INL/EXT-15-36101

Light Water Reactor Sustainability Program

Industry Application External Hazard Analyses Problem Statement

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EXECUTIVE SUMMARY

This report describes study scenarios within a topic of high importance to the nuclear industry today. We identify external Nature hazards that impose threat to a nuclear power plant (NPP). These hazards can originate at different times and areas, and can be related to each other. We aim to represent these hazards in simulations using realistic model representations of an NPP and hazards to study and understand the effect these external forces impose over time at a given facility.

We will define the problem we study as an “industry application,” hence the problem we define is a realistic representation of an NPP, including systems, structures, and components (SSCs), and the simulations we propose are of direct interest to an NPP owner and operator.

For this industry application within the Light Water Reactor Sustainability (LWRS) Program Risk-Informed Safety Margin Characterization (RISMC) R&D Pathway, we will create the Risk-Informed Margin Management (RIMM) approach to represent meaningful (i.e., realistic facility representation) event scenarios and consequences by using an advanced 3D facility representation that will:

- Identify, model and analyze the appropriate physics that needs to be included to determine plant vulnerabilities related to external events.
- Manage the communication and interactions between different physics modeling and analysis technologies.
- Develop the computational infrastructure through tools related to plant representation, scenario depiction, and physics prediction.

In order to enable probabilistic aspects of NPP external events modeling, we will be using event simulation as the quantification method. Successfully linking probabilistic simulation to external events physics is a key facet of advanced methods and will directly address problems such as highly time-dependent flooding scenarios.

One of the unique aspects of the RISMC approach is how it couples probabilistic approaches (the scenario) with mechanistic phenomena representation (the physics) through simulation. This simulation-based modeling allows decision makers to focus on a variety of safety, performance, or economic metrics.

The primary purpose of using industry applications in advanced safety analysis is to demonstrate advanced risk-informed decision making capabilities in relevant, realistic industry applications. The end goal of these activities is the full adoption of the RISMC tools by industry applied to their decision making process.

We identify four elements of an Industry Application:

(a) Demonstrate:
- Provide confidence and a technical maturity in the RISMC methodology (essential for broad industry adoption);
- Strong stakeholder interaction required;
- Address a wide range of current relevant issues (see also item (d));
- Three phase approach:
    (1) Problem Definition (3-6 months);
    (2) Early Demonstration (eDemo) (limited scope) (6-12 months);
    (3) Complete Application and Validation (Long Term- Methods, Tools, Data) (1-5 years).

(b) Advanced:
- Analyze multi-physics, multi-scale, complex systems;
- Use of a modern computational framework;

- A variety of Methods, Tools, and Data can be utilized (e.g. use of legacy tools and state-of-the-art tools);
- Be as realistic as practicable (with the use of appropriate supporting data);
- Consider uncertainties appropriately and reduce unnecessary conservatism when warranted'

(c) Risk-Informed decision making capabilities:
- Use of an integrated decision process;
- Integrated consideration of both risks and deterministic elements of safety.

(d) Relevant industry applications:
- There are several Industry Applications (IA) covering a wide range of current industry issues that will be studied.

We will focus this report on *Enhanced External Hazard Analyses*, with emphasis on the first phase of the demonstration approach, *Problem Definition*. In the body of this report we will define the problem with an industry perspective, and debate its merits under a margin management point of view of decision making for the plant owner/operator.

The early demonstration this Industry Application will solve includes two external hazards, seismic and flood. The flooding at the NPP is caused by both seismically-induced failure of an adjacent levy and seismically-induced internal flooding as a result of pipe breaks within the NPP. The early demonstration will assess the impact of the seismically-induced flooding using the RIMM process. Elements of this process include development of a generic NPP at a generic site, and generic levy and seismic hazard. The problem will assume multiple seismic events that produce ground motion at the generic site. These ground motions will be used to assess the probabilities of SSC failures at the NPP and the adjacent levy. Based on the probabilities of failure on piping systems and of the levy flooding, analysis will be run in those locations.

