Screening, Prioritization and Implementation Details (SPID) Guide for the Primary Approach – Fukushima NTTF 2.1:

Reference Number	Position	50.54 (f) letter Figure 1 Reference
1	Use of updated EPRI attenuation model	1
2	Use of existing site conditions	1
3	Use of existing structural models	6a, 6b
4	Scaling of responses to develop ISRS	6, 6a
5	Screening criteria for SSCs	6a, 6b
6	Use of IPEEE HCLPF to compare GMRS for screening	3, 5
7	Treatment of HF	3, 6a, 6b
8	Use of CDFM and separation on variables methods	6a
9	Approach for SFP evaluations	7a, 7b
10	Overall approach relative to RG 1.200 and ANS/ASME EE standard	ALL
11	Consideration of rock founded structures for developing ISRS	6a, 6b
12	Use of IPEEE hazard to compare to new hazard for screening	3, 5

Topic #1 / Figure #1 Step 1: Review and, if necessary, update EPRI (2004, 2006) attenuation model

Position:

Industry will review, and if necessary, update the EPRI (2004, 2006) attenuation model before calculating the ground motion response spectra at existing nuclear power plant sites.

Justification:

Based on the recent progress with the NGA East project, there is an opportunity to review and update the GMPE model. Based on discussions with ground motion experts beginning in October 2011, it is evident that there are now new / relevant data model and methods available that warrant a review and possible update to the EPRI 2004/2006 attenuation model. It has been ten (10) years since the EPRI (2004) SSHAC Level 3 workshops were held in 2002 and eight (8) years since the sigma component of the model assessment was updated beginning in 2005.

Follow-up studies under consideration:

EPRI to complete the 2004/2006 Attenuation Model review and, if necessary, update.

Topic #2 / Figure #1 Step 1 Bullet 4: Use of Existing site conditions

Position:

1. The industry will use available soil/site characteristic information. Utilities have the option to gather further information should they choose.
2. Subsurface site response models will go deep enough to characterize the lowest frequency of interest to the structure.
3. 30 convolution analyses will be used to define the mean and standard deviation of the site response.

Justification:

1. For the purpose of screening, the existing soil/site characteristic information will provide a sufficient accuracy.
2. Subsurface site response models are not needed down to depths that would be necessary to capture responses below frequencies of interest to nuclear facilities.
3. Statistical analyses have been done to demonstrate that 30 convolution analyses are sufficient to define the mean and standard deviation of the site response. The technical basis for this is contained in the February 22, 2010 Duke Power letter on the Lee Plant FIRS to the Document Control Desk of the NRC (Docket 05200018, Duke Power letter WLG2010.02-01, ADAMS ML100550350).

Follow-up studies under consideration:

- Industry to provide to NRC explanation of how sites with limited data will be handled within the EPRI seismic hazard study.
- Industry to provide guidance to utilities on nature of site response requested information required to make maximum use of existing information (being done currently).
- Industry to develop guidance on control point & soil/rock modeling for SSI (in layer, outcrop, etc.) in order to be able to ensure proper locations for the hazard are requested.

Topic #3 / Figure #1 Step 6 and 6a: Use of existing structural models (structural models from design basis or IPEEE studies) for any new recommendation 2.1 seismic response analysis

Position:

Existing structural models can be used in any seismic structural analysis generated to support SPRAs or SMAs.

Justification:

Those models have been used to develop seismic responses for the design, licensing and qualification of plant SSCs. Existing models are reasonably complex; usually have two or three dimensions and multiple frequencies and mode shapes. Typically, if a model complexity is increased, the contribution of the modes within the simpler model is decreased as modal mass is shifted to other modes, resulting in lower spectral peaks for the significant structural frequencies.

This effect has been reasonably shown through an analysis of the North Anna containment structure subjected to the August 23rd Mineral, VA earthquake.

Using the existing structural models will facilitate the timely completion of the SPRA/SMA effort with sufficient accuracy.

