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Light Water Reactor Sustainability Program

Demonstration of NonLinear Seismic Soil Structure Interaction and Applicability to New System Fragility Seismic Curves

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EXECUTIVE SUMMARY

Risk calculations should focus on providing best estimate results, and associated insights, for evaluation and decision-making. Specifically, seismic probabilistic risk assessments (SPRAs) are intended to provide best estimates of the various combinations of structural and equipment failures that can lead to a seismic induced core damage event. However, in general this approach has been conservative, and potentially masks other important events (for instance, it was not the seismic motions that caused the Fukushima core melt events, but the tsunami ingress into the facility).

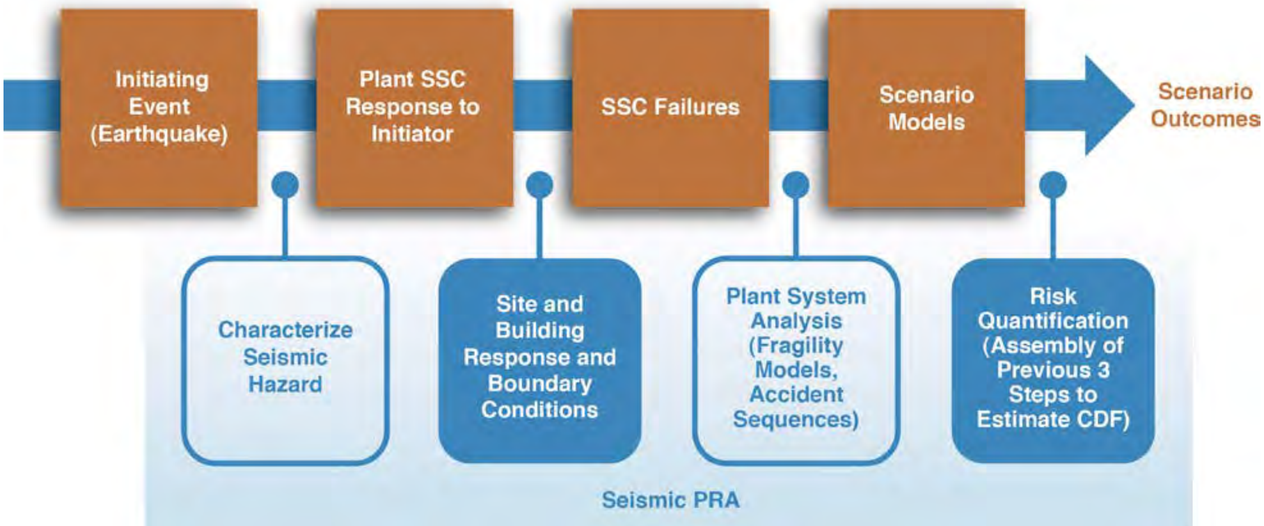


Figure 1: Seismic Risk Quantification Process

SPRAs are performed by convolving the seismic hazard (the frequency of certain magnitude events) with the seismic fragility (the conditional probability of failure of a structure, system, or component given the occurrence of earthquake ground motion). In this calculation, there are three main pieces to seismic risk quantification, 1) seismic hazard and nuclear power plants (NPPs) response to the hazard, fragility or capacity of structures, systems and components (SSC), and systems analysis. Figure 1 provides a high level overview of the risk quantification process.

The focus of this research is on understanding and removing conservatism (when possible) in the quantification of seismic risk at NPPs.

Risk analysts performing traditional SPRA calculations for nuclear power plants make assumptions that may cause risk results to be conservative. This research will provide methods, tools, and data that can be used to remove or minimize these conservatisms. Some of these conservative assumptions are:

- **NPP response scales linearly with ground motion**. This assumption effects the "Plant SSC Response to Initiator" box in Figure 1 where we see that as the earthquake intensity increases, the NPP response increases in a non-linear fashion
    - R&D solution: Determine the degree of reasonableness of this assumption by evaluating the response of a generic NPP to increasing levels of ground motion and tracking NPP response. The results documented in Section 5 indicate that even at the INL, which is a low to moderate seismic site, the assumption is likely conservative. (FY 2014)
    - R&D solution: Calculate the seismic core damage frequency (SCDF) using a traditional SPRA approach that uses linear seismic analysis with an advanced SPRA approach that uses NLSSI. This is the focus of the FY 2014 task that will be complete in December 2014. (FY 2014 & 2015)

- **Uniform Hazard Spectrum (UHS) is used to define ground motions for determining NPP in-structure response**. The UHS is conservative since it is an envelope of all ground motions (This is the “characterize seismic hazard” box in Figure 1).
    - R&D Solution: Evaluate the change in SCDF when using ground motions that a produced from a conditional mean spectrums (CMS), which is more representative of actual earthquakes (not an envelope of all earthquakes for a given site). (FY 2015 & 2016)
- **The use of parameters to define SSC fragility curves that are weakly correlated to the damage of that SSC**. For instance peak ground acceleration (PGA) is typically used to define failure of an SSC when this parameter is weakly correlated to damage of SSC’s (This is the “SSC Failures” box in Figure 1).
    - R&D Solution: Study the impact on system and component probability of failure when using current practice with fragility curves that base probability of failure on PGA versus fragility curves that have a strong correlation to failure of that system or component such as differential displacement between components in the facility. (FY 2015 & FY 2016)

Long term, the capability to perform advanced SPRA evaluations will be developed in the MOOSE framework. The numerical tool development work will start in FY 2015.

When looked at individually, any one conservative assumption or model by itself may not have a large impact on seismic risk. However when taken collectively, they may magnify each other and the overall results may be unduly conservative.

CONTENTS

ACRONYMS

DBE	Design Basis Earthquake
DOE	Department of Energy
INL	Idaho National Laboratory
NLSSI	NonLinear Soil Structure Interaction
NNSA	National Nuclear Security Administration
NRC	Nuclear Regulatory Commission
SASSI	System for Analysis of Soil Structure Interaction
SCDF	Seismic Core Damage Frequency
SSI	Soil-Structure Interaction

1. Introduction

Seismic Probabilistic Risk Assessments (SPRAs) should be performed so that the calculated risk numbers provide best estimates, removing conservatism as much as possible, to enable NPP owners to make decisions on where to reduce risk. The concern is that being conservative when performing the SPRAs will mask other potential sources of risk and focus disproportionate time and money on mitigating seismic risk. Traditional SPRA approaches may have conservatisms built into them that need to be removed. Therefore this research and development (R&D) is focused on providing advanced seismic SPRA methods, tools, and data with the goal of removing conservatism to the extent possible and to provide "best estimate" seismic risk numbers.

Traditional SPRAs at nuclear power plants make assumptions that may cause risk numbers to be conservative. Some of these assumptions are:

1. NPP response scales linearly with ground motion (this is the focus of the FY 2014 task that will be complete in December 2014)
2. Uniform Hazard Spectrum (UHS) is used to define ground motions for determining NPP in-structure response. The UHS is conservative since it is an envelope of all ground motions. More realistic ground motions can be produced using conditional mean spectrums (CMS).
3. The use of parameters to define SSC fragility curves that are poorly correlated to the damage of that SSC. For instance peak ground acceleration (PGA) is typically used to define failure of an SSC when this parameter is poorly correlated to damage of SSC's.

Future efforts will be focused on evaluating the conservatisms introduced into the SPRA process from items 2 and 3. The FY 2014 task will evaluate the conservatism introduced using the assumption listed in 1 and its effect on NPP SCDF. This will be accomplished by comparing the SCDF of a traditional SPRA with the SCDF of an advanced SPRA.

This report provides the following information:

- Brief background on SPRAs
- Vision for future implementation of advanced SPRAs
- Comparison of linear SSI analysis with NLSSI analysis using a generic NPP for increasing levels of ground motion to demonstrate the importance of including gapping and sliding and the discusses likely impact this will have on probabilities of failure
- Presents a detailed project plan for completing the generic advanced SPRA study started in FY 2014. This task compares a traditional SPRA with an advanced SPRA (this meets short to medium term goals),
- Discusses the importance of gathering beyond design basis data at existing NPP's and performing a demonstration to highlight areas to improve the SPRA methodology.
- Presents a long-term plan for developing tools using the MOOSE framework so that virtual reactors can be tested using virtual external scenarios to determine vulnerabilities (long term goals).

2. Background

SPRA includes the following technical elements, a) probabilistic seismic hazard analysis, b) seismic fragility evaluation, and c) seismic plant response analysis. SPRA involves the integration of plant fragility data over seismic a hazard curve and requires full consideration of uncertainty in seismic hazard, structural response and properties and capacities of nuclear power plant (NPP) components. Seismic PRA results are used to determine the various combinations of structural and equipment failures that can lead to a seismic induced core damage event, and the integration of these results to quantify the risk.

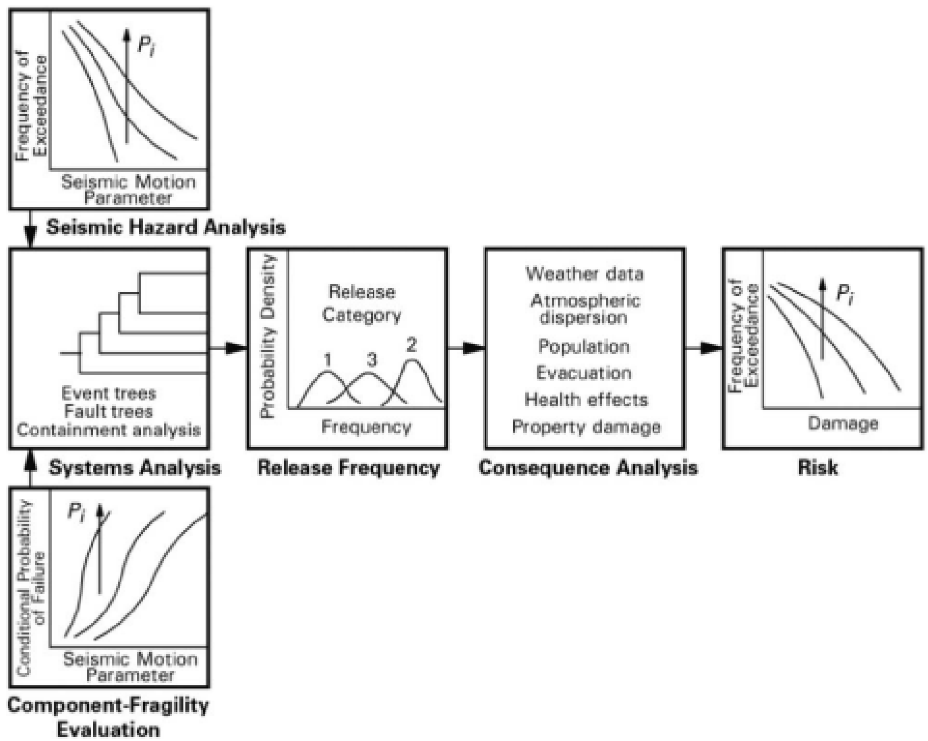


Figure 1: Schematic Overview of a Seismic PRA (ASME/ANS RA-Sa-2009)

SPRA determines the annual frequency of unacceptable performance, such as core melt and large release of radiation. Guidance for performing SPRA is provided in NUREG/CR-2300. The nuclear industry typically uses peak ground acceleration (PGA) in the SPRA calculations, which is now known to be a poor indicator of damage to the structure and equipment. The use of PGA to characterize fragilities of the structure introduces substantial and not yet quantified uncertainties into the risk calculations. Inclusion of seismic soil-structure interaction (SSI) effects into the SPRA process typically implements frequency domain software such as SASSI or CLASSI.

Both SASSI and CLASSI are equivalent linear software commonly used by the nuclear industry for evaluating NPP facility response to seismic ground motions. This software approach relies on the assumption (or approximation) of linear behavior both in the soils and the structures. The assumption of linear behavior is incorrect for larger magnitude earthquake events that cause nonlinear behavior such as:

- Material nonlinearity in both the soil and structure (concrete cracking). This material nonlinearity produces degradation in material stiffness for increasing levels of ground motion and an increase in energy dissipation. This is known as stiffness proportional energy dissipation.

- Geometric nonlinearities such as gapping and sliding. Energy is dissipated due to gapping and sliding between the soil and the structure.