We will also be seeking industry support in applying advanced research and development (R&D) methods and tools to evaluate external hazard risk and decision-making. The seismic portion of the industry application will focus on understanding the benefits of using advanced SPRA methods and tools to perform calculations for actual nuclear power plants (NPP). For the initial activity, INL will apply advanced NLSSI to a realistic NPP and system and a realistic soil site.

This document is organized to describe, first, the Risk-Informed Margin Management approach used in this Industry Application, in Section 1. Section 2 describes the analyses of external events, including multi-hazard analysis in more detail. Section 3 describes the current seismic and flooding ongoing work, while Section 4 describes the longer-term activities and provides a broader perspective on the objectives and long-term approach of the elements of the research focused on advancing external events PRA. An estimate of schedule, cost and planning for the next five years is given in Section 5. Lastly, the Appendix of this document provides a historical perspective on the regulatory and technical evolution and describes how this evolution has led to the needs and opportunities addressed by the current work.

CONTENTS

FIGURES

TABLES

ACRONYMS

AEC	Atomic Energy Commission
ASCE	American Society of Civil Engineers
ASPRA	Advanced Seismic Probabilistic Risk Assessment
CDF	Core Damage Frequency
CFR	Code of Federal Regulation
DBE	Design Basis Earthquake
DOE	Department of Energy
ECCS	Emergency Core Cooling System
EHEC	External Hazards Experimental Center
EPRI	Electric Power Research Institute
FLEX	Diverse and Flexible Coping Strategies
GDC	General Design Criteria
GI	Generic Issue
GMRS	Ground Motion Response Spectra
GP	Gaussian Process
IA	Industry Application
INL	Idaho National Laboratory
IPEEE	Individual Plant Examination for External Events
ISRS	In-Structure Response Spectrum
LERF	Large Early Release Frequency
LHS	Latin Hypercube Sampling
LWRS	Light Water Reactor Sustainability
MOOSE	Multiphysics Object Oriented Simulation Environment
NLSSI	Nonlinear Soil-Structure Interaction
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission

NTTF	Near-Term Task Force
NUREG	Nuclear Regulatory Report
PFHA	Probabilistic Flood Hazard Assessment
PGA	Peak Ground Acceleration
PRA	Probabilistic Risk Assessment
PSHA	Probabilistic Seismic Hazard Assessment
RAVEN	Risk Analysis and Virtual Control Environment
R&D	Research and Development
RELAP7	Reactor Excursion and Leak Analysis Program version 7
RG	Regulatory Guide
RIMM	Risk-Informed Margin Management
RISMC	Risk-Informed Safety Margin Characterization
ROM	Reduced Order Model
ROP	Reactor Oversight Program
SASSI	System for Analysis of Soil Structure Interaction
SCDF	Seismic Core Damage Frequency
SECY	NRC Action Memorandum
SDP	Significant Determination Process
SPH	Smooth Particle Hydrodynamics
SPID	Screening, Prioritization and Implementation Details
SPRA	Seismic Probabilistic Risk Assessment
SSCs	Structures, Systems, and Components
SSE	Safety Shutdown Earthquake
SSI	Soil-Structure Interaction

Industry Application External Hazard Analyses Problem Statement

1. INTRODUCTION

Design of nuclear power plant (NPP) facilities to resist external hazards has been a part of the regulatory process since the beginning of the NPP industry in the United States (US), but has evolved substantially over time. The original set of approaches and methods were entirely deterministic in nature and focused on a traditional engineering margins-based approach. In this approach, design is undertaken for each structure, system, and component (SSC) individually based on achieving a capacity that is expected to provide a minimum margin over some specific design load of interest. Neither the risk significance of the SSC nor its role within the facility is considered. The traditional approach also does not account for operator action, redundancy and other risk-related element.