Follow-up studies under consideration:

- A series of studies are being considered to be able to further substantiate the positions related to structure modeling:
  - complex (asymmetric) structure vs more simple symmetric structures
  - soil site response vs rock site response
- A survey is also being considered to assess if relevant existing response data exists between FEM and stick models.

Topic # 4 / Figure #1 Step 6 and 6a:

Scaling of in-structure response spectra (ISRS) based on previous analyses

Position:

Scaling can be used in developing ISRS for those cases where the new uniform hazard spectrum (UHS) shape is approximately similar to the spectral shape previously used to generate the ISRS. For soil sites, the scaling is appropriate as long as the previous analysis correctly considered the SSI effects. Scaling will be based on:

- previously developed ISRS
- shapes of the previous UHS/review level earthquake (RLE)
- shapes of the new UHS, and
- structural natural frequencies, mode shapes and participation factors

Justification:

Scaling of ISRS is considered a technically sound approach and has been used in previous SPRAs. Scaling was recently used in a Surry pilot SPRA. Guidance on scaling is provided in industry documents such as EPRI report NP-6041-SL Rev. 1 and EPRI report 103959.

Scaling will reduce the effort involved in performing detailed soil structure interaction (SSI) analyses for the new hazard/UHS, facilitating the timely completion of the SPRA effort for those plants that are screened-in.

Scaling of rock or soil sites where the shape of the new hazard spectrum is not similar to the previous spectrum is not recommended without detailed analysis that demonstrates the validity of the scaling approach.

Follow-up studies under consideration:

Potential follow on studies considered include:

- Industry to provide examples of the methods and processes used for scaling that has been documented in the past.
- The definition of *approximate similarity* of shape would be clarified with a case study to help define that boundary. Vogtle 1 & 2 spectra might represent a good case study for this issue since the Vogtle shape is in the grey area regarding degree of similarity.

Topic #5 / Figure #1 Step 6 and 6a:

Develop and use criteria to screen structures, systems and components (SSCs) that will be included in the SPRA/SMA systems analysis models (i.e., plant logic model).

Position:

The screening criteria will be developed with sound engineering basis to ensure proper focus is given to those SSCs that are risk significant. The following screening approaches can be used:

1. SSCs with high seismic capacity can be expected to be small contributors to the SCDF and can be excluded from the systems logic model. SSCs with a mean point estimate probability of failure less than 1E-7 can be considered to constitute a small contributor to SCDF. Capacity estimates of certain types of rugged components (e.g. piping) can be based on guidance from EPRI NP-6041-SL Rev. 1 to develop the point estimate frequency of failure. Therefore, detailed fragility or High-Confidence-of-Low-Probability-of-Failure (HCLPF) calculations are not needed for these inherently rugged components and these SSCs do not need to be reflected in the model.

2. Rank SSCs using PRA insights to prioritize the fragility calculations of the SSCs that are not screened out using other screening methods. Screening or ranking of SSCs from a preliminary SPRA plant logic model can be done by performing a parametric sensitivity analyses with assumed initial fragilities and ranges of fragility values. Those SSCs that do not contribute significantly to the SCDF of an accident sequence (e.g. about 1E-7, but this can somewhat vary for plants) may be screened out from having to perform detailed fragility calculations or removed from the SPRA model altogether. Another option under discussion would be to use PRA importance data from the internal events PRA to rank the list of SSCs that do not screen out from the other screening methods. The internal events PRA importance measures cannot be used directly, since only certain sequences (such as LOSP) are major contributors to seismic risk, and seismic correlation must be considered. A seismic-specific importance analysis must be used for this approach.

The above screening methods will ensure that the time and effort to develop detailed fragilities is focused on the most risk important SSCs.