Huang et al. (2011a, 2011b) developed a seismic PRA procedure for *surface-mounted or shallowly embedded* NPPs on rock sites using modern concepts of earthquake ground motion, response-history analysis, component fragilities, and Monte Carlo simulation. The goal was to improve estimates of seismic risk by substantially reducing the uncertainties associated with the ground motion representation, demands on components and fragility specifications. The procedure can be employed for intensity-, scenario-, and time-based assessments. Fragilities of SSCs are expressed in terms of meaningful demand parameters such as floor spectral acceleration, peak floor velocity, and story (framing) drift angle. The approaches used in this procedure to calculate seismic demands and to determine the probability of unacceptable performance at a given intensity of ground motion are fundamentally different from those used to date for assessment of nuclear structures. However, this methodology did not include nonlinear soil structure interaction (NLSSI) effects. NLSSI effects are important for realistically capturing NPP response during increasing levels of ground motion. The study by Huang et al. can be extended by implementing NLSSI effects and benchmarked by gathering and using existing data from NPPs that experienced beyond design basis earthquakes (BDBE) for comparison. Therefore, it is necessary to:

- Develop advanced SPRA methods
- Develop NLSSI tools
- Gather and use BDBE data from NPPs to verify and validate numerical tools

3. Need for Advanced SPRA Methods and Tools

The nuclear industry is currently addressing the Near Term Task Force (NTTF) recommendations. One specific recommendation that they are dealing with is recommendation 2.1 which states,

> Order licensees to reevaluate the seismic and flooding hazards at their sites against *current NRC requirements and guidance, and if necessary, update the design basis and SSCs important to safety to protect against the updated hazards."*

In response to this EPRI has developed a document titled "Seismic Evaluation Guidance Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," to provide guidance for conducting seismic evaluations.

On February 15, 2013 the NRC provided its endorsement of the EPRI-1025287 document. This document is intended to provide a process NPP owners can follow to meet the NTTF recommendation 2.1. The document includes a screening process that evaluates updated site-specific seismic hazard (based on the CEUS-SSC study). The screening process includes the following steps:

1. NPPs develop new site specific hazard curves based on CEUS;
2. Utilize screening process to eliminate certain plants from further review;
3. Perform SPRA or SMA for NPPs;
4. Submit proposed actions to evaluate seismic risk contributions (update seismic analysis where necessary).

The updated site-specific hazard curves (item number 1) have potential for higher magnitude, higher frequency content accelerations. This would cause site-specific seismic PRA parameters such as peak ground acceleration (PGA) to increase. The increase creates a potential for the core damage frequency numbers in these SPRA's (item number 3) to increase beyond what NRC sets as an acceptable limit. These exceedances create an opportunity to implement more advanced SPRA approaches that include NLSSI to remove conservatism from the numerical models.

Another NTTF recommendation that will have a continual impact on the seismic risk at nuclear facilities is recommendation 2.2. This recommendation requires that every 10 years NPPs address any new and significant information related to the seismic hazard:

> Initiate rulemaking to require licensees to confirm seismic hazards and flooding hazards every 10 years and address any new and significant information. If necessary, update the design basis for SSCs important to safety to protect against the updated hazards.

SPRAs are intended to provide best estimates of the various combinations of structural and equipment failures that can lead to a seismic induced core damage event, and the integration of these results to quantify the risk. The advanced SPRA tools and methods outlined in this report propose to increase the fidelity of the SPRA methodology by using high fidelity modeling and simulation tools to provide more realistic seismic facility calculations for given earthquake events. This approach will provide more realistic seismic accelerations at locations in the nuclear facility where critical components are located.

NPP owners who are resolving NTTF recommendation 2.1 may find that the traditional "conservative " approach to SPRA produces core damage frequency numbers that are above the NRC allowable limit.

The proposed advanced SPRA methodology and tools necessary to implement this methodology would provide the industry with more realistic SPRA's and may allow NPP owners to meet NTTF recommendation 2.1. Additionally these advanced tools and methods would provide for longer term industry needs as they "confirm" their seismic hazards every 10 years.

Actual earthquake recordings indicate that the seismic hazard is not well quantified. Three recent earthquakes in the last seven years have exceeded their design basis earthquake values (so damage to SSC's should have occurred) as shown in Figure 3.1. These seismic events were recorded at North Anna (August 2011, detailed information provided in [*Virginia Electric and Power Company Memo*]), Fukushima Daichii and Daini (March 2011 [TEPCO 1]), and Kaswazaki-Kariwa (2007, [TEPCO 2]). However, seismic walk downs at some of these plants indicate that very little damage occurred to safety class systems and components due to the seismic motion. There is an opportunity to gather the free field soil and in-structure acceleration time history data, process the data, try to gather existing analyses and SPRAs (might only be available for North Anna) and quantify margins built into the analyses. This process would highlight margins (areas of conservatism) in existing design and SPRA approaches. These conservatisms should be removed from SPRAs.

	KK 2007	Fukushima 2011	North Anna 2011
Design Value (g)	0.20	0.26 (Original) 0.45 (Update)	0.18
Recorded Value (g)	0.32	0.56	0.26

Figure 3.1: Peak recorded acceleration verses the design acceleration value.

4. Long Term Vision and Implementation

Laying out the short, medium, and long-term goals of this overarching activity are important to understanding current activities. Figure 2 provides a general flow towards more advanced methods and tools.

Short to Medium Term Goal

Provide DOE and Industry with robust analytical methods to evaluate larger seismic ground motions at critical infrastructure and nuclear facilities and implement protective measures such as seismic isolation (SI). The goal is to minimize seismic risk at nuclear facilities through cost effective analytical approaches and technologies

Long Term Goal

Development of advanced methods and realistic tools to evaluate the performance of virtual nuclear power plants and nuclear facilities to a wide range of external hazards including multiple event scenarios. Allows nuclear facility owners to virtually test external hazards before the actual facilities are tested with actual hazards. Allows owners to anticipate potential issues and resolve them

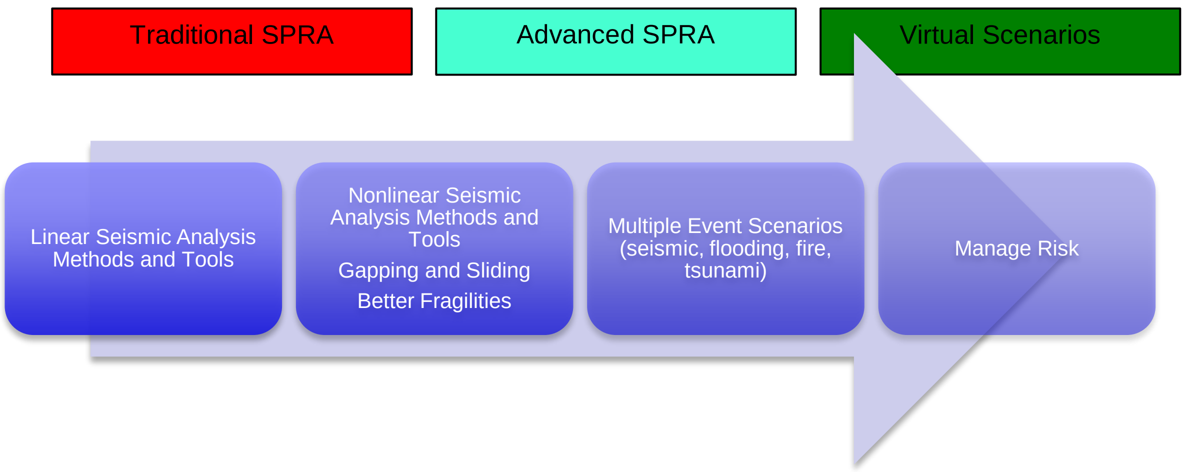


Figure 2: Evolution to long term goals

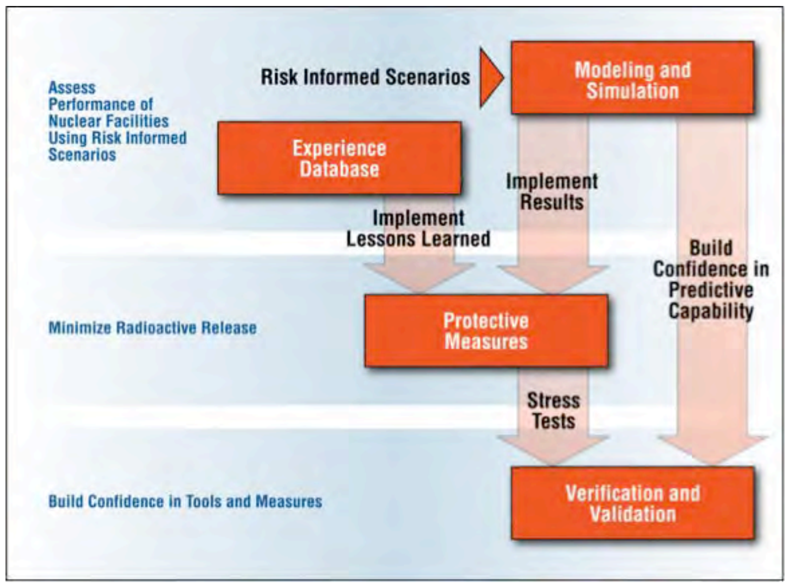


Figure 3: Future risk informed process to minimize radioactive releases to acceptable levels and manage risk

Initial R&D activities associated with advanced SPRA methodology development focuses on demonstrating the potential for NLSSI to reduce in-structure responses using more realistic models. This initial demonstrations project will use a safety-class system and compare SCDFs for traditional SPRAs to Advanced SPRAs (project plan provided in Appendix A). The next step will be working with EPRI and industry to get permission to use an actual NPP with actual safety-class systems placed on an actual NPP soil site to demonstrate the difference between traditional SPRA and advanced SPRA. In parallel with this activity will be development of soil-structure interaction analysis tools in MOOSE. Longer term these activities will be coupled with other MOOSE based capabilities so that virtual external events can be modeled using virtual reactors.

Future risk evaluations should follow a process similar to that shown in Figure 3. This process would start with risk informed external event scenarios such as seismic, flood, fire, tsunami, or a combination of these as initiating events. Verified and Validated (V&V) models would be used to simulate the external hazard initiators. Model results would be used to determine at risk systems and components and appropriate decisions made on what protective measures or mitigation could be needed. Of course implementation of experience data gathered from previous external hazard events at NPPs needs to be used in the decision making process.

Development of advanced SPRA V&Ved methodology and the necessary tools to perform the methodology is a multiyear effort. This multiyear effort includes three aspects:

1. development of a methodology for performing advanced SPRAs
2. development of tools to perform advanced SPRA
3. gathering data that will be used to V&V methods and tools.

Currently activities have been started in methodology and tool development activities.

Development of an appropriate methodology for evaluating risk has started with a demonstration activity that compares a traditional SPRA with an advanced SPRA.

5. Nonlinear Soil-Structure Interaction and Application to Advanced SPRA's

Currently the Department of Energy (DOE) and the nuclear industry perform seismic soil-structure interaction (SSI) analysis using equivalent linear numerical analysis tools. For lower levels of ground motion these tools produce reasonable in-structure response values for evaluation of existing and new facilities. For larger levels of ground motion, these tools may overestimate (and, in some instances, may underestimate) the in-structure response (and therefore structural demand) since they do not consider geometric nonlinearities (such as gaping and sliding between the soil and structure) and are limited in the ability to model nonlinear soil behavior. The current equivalent linear SSI (SASSI) analysis approach either joins the soil and structure together in both tension and compression, or releases the soil from the structure for both tension and compression, approximates material nonlinearities, and generalizes damping. For higher levels of ground motion, these approximations may produce higher in-structure responses. In traditional SPRAs the assumption is that in-structure response scales linearly with ground motion. This is a conservatism that can be removed by NLSSI.

Seismic hazard curves at nuclear facilities have generally continued to increase over the years as more information has been developed on seismic sources (i.e. faults), additional information gathered on seismic events, and additional research performed to updated attenuation equations and additional information on local site effects. Seismic hazard curves are convolved with seismic fragilities to quantify risk at NPPs. Therefore, when using traditional SPRAs, as seismic hazard curves increase at a NPP site so does the seismic risk since in-structure response scales linearly with ground motion (this assumes that the fragilities stay the same i.e. no facility modifications have occurred)..

As ground motions increase so does the importance of including nonlinear effects in numerical SSI models.

5.1 Nonlinear Soil-Structure Interaction

To include material nonlinearity in the soil and geometric nonlinearity using contact (gaping and sliding) it is necessary to develop a nonlinear time domain methodology. This methodology is in the process of development [Coleman et. al.]. NLSSI methodology opens the door to explore more realistic seismic behavior at nuclear facilities such as, gapping and sliding, inclined seismic waves coupled with gaping and sliding, nonlinear soil behavior, nonlinear structure behavior, and evaluate seismic isolation. NLSSI also provides a numerical approach used in advanced SPRA methodology.