Over time probabilistic and risk-informed approaches were also developed and implemented in US Nuclear Regulatory Commission (NRC) guidance and regulation. A defense-in-depth framework was also incorporated into US regulatory guidance over time. As a result, today, the US regulatory framework incorporates deterministic and probabilistic approaches for a range of different applications and for a range of natural hazard considerations. This framework will continue to evolve as a result of improved knowledge and newly identified regulatory needs and objectives, most notably in response to the NRC activities initiated in response to the 2011 Fukushima accident in Japan.

Although the US regulatory framework has continued to evolve over time, the tools, methods and data available to the US nuclear industry to meet the changing requirements have largely remained static. Notably, there is room for improvement in the tools and methods available for external event probabilistic risk assessment (PRA), which is the principal assessment approach used in risk-informed regulations and risk-informed decision-making. This is particularly true if PRA is applied to natural hazards other than seismic loading. Development of a new set of tools and methods that incorporate current knowledge, modern best practice, and state-of-the-art computational resources would lead to more reliable assessment of facility risk and risk insights (e.g., the SSCs and accident sequences that are most risk-significant), with less uncertainty, and reduced potential conservatisms. New tools would also benefit risk-informed approaches to assessing and managing margin, as discussed the remainder of Section 1 of this document.

Section 2 of this document describes the analyses of external events, including multi-hazard analysis in more detail. Section 3 describes the current INL work, while Section 4 describes the longer-term activities and provides a broader perspective on the objectives and long-term approach of the elements of the research focused on advancing external events PRA. The Appendix of this document provides a historical perspective on this regulatory and technical evolution and describes how this evolution has led to the needs and opportunities addressed by the current work.

1.1 The Risk-Informed Margin Management (RIMM) Approach

As noted, the new tools and methods being developed have a number of applications in that support the nuclear industry including, a risk-informed margins management approach. An effective RIMM application is one that balances costs with safety as illustrated notionally in Figure 1.

The focus on RIMM provides a technical basis to understand and manage hazards. At a nuclear facility, a hazard is a condition that is or causes a deviation in the normal operation of something. Examples of the types of hazards that may exist at a nuclear power plant (NPP) include different types of kinetic energy (e.g., motion from a seismic event) and potential energy (e.g., energy release by shorted equipment during a flood). These types of hazards complicate the determination of safety in any complex facility. However, in this industry application, we propose advanced methods to represent these potential impacts to safety by developing the technology to incorporate physics (via probabilistic and mechanistic modeling) into scenarios.

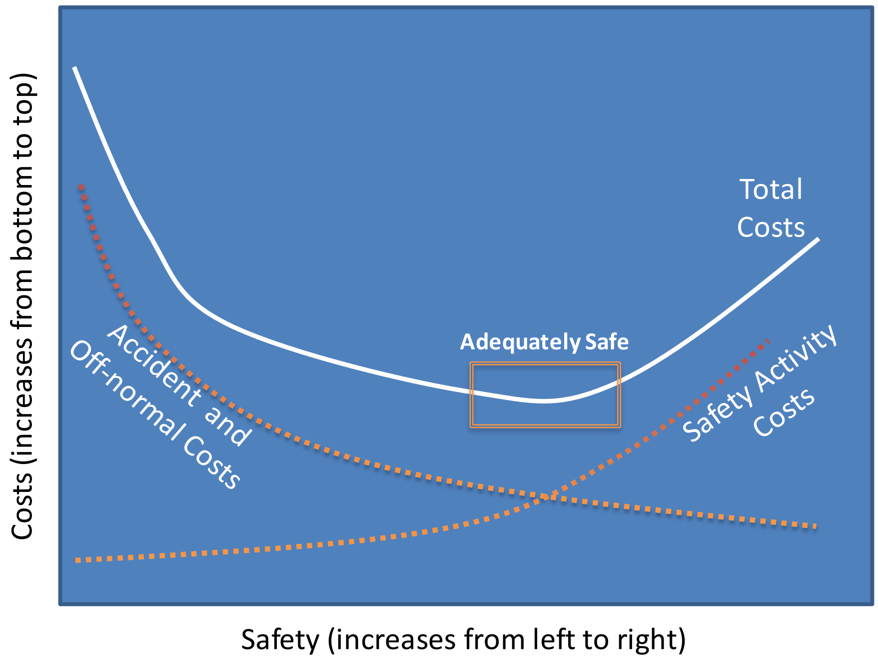

Figure 1. RIMM Balances Safety with Activities in Order to Promote Cost-Effective Decisions.