Justification:

1. A frequency of failure of 1E-7 would correspond to a contribution on the order of 1% to the SCDF for most plants. A plant specific criterion against which the HCLPF can be compared can be developed from the 1E-7 criterion using a mean point estimate approach, an assumed composite variability, and the site hazard. The calculated HCLPF criterion can be used in reviewing previous IPEEE, A-46 or design basis calculations to determine and judge if explicit fragility/HCLPF calculations are needed. Note that other screening criteria can be justified when the success terms of lower capacity SSCs are considered. Plant-specific criteria and justification would be documented for the screening.

2. Those SSCs that do not affect the sequence quantification can be modeled sufficiently with an assumed fragility, or eliminated from the models, without significant impact on the overall SCDF.

Screening methods have been used in past SPRAs and SMAs. Screening of SSCs helps focus the scope of the fragility/HCLPF calculations. It is expected that the above screening methods could substantially reduce the scope of the fragility or margin calculations required in the SPRA or SMA, and still meet the objective of identifying and ranking safety-significant SSCs.

Follow-up studies under consideration:

Industry development of examples of the screening methods and the process for the above positions.

Topic #6 / Figure #1 3, 5: Use of IPEEE HCLPF spectrum as an alternative to the SSE to compare to GMRS for screening/prioritizing plants

Position:

For plants that conducted an SPRA, focused scope SMA or full scope SMA during the IPEEE, the screening/prioritizing will be done by comparing either the IPEEE HCLPF spectrum or the plant SSE to the new GMRS. If the IPEEE HCLPF is used for screening, the IPEEE will be updated to represent the current plant configuration. If there were deficiencies in the quality of the IPEEE conducted, these deficiencies shall be resolved. Additionally, for plants that conducted focused scope SMAs, their IPEEEs will be updated to meet the requirements of NUREG-1407 for full scope SMA.

Additional screening criteria for reduced scope plants are recommended with the higher of the two levels below proposed to be compared to the GMRS:

- NUREG/CR 0098 spectra anchored to 0.15 g
- 1.2 times the SSE

The basis for this recommendation is based on a minimum level of margin existing beyond the SSE for all the plants based on results from past SPRAs and SMAs.

Justification:

The IPEEE established levels of plant seismic safety that should be incorporated into the screening process. The industry supports NRC recommendation offered during the March 1st public meeting.

Follow-up studies under consideration:

- Characterize the upgrades for focused scope plants into order to meet the full scope criteria.
- Describe the process to generate the HCPLF spectrum for plants that performed SPRA during IPEEE
- Upgrades to IPEEE required in order to establish pedigree/quality levels required in order to justify the screening

<u>Topic #7 / Figure #1: 3, 6, 6a:</u> Treatment of High Frequency Response and High Frequency Capacity

<u>Position:</u>

Plants with GMRS>SSE above 10 Hz will provide confirmation that SSCs, which may be affected by high-frequency ground motion, will maintain their functions important to safety. In addition, plants where GMRS > SSE only above 10 hz need not perform dynamic analyses of structures to develop ISRS.

<u>Justification:</u>

The majority of nuclear plant SSCs that have been seismically qualified are not sensitive to high frequency seismic input. This is because the displacements are much higher for low frequency seismic motion, as opposed to high frequency seismic motions. Past shake table testing has demonstrated that SSCs that fail in a ductile manner are not appreciably affected by high frequency seismic loading. Brittle failure modes and functional failure modes have been shown to be the only potential failure modes affected by high frequency loads.

The test program will use accelerations or spectral levels that are sufficiently high to address anticipated responses of various plants. Therefore, it would not be necessary for those few plants where GMRS > SSS only above 10 hz to perform structural analyses.

<u>Follow-up studies under consideration:</u>

Several follow up studies are recommended*:*

- Survey and consolidate past studies on high frequency effects (2 past EPRI studies and a white paper, along with other papers on the topic) to produce a white paper identifying SSCs with high frequency sensitivities (primarily relays/switches, but could include limited additional items)

- Initiate a small group of experts to develop focused testing program on high frequency sensitive components. Conduct testing on sample of components and extrapolate results as appropriate.