Results from the NLSSI report [Coleman et. al.] show the change in in-structure response when gaping and sliding is included in the analysis. These curves show a reasonable match at low levels of ground motion as expected since at low levels of ground motion the coupled soil structure response is linear. The curves show increasing divergence at high levels of ground motion. Figure 5.1 shows the locations in the generic NPP where the results for Figures 5.2, and 5.3, were taken from. These plots show the maximum acceleration values on the response spectrum versus the applicable multiple of DBE (i.e. 0.5, 1,1.5, 2, 3). These figures clearly show a nonlinear effect that is mainly produced by the ability to model gaping and sliding between the soil and structure.

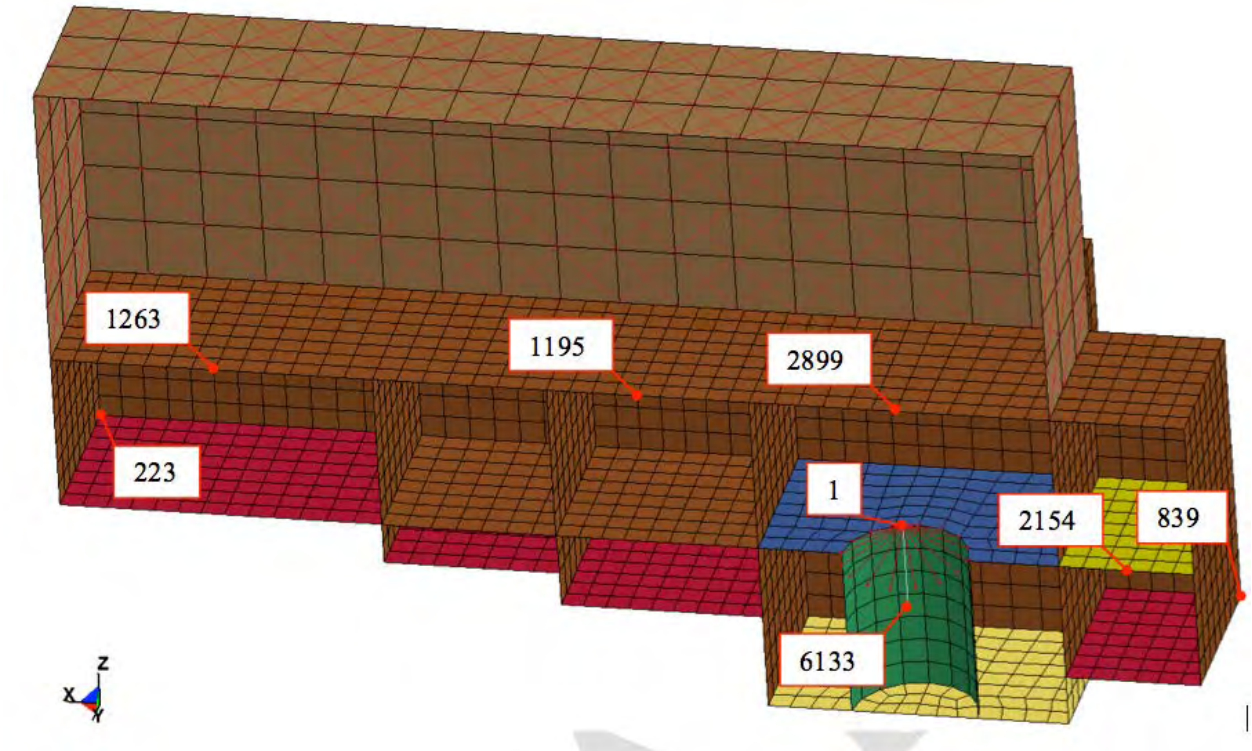


Figure 5.1: Structural model of generic NPP

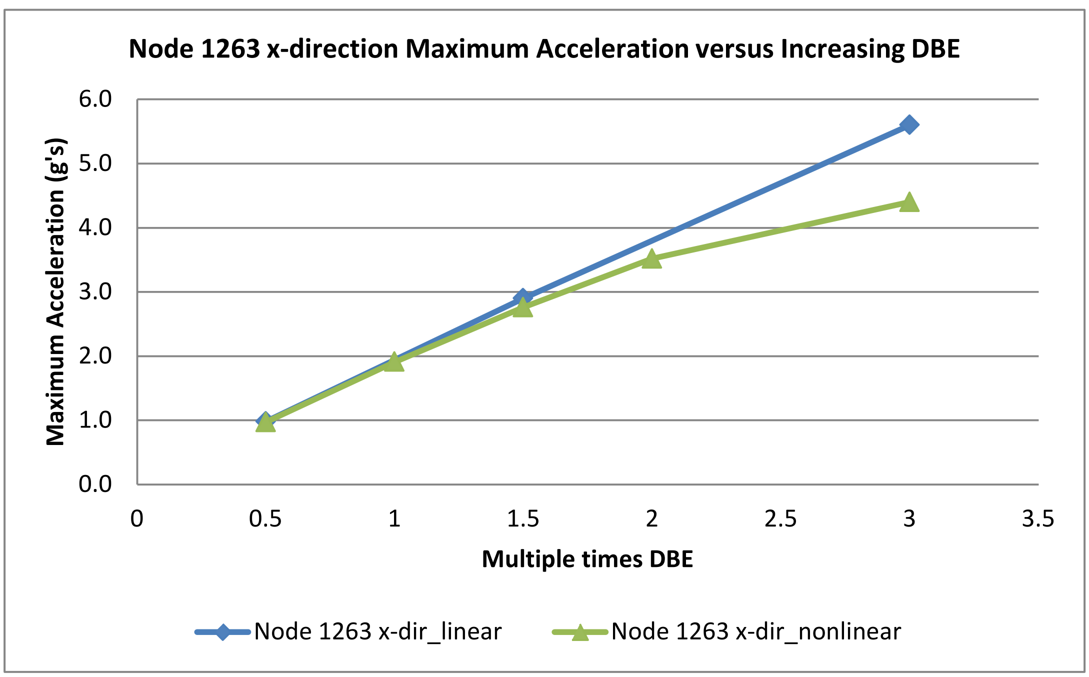


Figure 5.2: Maximum response spectrum acceleration at increasing levels of ground motion at INL site at in structure location, Node 1263

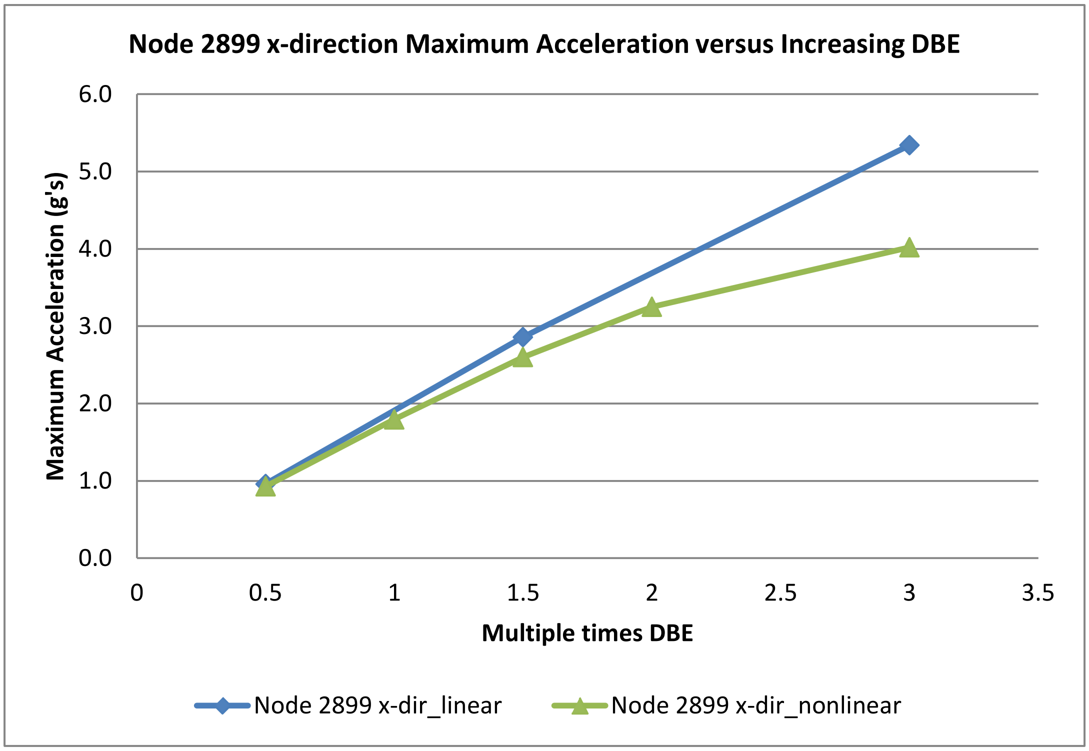


Figure 5.3: Maximum response spectrum acceleration at increasing levels of ground motion at INL site at in-structure location, Node 2899

The assumption of in-structure response scaling linearly with ground motion is not valid at higher ground motions

5.2 NLSSI Effect on SPRA's

If a safety class system or component was placed at locations shown in Figure 5.2 or 5.3 then using these nonlinear results would decrease probability of system failure when compared with the linear analysis. A project that compares a traditional SPRA with an advanced SPRA has been started to quantify the impact of using NLSSI on the seismic core damage frequency at a generic NPP (Section 6 provides more detail).

Comparison of linear analysis versus NLSSI documented in Figures 5.2 and 5.3 uses INL ground motion that was developed from the INL DBE (which is based on the mean hazard curve), which has a relatively mild hazard. Based on these plots consideration of nonlinearities of the generic NPP at the INL site start to become important at 1.5× to 2×DBE. To get a sense of how important inclusion of nonlinear effects are at other sites, the INL seismic hazard is plotted at DBE, 1.5×DBE, 2×DBE, and 3×DBE and compared with Savannah River National Laboratory (SRNL) and Los Alamos National Laboratories (LANL) as shown in Figure 5.4. This plot shows the horizontal SDC-5 DRS at Savannah River with a structural peak similar to 1.5×DBE (INL). It also shows that the horizontal LANL SDC-4 is well above 3×DBE (INL) and LANL SDC-3 is similar to 3×DBE. Many factors can contribute to in-structure response such as depth of soil to rock, depth of facility embedment, soil material properties, variation in soil layer thickness, structural configuration, and structural stiffness.

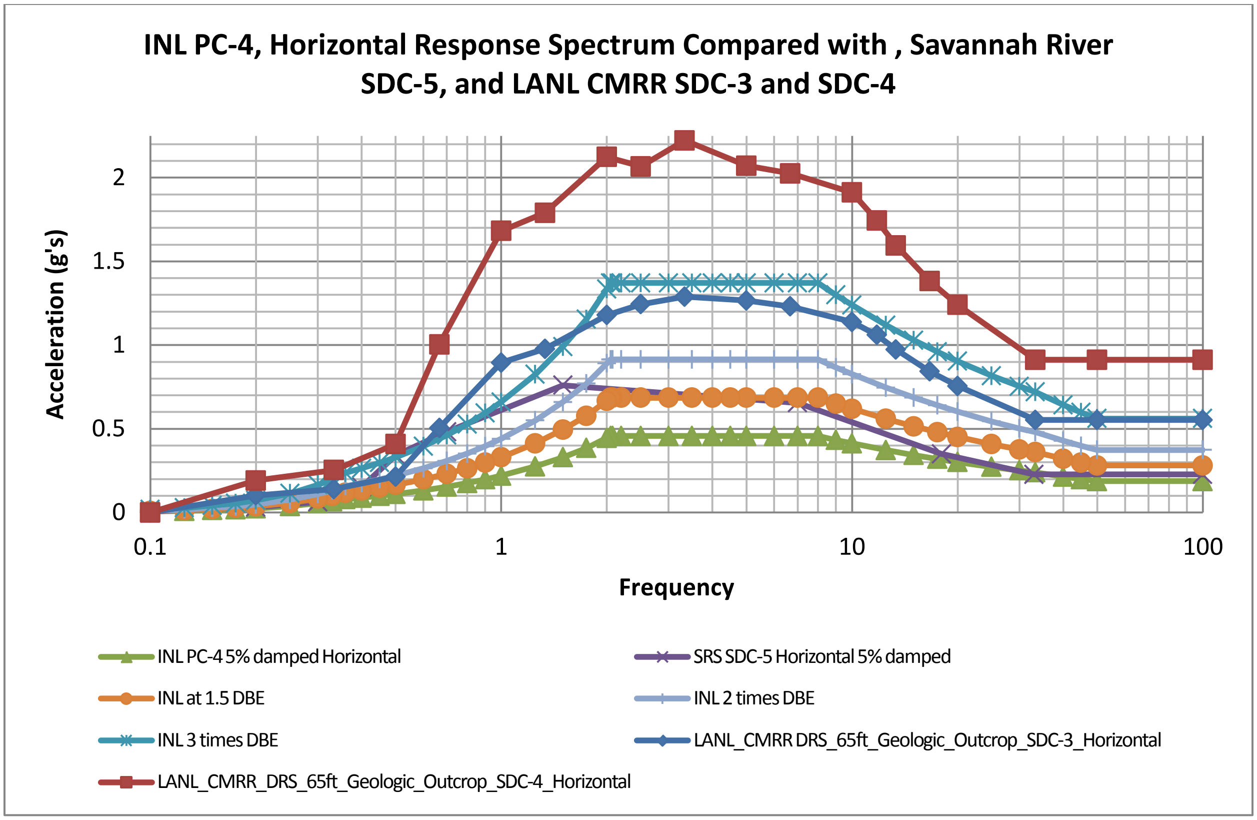


Figure 5.4: Horizontal response spectra at INL (for 4 levels of magnitude), SDC-5 SRNL, and SDC-3 and SDC-4 LANL.