A scenario happens when initiating events occur, system control responses (including operator actions) fail, and the consequence severity is not limited as well. External events hazards may impinge on a NPP in several ways:

- They may provide enabling events (conditions that permit the scenario to proceed);
- They may affect the occurrence of initiating events (a departure from a desired operational envelope to a state where a control response is required);
- They may challenge system controls or safety functions;
- They may defeat mitigating systems.

For this industry application within the Light Water Reactor Sustainability (LWRS) Program Risk-Informed Safety Margin Characterization (RISMC) R&D Pathway, we will create the RIMM approach to represent meaningful (i.e., realistic facility representation) event scenarios and consequences by using an advanced 3D facility representation that will:

- Identify, model and analyze the appropriate physics that needs to be included to determine plant vulnerabilities related to external events.
- Manage the communication and interactions between different physics modeling and analysis technologies.
- Develop the computational infrastructure through tools related to plant representation, scenario depiction, and physics prediction.

External hazards of interest have a primary impact on the nuclear facility that may also lead to secondary phenomena. Examples of external hazards that cause primary impact are seismic shaking, flooding, and high winds. Examples of secondary phenomena induced by a seismic scenario are dam and levy failure, landslide, internal flood, and internal fire.

A notional depiction of this 3D representation approach is shown in Figure 2. As shown in this figure, we "layer" the different analyses that play a role in a particular scenario. The approach has several defining attributes focused within four general areas:

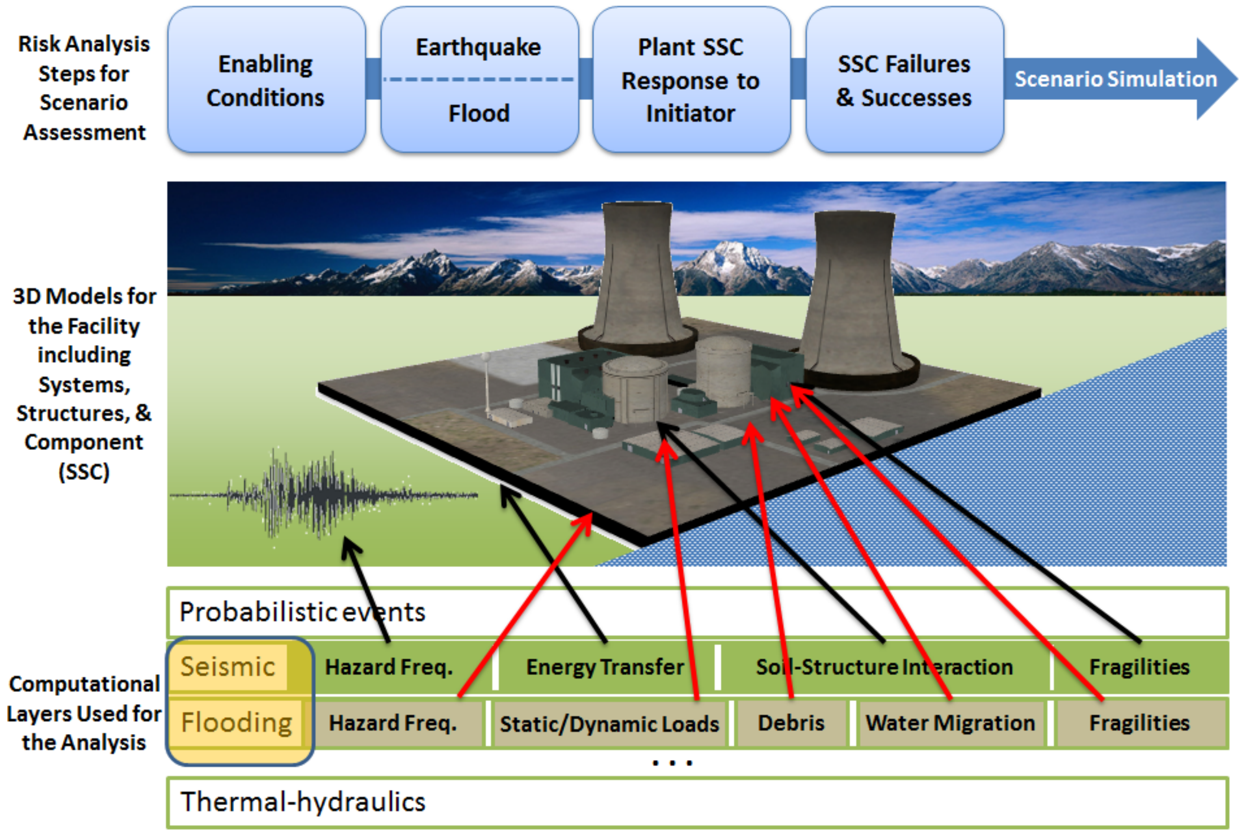

Figure 2. High-Level Features of the External Events Analysis Approach.

1. Enabling Conditions – The enabling conditions are those initial boundary conditions that play a role in defining what occurs (or not) during a specific external events scenario. For example, lack of adequate wall penetration sealers may result in increased flood hazard (and scenarios where water enter buildings via penetrations), while flood doors with proper seals may result in reduced flood hazard (and help to prevent flooding scenarios).
2. Flood Initiating Event Representation – Different types of floods result in a variety of different flooding hazard curves. These hazard curves are models representing the magnitude (how bad) and frequency (how often) of the flooding condition.
3. Plant Response – An approach to effectively representing hazards and their effect on the NPP physical behavior is simulated as part of the simulation. In some cases, multiple models of specific phenomenon may play a role in a sequence. For example, how spatial effects may drive a scenario (e.g., a pipe break caused by a seismic event may flood a pump room) could be determined using different methods for the different risk drivers found in a particular scenario. Impactful conditions on plant to be potentially included in the modeling for multiple NPPs on a site are:
    a. Dynamic forces from water
    b. Debris
    c. Scouring of the plant site
    d. Migration of water on the plant site

4. Structures, Systems, and Component Impacts – A representation of key SSCs will be modeled within the 3D risk analysis model for a particular NPP. We will be able to use this model to simulate potential hazard-specific susceptibilities (e.g., energy from a seismic event may fail a component, flooding may disable many components in a room). Potential impacts to be modeled include:
    a. Inundation
    b. Spraying
    c. Mechanical insults
    d. Debris issues
    e. Migration of water throughout buildings

In order to enable probabilistic aspects of NPP external events modeling, we are using event simulation as the quantification method. Successfully linking probabilistic simulation to external events physics is a key facet of advanced methods and will directly address problems such as highly time-dependent flooding scenarios.

One of the unique aspects of the RISMC approach is how it couples probabilistic approaches (the scenario) with mechanistic phenomena representation (the physics) through simulation. This simulation-based modeling allows decision makers to focus on a variety safety, performance, or economic metrics. For example, while traditional risk assessment approaches for external hazards attempt to quantify core damage frequency (CDF), RIMM approaches may instead wish to consider other metrics such as:

• Magnitude of the hazard – for example, the height of water on buildings, or the height of water inside strategic rooms. The "magnitude" might be measured (during the simulation) by metrics such as water height, seismic energy, water volume, water pressure, etc.

• Damage to the plant (but not core damage) – for example, we may be interested in scenarios in which the facility does not see core damage, but would still experience extensive (or even minor) damage. The "damage" might be measured (again during the simulation) by metrics such as total number of components failed, cost of components destroyed, structures rendered unusable, the length of time the facility is impacted (hours versus months), etc.