- Develop methods to incorporate high frequency test results into fragility/margins programs

- Plants affected by HF will provide, as input to the HF testing program:
    - Types and make / models of plant unique high frequency sensitive components
    - Typical Levels (either g values or response spectra) expected for testing of HF sensitive components

Topic #8 / Figure #1 6a: Use of either CDFM (hybrid approach) or the separation of variable methods to develop fragilities

Position:

The hybrid / CDFM approach is an acceptable method for generating fragilities within a SPRA.

Use the CDFM approach, as much as possible, to calculate fragilities for seismic PRAs. The CDFM method will determine a HCLPF value. Using an assumed composite variability (e.g. in the range of 0.35 - 0.45) the median value can be calculated to define the fragility parameters. The fragility parameters are then used in the systems model to convolve with the hazard. One disadvantage is that the assumed composite variability may give conservative estimates of seismic CDF. For those SSC that are determined to be the dominant risk contributors or are risk-significant in the seismic accident sequences, better estimates of Am and Beta(u) and Beta(r) can be done using the fragility approach and then used in the integration.

Justification:

CDFM is a simpler method for the majority of engineers to learn and apply, as compared to the fragility method.

Follow-up studies under consideration:

Based on the NRC comments during the March 1-2 meetings, the following follow up studies are being considered:

1. Generate argument for individual HCLPF values calculated by the two approaches being similar based on past studies
2. Comparison of CDFM and separation of variables fragility results for individual or multiple sequences

<u>Topic #9 / Figure #1 Step 7a and 7:</u> Approach for spent fuel pool (SFP) evaluations

<u>Position:</u>

Only those failure modes associated with the reinforced concrete SFP enclosure will be evaluated.

<u>Justification:</u>

By design, a failure of the systems connected to the SFP cannot result in a significant loss of inventory of the SFP. In addition, there are redundant and diverse means to provide make up water to the SFP for losses of inventory. The National Research Council conducted a study (*Safety and Security of Commercial Spent Fuel Storage - Public Report*, 2006) on spent fuel pool safety and concluded that spray systems such as those available in response to B.5.b (or additional supplemental capability using FLEX equipment) at all US nuclear plants provide adequate cooling to prevent zirconium cladding fires, as documented below:

"*Preinstalled water spray systems above or within the pool could also be used to cool the fuel in a loss-of pool coolant event. The committee carried out a simple aggregate calculation suggesting that a water spray of about 50 to 60 gallons (about 190 to 225 liters) per minute for the whole pool would likely be adequate to prevent a zirconium cladding fire in a loss-of-pool-coolant event. A simple, low-pressure spray distribution experiment could verify what distribution of coolant would be sufficient to cool a spent fuel pool.*"

Seismically induced internal flooding (inventory loss from mechanisms such as sloshing) are beyond the scope of NTTF Recommendation 2.1 and in included in NTTF Recommendation 3 and therefore should be excluded from the scope of this evaluation.

In addition, EPRI is leading a study to develop a consensus approach to assessing the risk associated with the SFP inventory and cooling loss from a spectrum of causes.

<u>Follow-Up Studies:</u>

- Verify the design of all SFPs to confirm they are fail safe.
- Provide description of potential methods to supplement inventory loss from the SFP

Topic #10 / Figure #1 Steps 6 and 6a:

Invoke requirements from the ASME/ANS PRA Standard and R.G. 1.200 that are consistent with the nature of this application in performing a SPRA or SMA.

Position:

For this application, the requirements corresponding to Capability Category I of the AMSE/ANS external events standard will generally be applied in the performance of elements of a SPRA or SMA. In limited cases, exceptions to the Standard requirements may be taken. The intent of the standard will be met.

Justification:

The PRA Standard is intended to identify the degree of detail and plant specificity in a risk assessment that reflects the nature of the application for which the risk assessment is being used. The application in this case is to gain an updated understanding of the risk of seismic events at nuclear power plants in light of new information about seismic hazard. This includes developing a new or changed understanding of risk outliers due to seismic events.

