEI 12-01, Revision 0. Any necessary enhancements to the Emergency Plan resulting from this assessment will be incorporated into the Emergency Plan implementing procedures which will be submitted to the NRC for review within 180 days prior to fuel load.

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 created a new FSAR appendix (1E) to address the NRC's post-Fukushima recommendations (see Attachment 5 to this letter). This appendix was included in Revision 8 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)" in the next revision to the COLA.

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. Operation of the ACIWA system is discussed in DCD 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 (FSAR Subsection 19.4.6). 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 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 FSAR 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 <u>Tier 1 NRC Recommendation/Responses</u>

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.

Additionally, it was verified that both the existing STP 3 & 4 COLA results and the estimated CEUS SSC results for the STP sites are enveloped by the STP 3 & 4 SSE design spectrum.

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

Conservatisms 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 breech 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:

- Overtoppping 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 FSAR 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 DCD and FSAR Appendix 1C (Table 1C-3). 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 DCD Subsection 19E.2.2.3..

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. (DCD Tier 1 Subsection 2.12.11 and DCD and FSAR. Tier 2 Appendix 1C).

- The batteries have a capacity of 8 hours (DCD Subsection 8.3.2.1.3.1). This capacity can be extended well beyond 8 hours if load shedding is performed.

- The Reactor Core Isolation Cooling system can provide core cooling for at least 8 hours during SBO conditions without reliance on AC power (DCD Appendix 19E.2.1.2.2).

- 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. Operation of the ACIWA system is described in DCD Subsection 5.4.7.1.1.10.

- 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 FSAR Section 13.4S and DCD and FSAR Section 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 in accordance with the SBO Rule will be developed during implementation of operational programs as described in FSAR Section ~~13.4S~~ 13.5.

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. Operation of the ACIWA system is discussed in DCD 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 (FSAR Subsection 19.4.6). 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 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 FSAR 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 DCD and FSAR 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 DCD 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 DCD 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. Detailed design information regarding the enhancements discussed below will be submitted to the NRC 180 days prior to fuel load. The enhancements will be completed prior to fuel load.

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 quick and accessible 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~~ until offsite resource capability is reasonably assured. These power supplies will be stored in a Seismic Category I building and will be easily accessible for timely installation.

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 (DCD and FSAR Sections 13.5 and Appendix 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 DCD and FSAR Section 13.2 and FSAR Section 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 (FSAR Subsections 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 (FSAR ~~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 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Rev. 0

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

New Commitment

Commitment ID	Commitment	Completion Date
12-27891-02	An assessment of staff and communications capabilities necessary to respond to a beyond design basis multi-unit event will be conducted in accordance with NEI 12-01, Revision 0. Any necessary enhancements to the Emergency Plan resulting from this assessment will be incorporated into the Emergency Plan implementing procedures which will be submitted to the NRC for review within 180 days prior to fuel load.	180 days prior to fuel load