The defining difference between these new RIMM metrics and traditional ones such as CDF is that they represent observable quantities (e.g., the number of components failed, the costs related to the event, the height of water in a room, the duration of the event) rather than just a statistical average of an event frequency. We believe these new metrics that are provided by the RISMC simulation yield enhanced decision-making capabilities for nuclear power plants.

1.2 RIMM Industry Applications

Advanced safety analysis focuses on modernization of nuclear power safety analysis using verified and validated methods and tools; implementing state-of-the-art modeling techniques; taking advantage of modern computing hardware; and combining probabilistic and mechanistic analyses to enable a risk-informed safety analysis process. The modernized tools will maintain the current high level of safety in our nuclear power plant fleet, while providing an improved understanding of safety margins and the critical parameters that affect them. Thus, the set of tools will provide information to inform decisions on plant modifications, refurbishments, and surveillance programs, while improving economics. The set of tools will also benefit the design of new reactors, enhancing safety per unit cost of a nuclear plant.

Risk-informed approaches provide a technical basis for understanding and managing hazards (i.e., safety risks). In addition, risk-informed approaches can be used to estimate costs (i.e., economic risks) to support safety decisions. While the focus of advanced safety analysis is on "facility" safety, it should be noted that these facilities are managed by diverse organizations (i.e., the nuclear industry, the Department of Energy (DOE), and associated oversight organizations). The benefits to be derived from the RISMC products will be applicable to all three groups.

The primary purpose of industry applications in advanced safety analysis is to demonstrate advanced risk-informed decision making capabilities in relevant, realistic industry applications. The end goal of these activities is the full adoption of the RISMC tools by industry applied to their decision making process.

The four elements of the above proposition are further explored below:

(e) Demonstrate

- Provide confidence and a technical maturity in the RISMC methodology (essential for broad industry adoption)
- Strong stakeholder interaction required
- Address a wide range of current relevant issues (see also item (d))
- Three phase approach
    (4) Problem definition (3-6 months)
    (5) Early Demonstration (eDemo) (limited scope) (6-12 months)
    (6) Complete Application and Validation (Long Term- Methods, Tools, Data) (1-5 years)

(f) Advanced

- Analyze multi-physics, multi-scale, complex systems
- Use of a modern computational framework
- A variety of Methods, Tools, and Data can be utilized (e.g. use of legacy tools and state-of-the-art tools)
- Be as realistic as practicable (with the use of appropriate supporting data)
- Consider uncertainties appropriately and reduce unnecessary conservatism when warranted

(g) Risk-Informed decision making capabilities

- Use of an integrated decision process
- Integrated consideration of both risks and deterministic elements of safety

(h) Relevant industry applications

- There are four Industry Applications (IA) carefully selected to cover a wide range of current industry issues (in order of importance):
    IA1 – Performance-Based ECCS Cladding Acceptance Criteria
    IA2 – Enhanced External Hazard Analyses (multi-hazard)
    IA3 – Reactor Containment Analysis
    IA4 – Long Term Coping Studies/FLEX

The focus of this report is on IA2, with emphasis on the first phase of the demonstration approach: Problem Definition. In the next chapters we will define the IA2 problem with an industry perspective, and debate its merits under a margin management point of view of decision making for the plant owner/operator.

2. EXTERNAL EVENTS AND MULTI-HAZARD ANALYSIS

2.1 The Nuclear Industry Perspective

The NRC Near-Term Task Force (NTTF) recommendations 2.1, 2.2 and 2.3 are related to seismic and flooding hazard and safety. Recommendation 2.2, which requires a longer-term rule making activity, is focused on requiring periodic (10 year) reevaluation of natural hazards at operating NPPs. Recommendation 2.1 focused on the reevaluation of seismic and flooding hazard and risk at operating NPPs. Recommendation 2.3 was implemented immediately and focused on immediate seismic and flood walkdowns of the facility to confirm that the NPPs current licensing basis was being met. The Recommendation 2.3 activities have, essentially, been completed.