Per Table 1-1.3-2, Capability Category I is appropriate for applications with the following characteristics:

1. **Scope and level of detail**. Resolution and specificity sufficient to identify the relative importance of the contributors at the system or train level (and for Fire PRA, at a fire area level), including associated human actions.
2. **Plant-specificity**. Use of generic data/models acceptable except for the need to account for the unique design and operational features of the plant.
3. **Realism**. Departures from realism will have moderate impact on the conclusions and risk insights as supported by good practice.

Thus, for this application, a study with elements that generally satisfy the requirements in Capability Category I is adequate. To develop an improved understanding of the impact of new hazard estimates, screening approaches are approached to limit the scope of detailed analyses. Where more detailed analyses are essential to achieve an adequate level of understanding (e.g., with respect to “realism”), these analyses will be performed or alternative measures will be taken (such as making plant changes to address the impacts).

Applying the approach that has been specified for the seismic hazard is expected to satisfy Capability Category II in most respects. In the case of the fragility analysis, a mixture of plant-specific and generic information will be employed. The plant model will be plant-specific.

Follow-up studies under consideration:

A review of the Supporting Requirements will be performed to confirm the judgment that Capability Category I is appropriate, or to identify where departures are acceptable (i.e., when Capability Category I does not need to be met), or Capability Category II should be applied.

Topic #11 / Figure #1 6, 6a: Consideration of rock founded structures for developing ISRS

Position:

The original definition of rock (> 3,500 ft/sec) can be used for the development of the ISRS. Fixed based models can be used in the dynamic analyses of rock-founded structures using this original definition of rock.

Justification:

Past experience has shown that the amplified response spectra in the 1-10 Hz are approximately the same from a fixed based model vs. a model that uses soil-structure interaction (SSI) analysis. Therefore, it is conservative to use fixed base dynamic analyses for rock-founded structures even when the shear wave velocities are not as high as 9200 ft/sec.

Follow-up studies under consideration:

An example of the Containment Structure at a plant is being analyzed to demonstrate the validity of this position. The initial results of this example indicate that there is almost no impact on the ISRS spectral ordinates in the 1 to 10 Hz range whether the structure is analyzed as a fixed-base or with SSI.

Topic #12 / Figure #1, steps 3, 5: Use of IPEEE hazard to compare to new hazard for screening of plants

Position:

In the screening process to determine whether a plant needs to perform a SPRA or a margin analysis, the GMRS/SSE comparison will not be the only consideration. Plants can be screened out (no SMA or SPRA required) if the mean estimate of seismic CDF decreases for the new hazard curve compared to the hazard curves used during the IPEEE (EPRI-1989 and LLNL-1992).

Justification:

It is possible that for some plants, the GMRS may exceed the SSE marginally at certain discrete frequencies in the 1 to 10 hz range, however, a point mean estimate of the CDF, based on the IPEEE plant HCLPF (or CDF) with new hazard curve vs. the previous IPEEE hazard curves (EPRI-1989, LLNL-1992) may show that the delta CDF is negative. This would imply that the seismic risk at the plant has decreased. Therefore, that plant does not need to conduct a new SPRA on SMA and can be screened out. It is noted that the 50.54(f) letter dated March 12, 2012 (Recommendation 2.1 Seismic, p.6, third paragraph) alludes to the use of hazard curves for screening and prioritization. Also, NRC used this type of approach in their safety/risk assessment of GI-199, as documented in Information Notice 2010-18.

Follow-up studies under consideration:

Industry could perform point estimates of an example plant with USGS based hazard curves to show that it is possible that even though the GMRS marginally exceeds the SSE at certain frequencies in the 1 to 10 Hz. range, the delta CDF from the point estimates is negative, i.e., the seismic risk at this plant has decreased.