Not considering nonlinear behavior for sites that have moderate to high ground motions may produce overly conservative in-structure analysis results (it is also possible that NLSSI will show some linear results are non-conservative for higher ground motion due to uplift and impact). In seismic risk space, this conservatism may cause nuclear facility owner/operators to invest more money than necessary to mitigate the perceived seismic risk. NLSSI has potential to remove conservatism from seismic probabilistic risk assessments (SPRA) so that best estimate seismic risk numbers can be computed. The next step in the process is application of the NLSSI methodology at a real NPP using an advanced SPRA approach.

5.3 MOOSE Tools Development

The long-term goal of this program is to develop advanced methods and realistic tools to evaluate the performance of virtual nuclear power plants and nuclear facilities to a wide range of external hazards including multiple event scenarios. This would allow nuclear facility owners to virtually test external hazards before the actual facilities are tested with actual hazards. Allows owners to anticipate potential issues and resolve them.

To achieve this long-term goal it is necessary to use the MOOSE framework since development using this framework is developing the capability to model a virtual reactor. To achieve this long-term goal it is necessary to leverage this virtual reactor with structural dynamics capabilities. These would provide the capability to model and quantify external hazard propagation (such as seismic waves).

Adding structural dynamics capability into MOOSE started this year with inclusion of a Newmark Beta time integrator. This solves the second order differential structural dynamics equation:

$$\boldsymbol{M a}_{n+1} + \boldsymbol{C v}_{n+1} + \boldsymbol{K d}_{n+1} = \boldsymbol{F}_{n+1}$$

This provides the capability to solve structural dynamics problems using MOOSE.

Also initiated a contract with UC Davis that will implement a nonlinear soil constitutive model in MOOSE. Additional capabilities need to be added to using MOOSE framework. Figure 5.5 provides an overview of additional capabilities that will be added using MOOSE framework to enhance capabilities to perform seismic analysis.

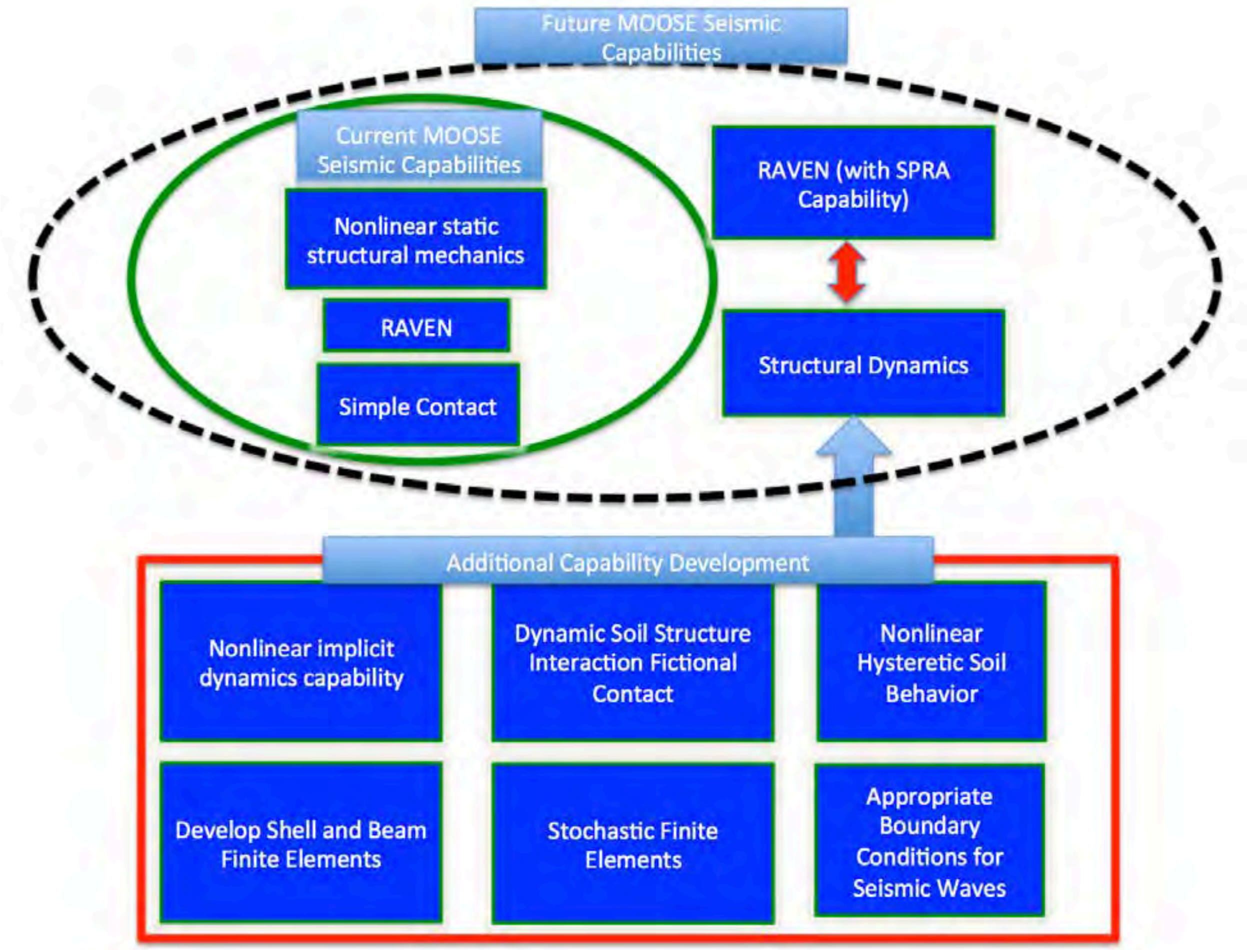


Figure 5.5: Enhanced seismic capability development using MOOSE framework

6. Demonstration Project Plan

The advanced SPRA demonstration is focused on understanding the effect of local nonlinearities on the seismic core damage frequency (SCDF) of a nuclear power plant (NPP). The goal of this effort is to compare SCDFs obtained by a traditional NPP seismic probabilistic risk assessment (SPRA) and a SPRA based on Nonlinear Soil-Structure Interaction (NLSSI) analysis. Soil-structure interaction (SSI) response analysis for a traditional SPRA uses linear geometry, soil properties, and structure properties. The NLSSI analysis will consider geometric and soil nonlinearities.

This initial study is intended to obtain a first estimate on the potential reduction in SCDF that might be achieved by NLSSI analysis relative to the SCDF obtained by a traditional SPRA. The study will consider a representative NPP reinforced concrete reactor building and representative plant safety system. Simplifications in the seismic hazard, structure model, soil properties, and plant system will be introduced to limit the analytical effort in this initial study. Complexity can be added in subsequent phases of this project.

The selected representative NPP structure is a pressurized water reactor building example. It consists of a prestressed concrete containment structure and reinforced concrete internal structure. The structure and its stick model representation are shown in Figure 6.1.

Results from this study will be available in December 2014. These results will demonstrate the importance of including gapping and sliding in SPRAs.

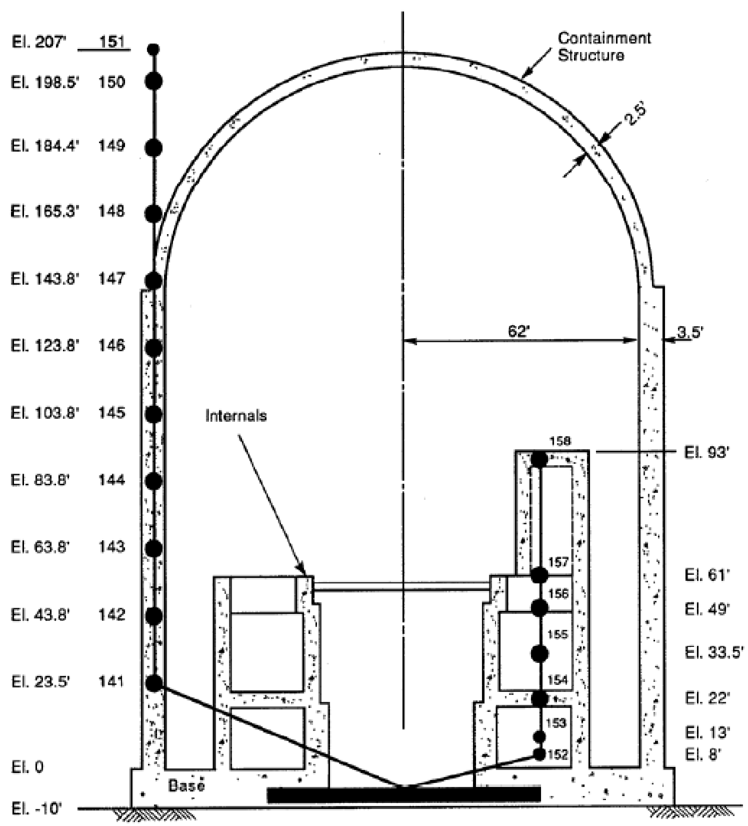
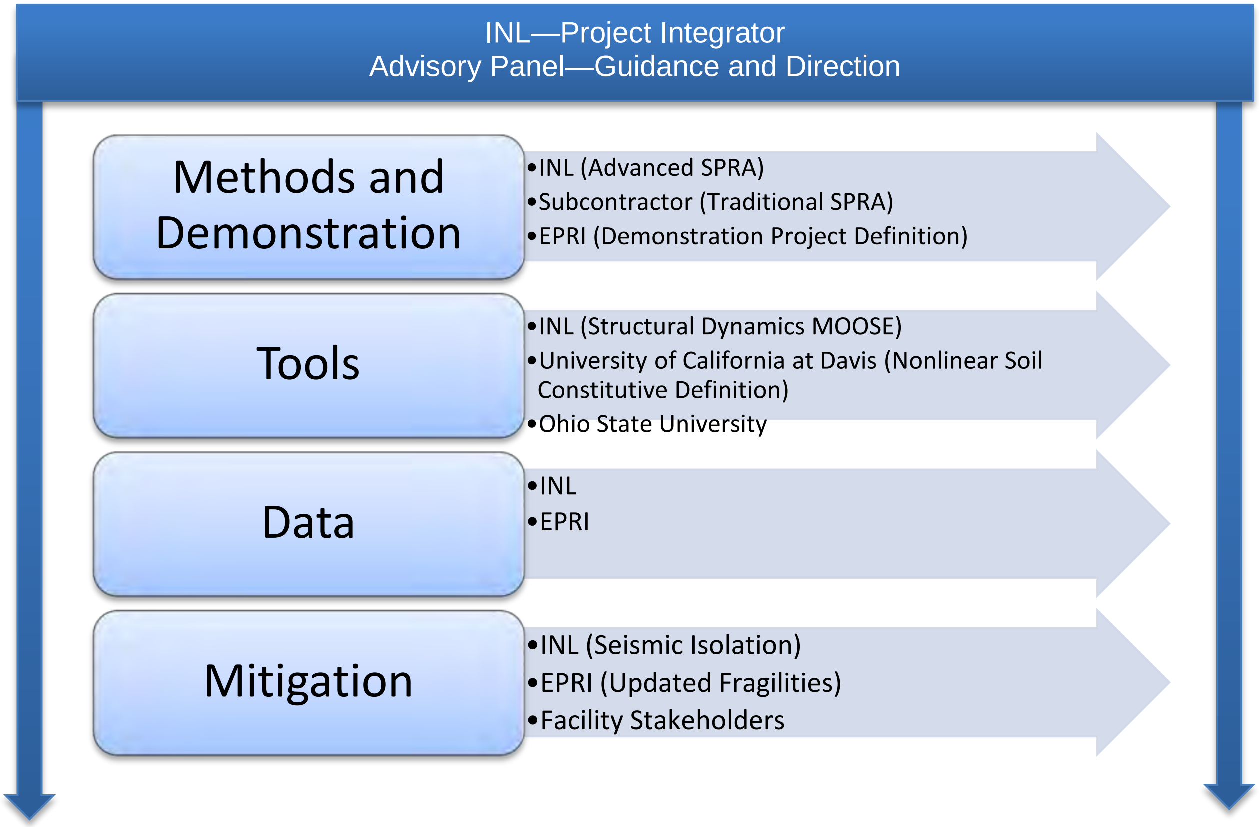


Figure 6.1: Stick Model of the Representative NPP Structure

7. Collaboration

Collaboration is important to the success of this project. By building a strong team to guide the activities and perform the necessary tasks the outcome will be successful. The figure below provides an overview of the team that has been assembled to support this advanced SPRA project.



8. Mitigation

Discussion in this report has been on development of advanced methods and tools and data. The data will be used to verifying and validating (V&V) the methods and tools. Using V&Ved advanced methods and tools will highlight potential risks in NPPs. The final gap to close is mitigating these risk using various approaches such as emergency response facilities such as FLEX, operating procedures in NPPs, modifications to systems and components inside NPPs which could include:

- Modifying existing connection hardware
- Stiffening systems and components (more pipe hangers, ….)
- Seismic isolation of systems and components
- Providing additional levels of redundancy and/or diversity for vulnerable systems/components

9. FY 2014 Accomplishments

- Implemented Newmark-Beta Time Integration Method in MOOSE
    - This is a numerical time integration method
    - Provides the capability to solve dynamic simulations using MOOSE
- Formed an advisory panel to guide the SPRA work
    - This advisory panel consists of three industry experts who will help guide the direction of advanced SPRA activities.
    - Advisory panel members include:
        - Bob Kennedy (Consultant and industry expert)
        - Bob Budnitz (LBL and SPRA expert)
        - Nilesh Choskshi (Just retired from NRC where he lead response to NTTF 2.1)
- Initiated the FY 2014 demonstration project
    - This project uses a generic PWR NPP with an emergency cooling pump system.
    - Brought on subcontractor SGH to perform the traditional SPRA analysis and calculation of fragilities
    - Kick off meeting held on July 23rd with advisory panel and subcontractor SGH
    - Developed project plan for demonstration project
    - Started building system fault trees using SAPHIRE
    - Performed calculations for the traditional SPRA
    - Started calculations for the advanced SPRA
- Demonstrated the importance of using NLSSI in SPRA's
    - This was accomplished by comparing the results of a linear SSI analysis with a nonlinear SSI analysis.
    - The results of this analysis show that the assumption that in-structure response scales linearly with ground motion is not valid at higher levels of earthquake ground motion.

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Appendix A
Project Plan

Nonlinear Soil-Structure Interaction Analysis in the MOOSE Framework Project Plan

A.1 BACKGROUND

Idaho National Laboratories (INL) has an ongoing research and development (R&D) project to understand the effect of local nonlinearities on the seismic core damage frequency (SCDF) of a nuclear power plant (NPP). The goal of this effort is to support a study that compares SCDFs obtained by a traditional NPP seismic probabilistic risk assessment (SPRA) and a SPRA based on Nonlinear Soil-Structure Interaction (NLSSI) analysis. Soil-structure interaction (SSI) response analysis for a traditional SPRA uses linear geometry, soil properties, and structure properties. The NLSSI analysis will consider geometric and soil nonlinearities.

This initial study is intended to obtain a first estimate on the potential reduction in SCDF that might be achieved by NLSSI analysis relative to the SCDF obtained by a traditional SPRA. The study will consider a representative NPP reinforced concrete reactor building and representative plant safety system. Simplifications in the seismic hazard, structure model, soil properties, and plant system will be introduced to limit the analytical effort in this initial study. Complexity can be added in subsequent phases of this project.

A.2 PLAN OVERVIEW

The intent of this Project Plan is to establish an outline for the technical approach to be implemented, and to minimize differences between the two SPRA approaches to the extent possible. The only significant difference between the two approaches should be the introduction of a geometric nonlinearity.

The plan includes the following elements:

- Earthquake Ground Motion: Seismic hazard, ground response spectra, and associated spectrum-compatible acceleration time histories.
- Soil Properties: Median soil profile (unit weight, Poisson's ratio, strain-compatible shear modulus, and damping), dynamic soil properties, and associated variability's.
- Representative NPP Structure: Structure configuration and associated fixed-base model.
- Plant System: Components of the representative plant system, locations in the structure, failure modes to be considered, fault trees and event trees, and system logic model.
- Seismic Response Analysis: Median SSI model, and response analysis methods for the traditional and NLSSI analysis approaches.
- Seismic Fragility Evaluation: Subset of components requiring seismic fragility evaluation, seismic fragility evaluation methodology for traditional SPRA, and approach for determination of seismic capacity distributions for NLSSI analysis SPRA.
- SCDF Quantification: Methods for calculating the SCDFs by the traditional SPRA and NLSSI analysis SPRA.
- Interfaces: Items requiring coordination between INL and Simpson Gumpertz & Heger Inc. (SGH).
- Resources: Sources of information and data that may be used in the project.

A.3 TECHNICAL APPROACH

A.3.1 Earthquake Ground Motion

The earthquake ground motion will be based on the seismic hazard for an existing NPP. Figure 3-1 shows the seismic hazard curve expressed in terms of the horizontal peak ground acceleration (PGA) and associated mean annual frequency of exceedance (MAFE). The Reference Earthquake for the traditional SPRA seismic fragility evaluation will nominally be defined as the uniform hazard spectra (UHS) for the existing NPP at a horizontal PGA of 0.4g. This PGA corresponds to a MAFE of 1.0E-04. Figure 3-2 shows these 5% damped horizontal and vertical Reference Earthquake UHS. As a simplification for this study, the shape of the UHS will be considered to be invariant with MAFE. This study will use a suite of thirty sets of earthquake ground motion time histories compatible with the Reference Earthquake UHS. The horizontal time histories account for variability of the spectral acceleration in any arbitrary direction to the geomean of the two horizontal components. The vertical time histories include variability introduced when the vertical UHS are generated from the horizontal UHS by vertical-to-horizontal (V/H) ratios. Figures 3-3 to 3-5 show the 5% damped response spectra for the horizontal and vertical time histories and the target mean 1.0E-04 UHS.

INL will perform a number of NLSSI analyses at multiple ground motion levels, expected to range from 3 to 5. Each ground motion level will be defined by a PGA (since the UHS shapes are assumed to be constant). PGAs for consecutive ground motion levels will represent progressive increases by factors of 1.5 times. INL will scale the ground motion sets to generate multiple suites using the ratios of the ground motion level PGAs to the Reference Earthquake PGA. The ground motion levels will be selected by trial so that the resulting plant-level conditional probabilities of failure fall in the range between 10% and 80%, and can produce a good fit to a lognormal probability distribution.

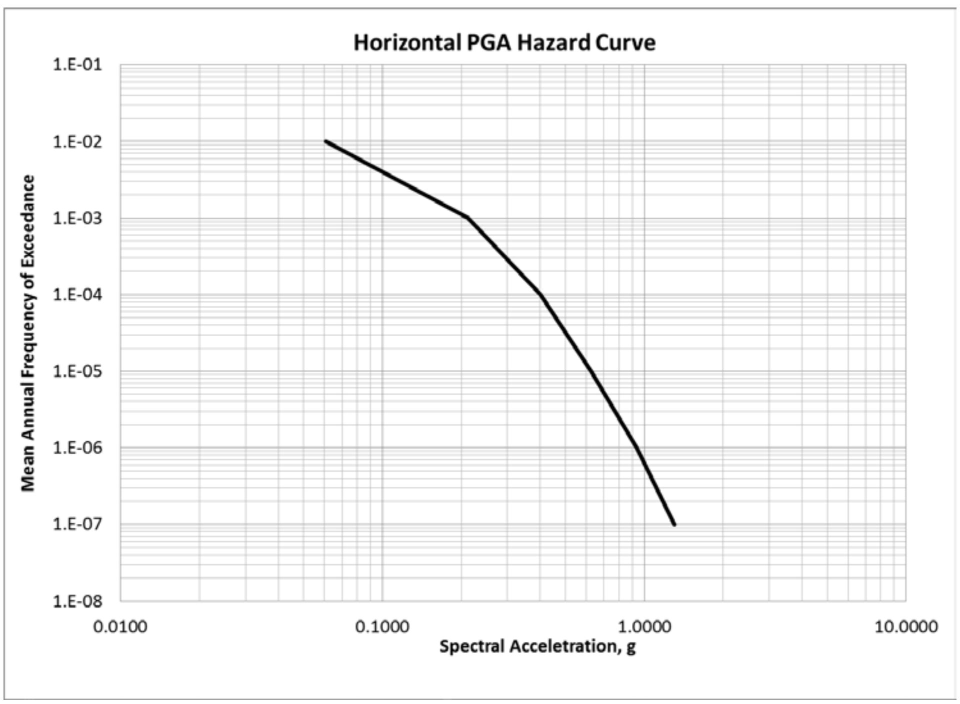

Figure 3-1 – Seismic Hazard Curve for Horizontal PGA

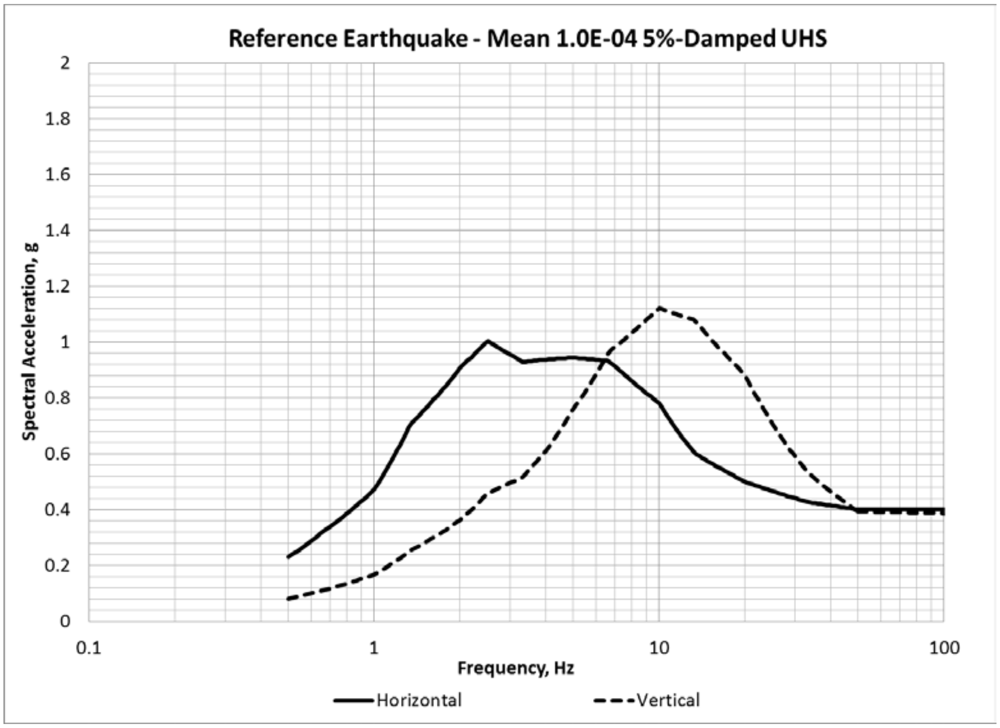

Figure 3-2 – UHS for Mean Annual Frequency of Exceedance of 1.0E-04

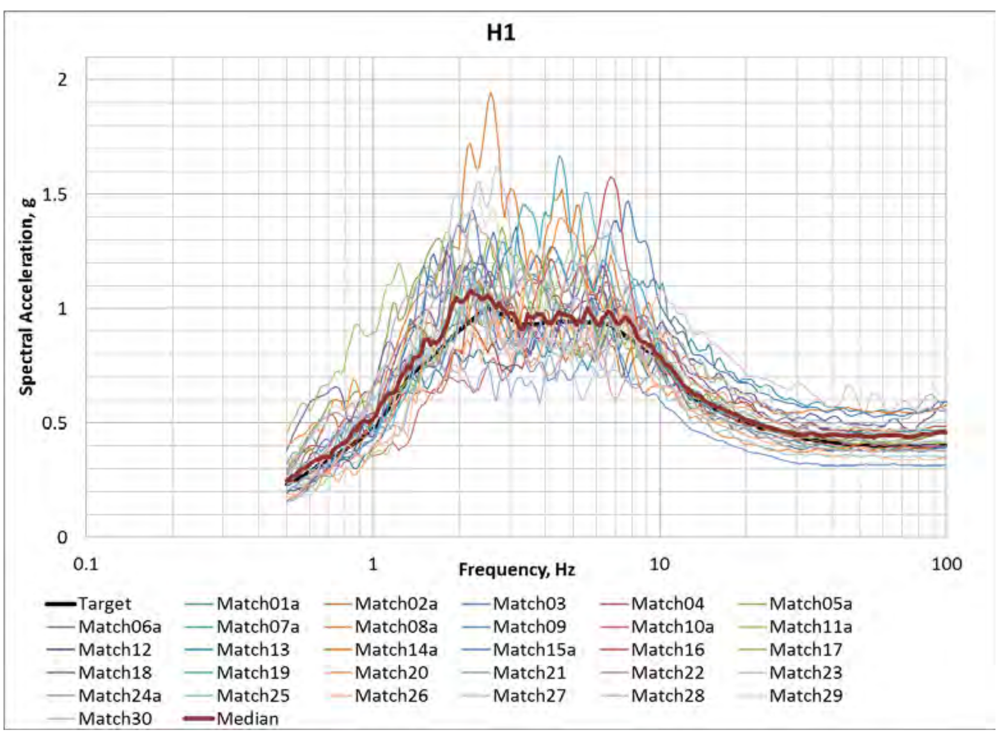

Figure 3-3 – Response Spectra for Reference Earthquake Time Histories, 5% Damping, Horizontal Component H1

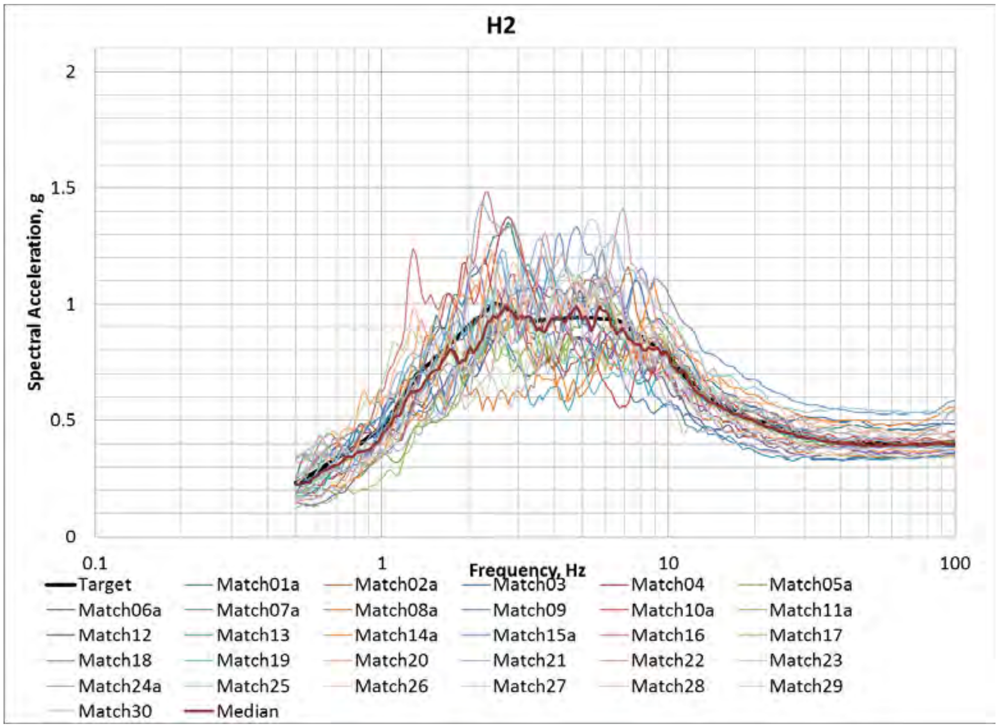

Figure 3-4 – Response Spectra for Reference Earthquake Time Histories, 5% Damping, Horizontal Component H2

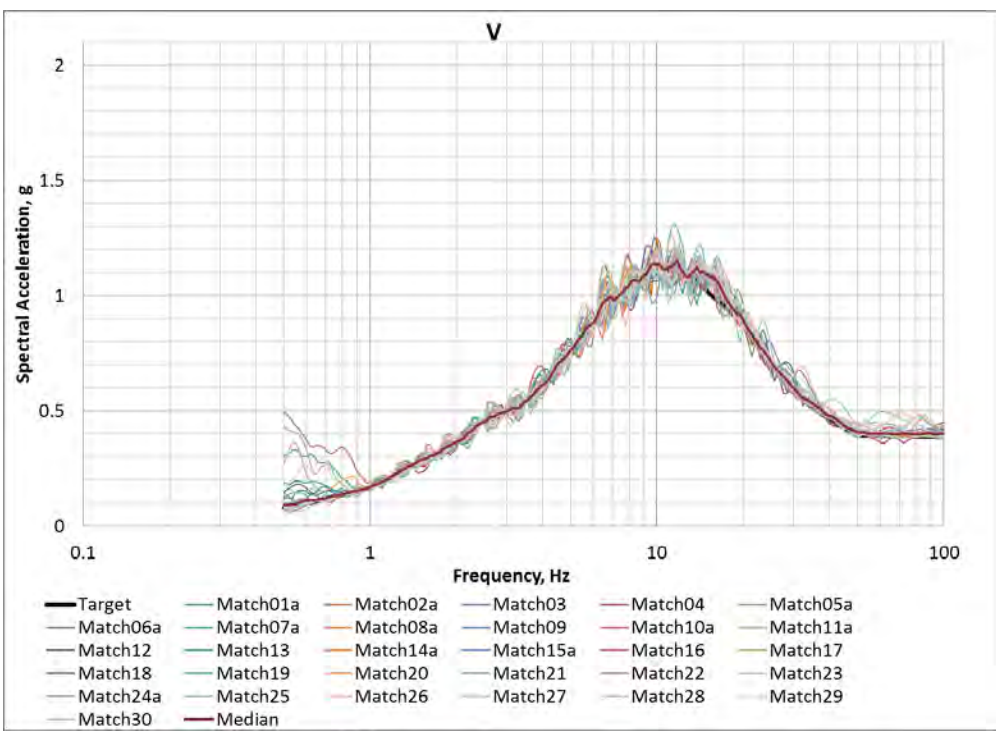

Figure 3-5 – Response Spectra for Reference Earthquake Time Histories, 5% Damping, Vertical Component V

A.3.2 Soil Properties

This study will use a soil profile representing the basalt under the Advanced Test Reactor (ATR). The soil properties listed in Table 3-1 will be considered to be uniform with depth. Excluding soil damping, these values were used in the TRA-670 probabilistic seismic response analysis.

Table 3-1 – Soil Properties

Property	Median	Lognormal Std. Deviation
Unit Weight	159 lb/ft^3	-
Poisson's Ratio	0.35	-
Shear Wave Velocity	3,720 ft/sec	0.27
Shear Modulus	68,320 k/ft^2	0.55
Damping	2%	0.4

A.3.3 Representative Nuclear Power Plant Structure

The selected representative NPP structure is a pressurized water reactor building example obtained from the SASSI2000 User Manual. It consists of a prestressed concrete containment structure and reinforced concrete internal structure. The structure and its stick model representation are shown in Figure 3-6. The median mass, section, and material properties for the stick model are shown in Figure 3-7. The concrete modulus of the internal structure is reduced by a factor of 0.5 to obtain a fundamental frequency near the peak of the UHS. The structure is modeled using linear elastic properties only. Table 3-2 lists frequencies and fractions of total mass participation for the significant modes of the fixed-base structure model. Figure 3-8 shows the mode shapes of the significant horizontal vibration modes. The median damping will be 5% of critical damping.

Table 3-2 – Frequencies and Percentages of Total Mass Participation of Significant Fixed-Base Modes

Mode	Frequency (Cyc/sec)	Percentages of Total Mass Participation			Description
		UX	UY	UZ	
1, 2	5.27	45.6%	45.6%		1st horizontal mode for containment
3, 4	8.46	9.2%	9.2%		1st horizontal mode for internals
5, 6	12.37	20.4%	20.4%		2nd horizontal mode for internals
7	15.64			50.7%	1st vertical mode for containment
8, 9	16.24	9.4%	9.4%		2nd horizontal mode for containment
10	27.83			32.4%	1st vertical mode for internals
13, 14	32.89	7.9%	7.9%		3rd horizontal mode for internals

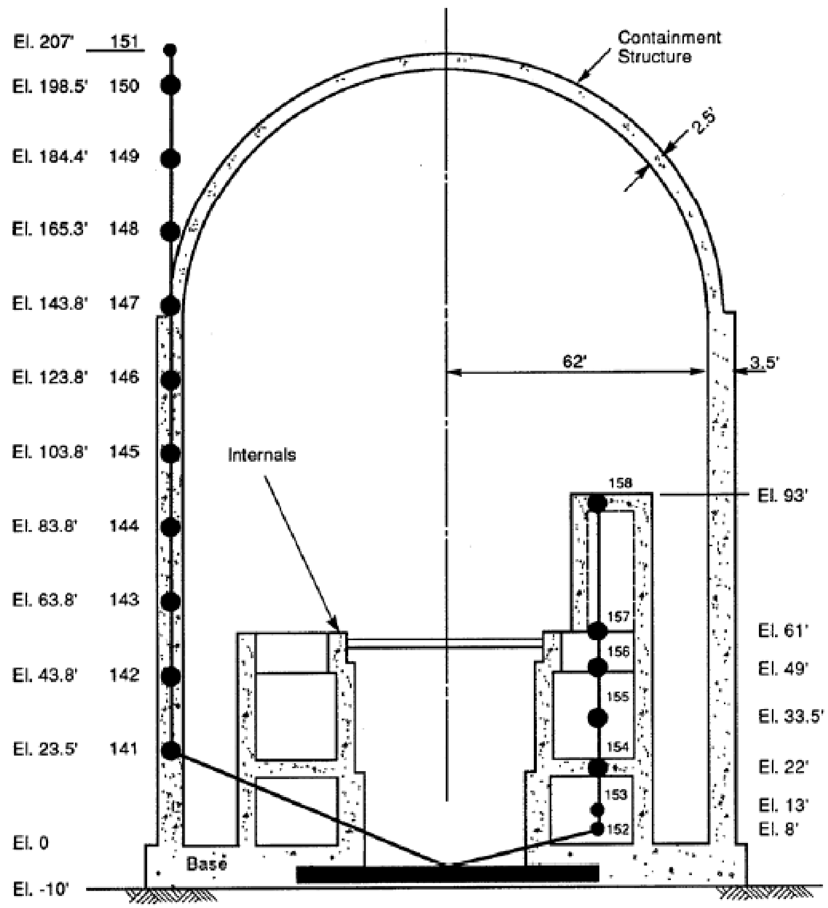

Figure 3-6 –Stick Model of the Representative NPP Structure

Table 4.6-1. Properties of the Structural Models of the Containment Building and Internals
(Concrete Modulus E = 6.9 x 10^5 ksf, G = 2.7 x 10^5 ksf)

Joint Properties			Member Properties			
Mass No.	$M_i g$ (kips)		Location between Joint No.	Area (ft^2)	Shear Area (ft^2)	Moment of Inertia x 10^{-6} (ft^4)
base	20000	C				
1	46000	O	base to 1	1400	700	2.8
3	4200	N	1 to 2	1400	700	2.8
4	4200	T	3 to 4	1400	700	2.8
5	4200	A	4 to 5	1400	700	2.8
6	4200	I	5 to 6	1400	700	2.8
7	4610	N	6 to 7	1400	700	2.8
8	3020	M	7 to 8	990	500	1.9
9	2470	E	8 to 9	990	500	1.5
10	2120	N	9 to 10	990	500	0.8
11	190	T	10 to 11	990	500	0.2
12	2800	I	base to 12	2000	1320	1.1
13	2510	N	12 to 13	2560	1560	1.2
14	6290	T	13 to 14	2210	1460	1.2
15	3760	E	14 to 15	1960	730	1.3
16	8540	R	15 to 16	1740	600	0.9
17	1220	N	16 to 17	780	360	0.2
18	820	A	17 to 18	190	70	0.004

Figure 3-7 – Properties for the Representative NPP Structure Model

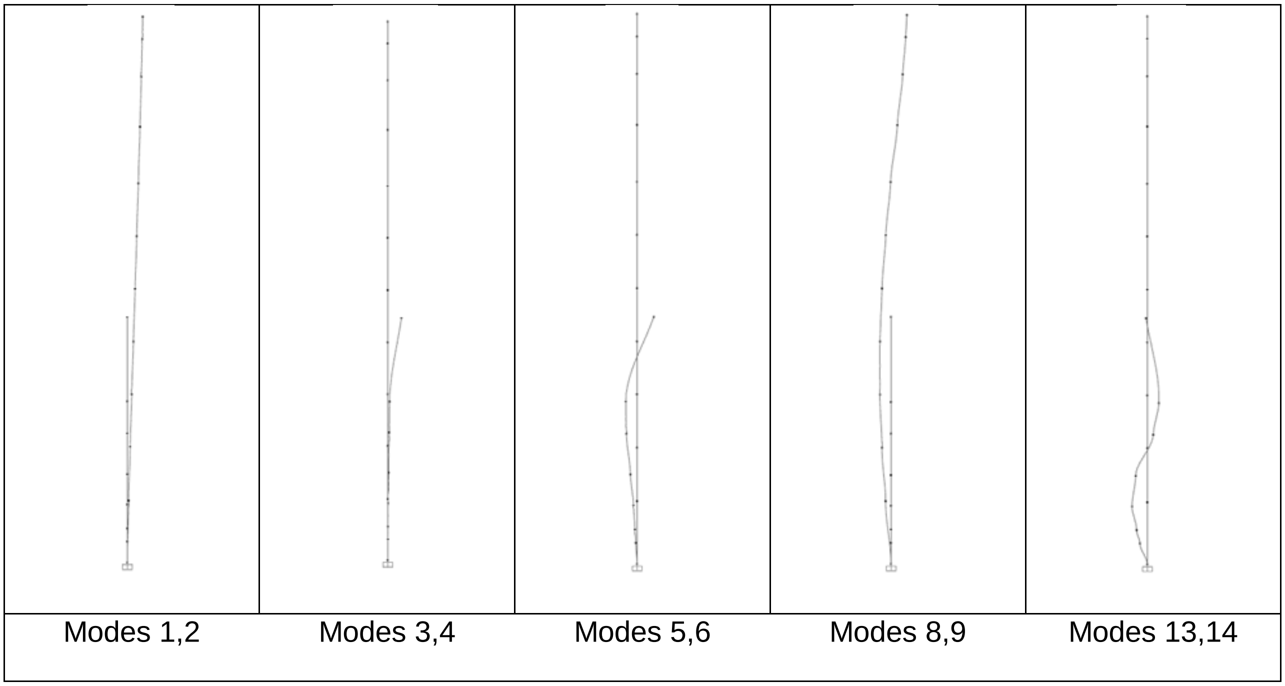

Figure 3-8 – Fixed-Base Structure Model Mode Shapes

A.3.4 Plant System

A.3.4.1 Equipment Components

The plant system will be Emergency Cooling Pump 670-M-11 at the ATR. This pump has an electric motor that is powered by Battery 670-E-58. The motor is started by associated control logic. The system consists of the following components:

- Emergency Cooling Pump 670-M-11 (Figure 3-9)
- Battery 670-E-58 (Figure 3-10)
- Distribution Panel 670-E-23 (Figure 3-11); Circuit Breakers 670-E-23-CB1 and -CB2
- Low Voltage Switchgear 670-E-28 (Figure 3-12); Relay 670-E-28-CR2
- Medium Voltage Switchgear 670-E-1 or 670-E-2 (Figure 3-13, breakers removed); Primary Pump Relay RLY-A
- Flow Indicator Switch FIS-1-24 (Figure 3-14)
- Concrete Block Wall 2B-G2-1 is a seismic interaction hazard to Distribution Panel 670-E-23 and Switchgear 670-E-28.

The plant system components are all located on the internal structure. Emergency Cooling Pump 670-M-11 and Medium Voltage Switchgear 670-E-1 are located at Elevation 22 ft. The remaining system components are located at Elevation 61 ft.

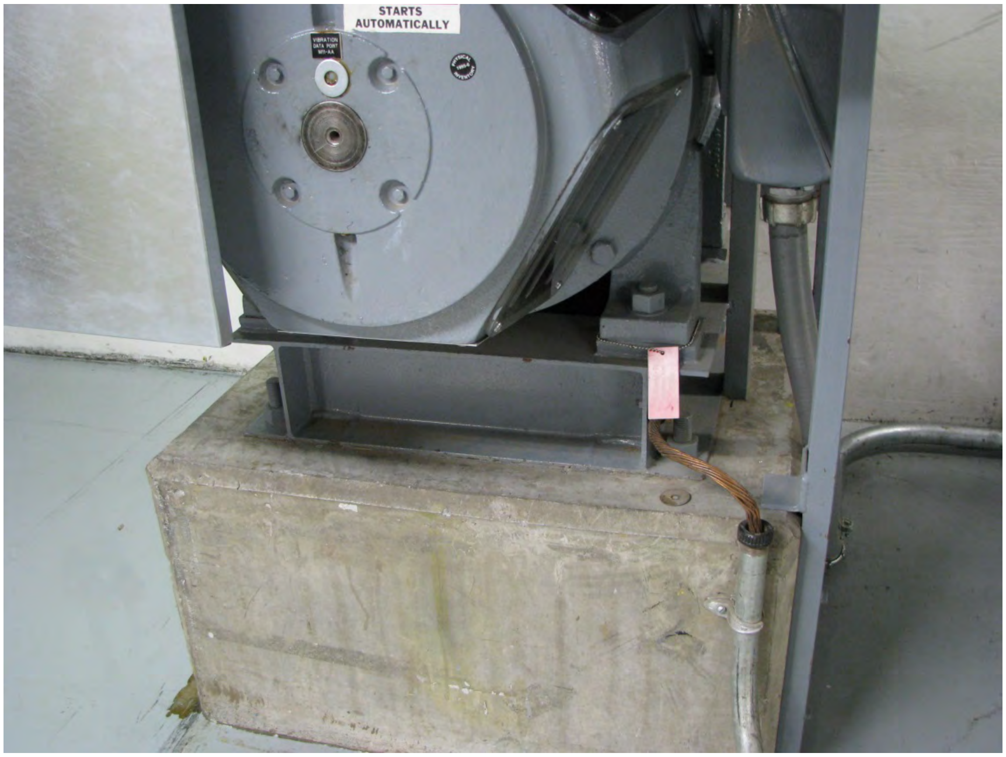

Figure 3-9 – Emergency Cooling Pump 670-M-11

Figure 3-10 – Battery 670-E-58

Figure 3-11 – Distribution Panel 670-E-23

Figure 3-12 – Low Voltage Switchgear 670-E-28

Figure 3-13 – Medium Voltage Switchgear 670-E-2

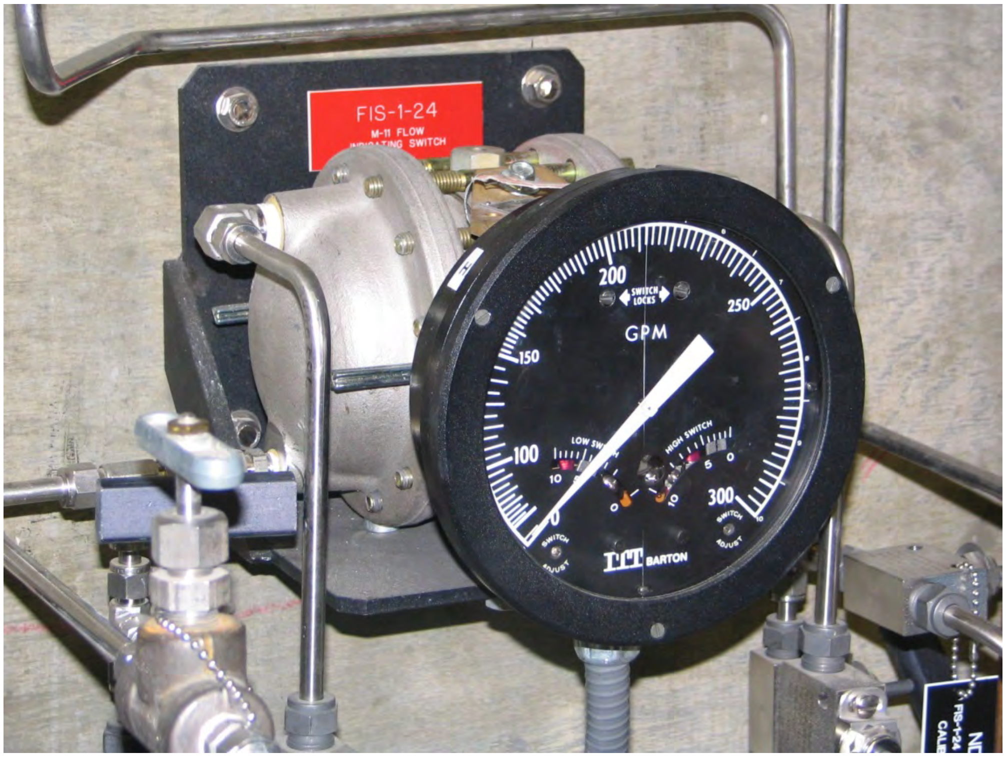

Figure 3-14 – Flow Indicator Switch FIS-1-24

- **A.3.4.2 System Logic Model**

Figure 3-15 shows the logic model for the selected plant system. Figure 3-16 shows the associated event tree. Failure of Low Voltage Switchgear 670-E-28 and Flow Indicator Switch FIS-1-24 are excluded. Collapse of Concrete Block Wall 2B-G2-1 will damage Distribution Panel 670-E-23.

(Note: The plant logic model and event tree are to be developed by INL consistent with the final selection of components to be included.)

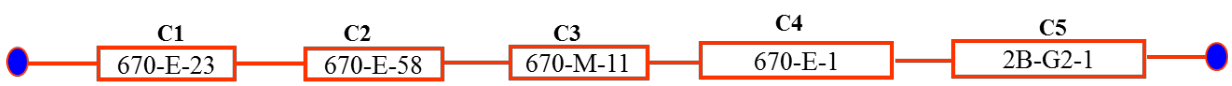

Figure 3-15 – System Logic Model

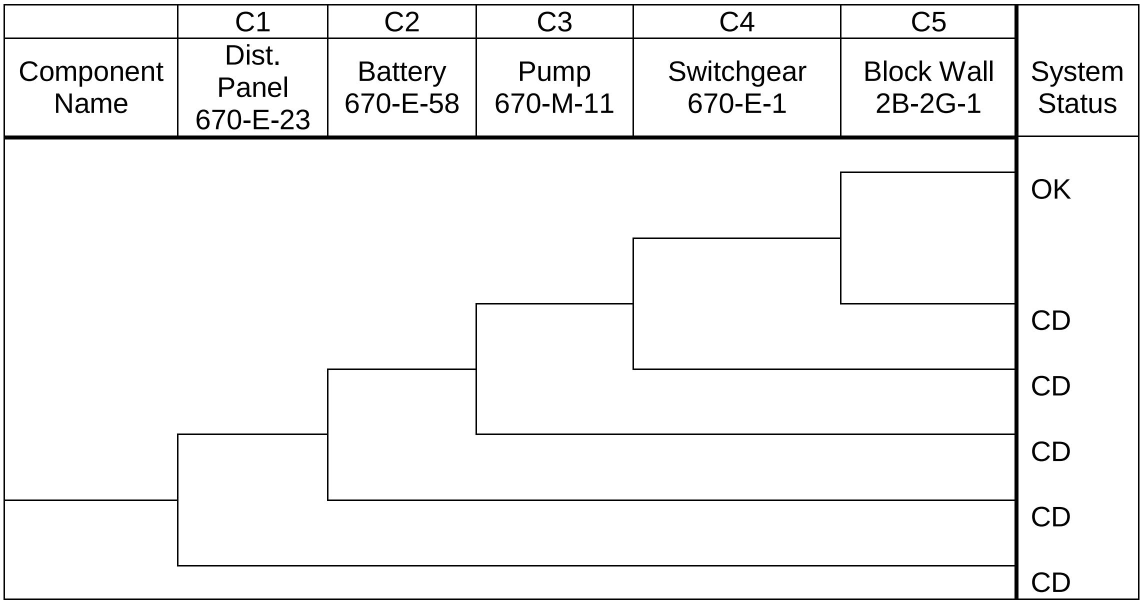

Figure 3-16 – Event Tree

A.3.5 Traditional Seismic Probabilistic Risk Assessment

A.3.5.1 Probabilistic Seismic Response Analysis

SGH will perform a probabilistic seismic response analysis of the representative NPP structure using methods typically implemented in a traditional SPRA. Probability distributions for in-structure response spectra (ISRS) at the locations of components of the selected plant system will be generated. The analysis will consist of the following steps:

1. Ground motion input will consist of thirty sets of acceleration time histories (Section 3.1).
2. The fixed-base eigensolution and mass matrix will be generated for the structure model with median properties using computer program SAP2000 (Section 3.3).
3. Foundation impedances for the median soil profile under the structure (Section 3.2) will be generated using computer program CLASSI.
4. Probability distributions of the structure frequency, structure damping, soil stiffness, and soil material damping will be represented by scale factors with median values of 1.0 and associated lognormal standard deviations. Representative lognormal standard deviations for structure frequency and damping of 0.15 and 0.35, respectively, will be used. Lognormal standard deviations for soil shear modulus and damping of 0.55 and 0.8 will be used (Table 3-1).
5. Probabilistic response analysis will be performed by the Latin Hypercube Sampling (LHS) approach for thirty simulations using computer program CLASSI. Stratified sampling will be used to sample each of the scale factors representing the probability distributions for the variables considered (i.e., structure frequency and damping, soil stiffness and damping, earthquake acceleration time histories). Latin Hypercube experimental design will be used to create the combinations of samples for the simulations.
6. Five percent damped median and 84% ISRS will be generated at the component locations. Three and a half percent damped median ISRS will also be generated.

A.3.5.2 Seismic Fragility Evaluation

SGH will develop seismic fragilities for up to five components of the selected plant system by the Separation of Variables Method presented in Electric Power Research Institute (EPRI) TR-103959, supplemented by guidance in EPRI 1019200. The seismic fragility will be expressed as the probability of component failure conditional on the horizontal PGA. The seismic fragility evaluation will use existing documentation (i.e., Screening and Evaluation Work Sheets, screening calculations, and seismic fragility calculations) developed in the previous ATR DOE/EH-0545 seismic evaluation and SPRA.

As noted below, the equipment components will typically be evaluated for functional and anchorage failure modes. The general approaches for seismic fragility evaluation for these failure modes are summarized as follows:

1. For equipment functional failure, the median seismic capacity will be defined by a 5% damped in-structure spectral acceleration of 4.8g, as recommended in EPRI 1019200. The median seismic demand will be taken as the median peak 5% damped in-structure spectral acceleration including peak clipping. The latter will be determined by "Method 3" in which each of the thirty ISRS from the probabilistic response analysis are clipped by clipping factors that follow the probability distributions determined in EPRI TR-103959.
2. For equipment anchorage failure, the median seismic demand will be based on the median spectral acceleration at the median equipment frequency. Median capacities of post-installed anchors will be based on the DOE/EH-0545 values scaled by a nominal factor of safety of 3.0. Median capacities of cast-in-place anchors will be determined by adjusting DOE/EH-0545 values to median strength reduction factors and material strengths, or will be calculated by ACI 349-06 provisions using median strength reduction factors of 1.0 and median material strengths.

Details on the seismic fragility evaluations of the specific components are as follows. Configurations and anchorage details may be adjusted to achieve a mix of fragilities controlled by functional and anchorage failure.

- Emergency Cooling Pump 670-M-11: The pump is considered rugged for function, and seismic fragility evaluation will consider only anchorage failure. DOE/EH-0545 screening of the pump anchorage was documented in ARES Calculation No. 0602301.01-S-113. The seismic fragility for anchorage failure may be obtained by extending the screening calculation. As a simplification, pump inertial loads for anchorage evaluation will be based on the spectral acceleration for 5% damping rather than 3% damping.
- Battery 670-E-58: The seismic fragility evaluation will consider functional and anchorage failure modes. The battery satisfied the DOE/EH-0545 screening caveats. The seismic fragility for functional failure may consequently be a screening-based fragility or alternatively developed from the battery Generic Equipment Response Spectra (GERS). Seismic fragility evaluation of Batteries 670-E-58 and 670-E-59 was documented in SGH Calculation No. 098122-CA-18. In this evaluation, Battery 670-E-59 was found to control over Battery 670-E-58. The existing seismic fragility evaluation of Battery 670-E-59 will be adapted to this study and used as a substitute for Battery 670-E-58.
- Distribution Panel 670-E-23 containing Circuit Breakers 670-E-23-CB1 and -CB2: The seismic fragility evaluation will consider functional and anchorage failure modes. The distribution panel satisfied the DOE/EH-0545 screening caveats. The seismic fragility for functional failure of the distribution panel and circuit breakers may consequently be a screening-based fragility. DOE/EH-0545 screening of the panel anchorage was documented

in ARES Calculation No. 0602301.01-S-109. This calculation leveraged off INL EDF-4316. Further review is required to determine how these existing calculations can be adapted to this study.

- Medium Voltage Switchgear 670-E-1 containing Primary Pump Relay RLY-A: The switchgear satisfied the DOE/EH-0545 screening caveats, but the anchorage and relay did not. The seismic fragility for structural failure of the switchgear may consequently be a screening-based fragility. The switchgear anchorage and relay were found to be outliers and should consequently be excluded from fragility evaluation.
- Concrete Block Wall 2B-G2-1: This block wall is a seismic interaction hazard to 670-E-23. It was evaluated as part of Wall Group 1 in ARES Calculation No. 0602301.01-S-007.

A.3.5.3 Seismic Core Damage Frequency Quantification

SGH will provide INL the component seismic fragilities. INL will perform the SPRA system analysis for the selected plant system, and quantify the SCDF using computer program SAPHIRE.

A.3.6 Seismic Probabilistic Risk Assessment by Nonlinear Soil-Structure Interaction Analysis

A.3.6.1 Nonlinear Soil-Structure Interaction Analysis

INL will perform nonlinear analysis of the representative NPP structure considering the geometric and soil nonlinearities of interest to this study. Only local soil nonlinearities at the foundation interface will be considered. ISRS at the locations of components of the selected plant system will be generated. This analysis will consist of the following steps:

1. Ground motion input will consist of thirty sets of acceleration time histories (Section 3.1).
2. A median-centered nonlinear model of the structure and soil using a NLSSI method will be developed. Median soil and structure properties will follow Sections 3.2 and 3.3. Nonlinearities will include uplift and sliding at the foundation-soil interface. The median coefficient of friction and associated lognormal standard deviation will be 0.70 and 0.25, respectively. The probability distribution for the coefficient of friction will be represented by scale factors with a median value of 1.0 and associated lognormal standard deviations.
3. Thirty realizations of the NLSSI model will be developed to match the corresponding CLASSI SSI model realizations (Section 3.5.1). Structure damping, soil stiffness, and soil damping scale factors for each model realization will be obtained from the CLASSI probabilistic response analysis. The structure material stiffness scale factors will be the squares of the structure frequency scale factors. In addition, scale factors for the foundation-soil coefficient of friction will be assigned by the stratified sampling approach.
4. Each of the thirty NLSSI model realizations will be combined with the earthquake acceleration time history set assigned in the CLASSI probabilistic response analysis.
5. For each of the ground motions levels, each of the thirty NLSSI model realizations will be analyzed for the associated earthquake acceleration time history set.
6. For each of the thirty simulations for each of the ground motion levels, output will consist of 5% and 3.5% damped ISRS at the component locations, and 5% damped ISRS and acceleration time histories at the foundation centroid.

A.3.6.2 Component Response Distributions

The NLSSI analysis approach uses component seismic capacity distributions, a different form of seismic fragility function that is conditional on an in-structure response quantity that controls each component failure limit-state instead of the input ground motion parameter. The in-structure response quantities controlling the failure of the components included in the plant system models are spectral accelerations at the component locations in the structure. For each spectral acceleration response of interest, SGH will construct a lognormal response distribution for each hazard level as described below.

Response Distributions for Anchorage Failure Modes

Thirty frequency values will be randomly sampled around the median component frequency using the stratified sampling approach. Lognormal standard deviations for frequency uncertainty will be estimated on a component-specific basis.

The median response will be the median of thirty spectral acceleration responses extracted at the sampled frequencies, one from each 5% damped ISRS.

The lognormal standard deviation will be determined using the separation-of-variables approach from the following sources:

Combined structure response and component frequency variability will be based on the ratio of the 84th percentile response to the median response of the sampled frequency response.

Component damping uncertainty will be based on the ratio of the median 3.5% damped response to the median 5% damped response at the nominal component frequency.

Component mode shape uncertainty will be assigned a lognormal standard deviation of 0.05.

For earthquake component combination randomness, the ratio of the median absolute sum of spectral acceleration components to the square root of the sum of squares (SRSS) combination will be considered equal to three standard deviations from the median.

Response Distributions for Functional Failure Modes

Thirty clipped spectral accelerations will be calculated for each component using the 5% damped ISRS and the median clipping factor definition in EPRI TR-103959.

The median response will be the median of the thirty clipped values.

The combined lognormal standard deviation for structure response and clipping factor variability will be determined by "Method 3" in which each of the thirty ISRS from the probabilistic response analysis are clipped by clipping factors that follow the probability distributions determined by EPRI TR-103959.

A.3.6.3 Component Capacity Distributions

The component capacity distribution describes the conditional probability of failure given the in-structure spectral acceleration. As noted previously, the equipment components will typically be evaluated for anchorage and / or functional failure modes. SGH will develop lognormal capacity distributions for these two failure modes in the components of the selected plant system as described below.

Capacity Distributions for Anchorage Failure

The median seismic capacity for anchorage will be determined as follows:

1. The relative magnitudes of the horizontal and vertical spectral acceleration median demands from the reference earthquake will be reviewed. The resulting anchor bolt forces and

anchorage capacities for pullout and shear will be compared to identify the controlling mode of failure and the associated controlling spectral acceleration direction.

2. The relative ratios of the spectral acceleration components will be assumed to remain equal, on average, to the ratios of the median demands
3. The median anchor bolt demand will be combined from three components using the SRSS method.
4. The median spectral acceleration capacity in the controlling direction will be determined so that the resulting anchor bolt demand is equal to the median anchor both capacity.
5. The lognormal standard deviation for anchorage capacity will be determined using the separation-of-variables approach from the following sources:
6. Anchorage strength uncertainty depends on the mode of failure (i.e., steel or concrete) and the anchor type (i.e., post-installed, cast-in-place). The evaluation will be based on a lognormal standard deviation of 0.15 for the compressive strength of concrete and EPRI TR-103959 provisions for anchor bolt failure.
7. ISRS component ratio variability will be estimated by evaluating the distribution of the SRSS bolt demand from the thirty ISRS sets at the reference earthquake. The lognormal standard deviation will be the ratio of the 84th percentile value to the median.

Capacity Distributions for Functional Failure

The median seismic capacity for functional failure will be determined as the median 5% damped spectral acceleration capacity determined by screening (Section 3.5.2).

The lognormal standard deviation for functional failure will be determined for the qualification method according to EPRI TR-103959 and / or EPRI 1019200.

Concrete Block Wall Failure

The block wall capacity distribution will be developed in a manner similar to the anchor bolt capacity. The controlling failure mode and component frequency will correspond to the out of plane direction of the wall. No earthquake component ratio variability will be required.

A.3.6.4 Conditional Probabilities of Component Failure

The conditional probabilities of component failure will be calculated explicitly by convolution of the component capacity distributions and the probabilistic response distributions at each hazard level. Lognormal distributions will be fit to the conditional probabilities of failure as a function of ground motion level PGA.

A.3.6.5 SCDF Quantification

SGH will provide INL the conditional probabilities of component seismic failure for the PGA levels representative of each ground motion level. INL will perform the SPRA system analysis for the selected plant system, and quantify the SCDF using computer program SAPHIRE. The SCDF quantification will use the same number of initiating events as the traditional SPRA approach.

A.3.7 Benchmark Comparison

A benchmark problem will be performed to establish equivalence between the CLASSI and ABAQUS models. SGH will perform thirty CLASSI simulations using a median-centered model of the structure and soil. INL will perform thirty ABAQUS simulations using a median-centered model of the structure and soil with soil uplift and sliding not permitted. The input ground motion will be the Reference Earthquake. The base shears and ISRS distributions will be compared to verify that the two benchmark models obtain similar responses. If needed, the computer models will be revised to achieve an acceptable comparison.

The calculated responses will be reviewed as follows:

The median base shears will be reviewed to confirm that sliding and / or uplift will occur at or somewhat above the Reference Earthquake. If this is not the case, the seismic hazard will be adjusted to achieve this behavior.

The ISRS from the benchmark analysis using CLASSI will be used to estimate preliminary seismic fragilities. Preliminary risk quantification will be performed to confirm that the SCDF is in the range of about 1.0E-05. If this is not the case, the seismic hazard will be adjusted to achieve a SCDF closer to the expected value.