It is important to note that, at the time of the Fukushima accident, the NRC was already working on a reevaluation of the seismic safety of US NPPs as a result of the NRC's Generic Issue 199 (GI-199) evaluation that had been ongoing for several years. In response to GI-199, the NRC was already actively working with

DOE and EPRI on projects to develop new seismic source characterization and seismic ground motion models for the central and eastern US. The NRC was also in the final stages of developing NUREG 2117, to provide additional practical guidance on conducting hazard assessment studies. The NRC was also working on Generic Issue 204, “Flooding of Nuclear Power Plant Sites Following Upstream Dam Failure.” The NTTF Report issued by the NRC considered this ongoing work.

In March 2012, the NRC issued a 50.54(f) Request for Information letter to all operating NPPs. Enclosure 1 of that letter, “Recommendation 2.1: Seismic,” described the actions related to seismic hazard and risk reassessments for licensees to take in response to the letter. In response to the 50.54(f) letter, EPRI and NRC staff and contractors worked together to further refine the process initially described in Enclosure 1 and to address several specific technical areas where additional guidance would bring greater efficiency and reduce uncertainty in the conduct of the hazard and risk assessment activities. The outcome of that collaboration was EPRI Report 1025287, “Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” which was endorsed by the NRC. The SPID report (as it is commonly called) provides targeted information and is not intended to be general SPRA implementation guidance, although some of the technical approaches in the report are likely to be used in the future. Figure 3 of the SPID (reproduced below) lays out Phase 1 of the reevaluation process, which was collaboratively enhanced from that originally provided in Enclosure 1 of the 50.54(f) letter.

In Phase one of the process implemented by the NRC, the seismic hazard was reevaluated for all operating reactors using the guidance for new reactors in place at the time of the evaluation. Those NPPs for which the new hazard assessment, expressed in terms of a Regulatory Guide 1.208 Ground Motion Response Spectrum, exceeded the original design basis in the 1 to 10 Hz frequency range “screened into” further safety/risk evaluation. Those NPP for which the GMRS exceeds 1.3 times the design basis in the 1 to 10Hz range are required to perform a seismic PRA.

Plants that screen in to additional risk assessment, but do not exceed the design basis by 30% can chose to do a seismic margin assessment instead. Although this option was provided, and the NRC issued Interim Staff Guidance JLD-ISG-2012-04, “Guidance on Performing a Seismic Margin Assessment in Response to the March 2012 Request for Information Letter,” all NPPs that have screened in have chosen to do an SPRA. In phase two of the process (which is being refined at the time of this writing), the NRC will use the results of the SPRA to determine if future regulatory action is needed on a plant-by-plant basis.

Additional screening criteria for high frequency exceedance are also considered in both the 50.54(f) letter and the SPID. Addressing high frequency exceedance let to a separate process, led by EPRI, that involved several steps. First, a new shake table-testing program was conducted on potentially high-frequency sensitive equipment by EPRI. This limited the number of types of equipment of concern. Next, a protocol is being developed to assess the impact of potentially sensitive equipment to higher ground motion levels. This protocol is expected to address the inclusion of high frequency sensitive equipment in SPRA.

At the same time, reevaluation of flooding hazard and risk was also being conducted in response to the NRC 50.54(f) letter. Because flood hazard and risk assessment processes have not been developed over the past decades in the way that seismic tools and methods have, the flood safety reevaluation process turned out to be far more problematic than the seismic reevaluation. The flood hazard assessment process is still deterministic in nature, resulting in large uncertainties, a lower assurance of safety overall, and potentially very conservative review flood levels for some facilities. Because flood hazard assessment was not probabilistic, and tools for flood PRA have not been developed to a generally implementable degree, flood PRA techniques could not be applied as part of the 50.54(f) process. This is a significant shortcoming that new tools and methods can address.

Generally, the recent activities have brought to light a large number of technical challenges and shortcomings in the current set of tools and methods available for assessment of NPP safety in light of natural hazards. New tools and methods could create significant benefit for the nuclear industry, while better demonstrating NPP safety and increasing regulatory assurance. This is particularly true if the NRC implements NTTF Recommendation 2.2, which would require reevaluation of risk from natural hazards on a period basis.

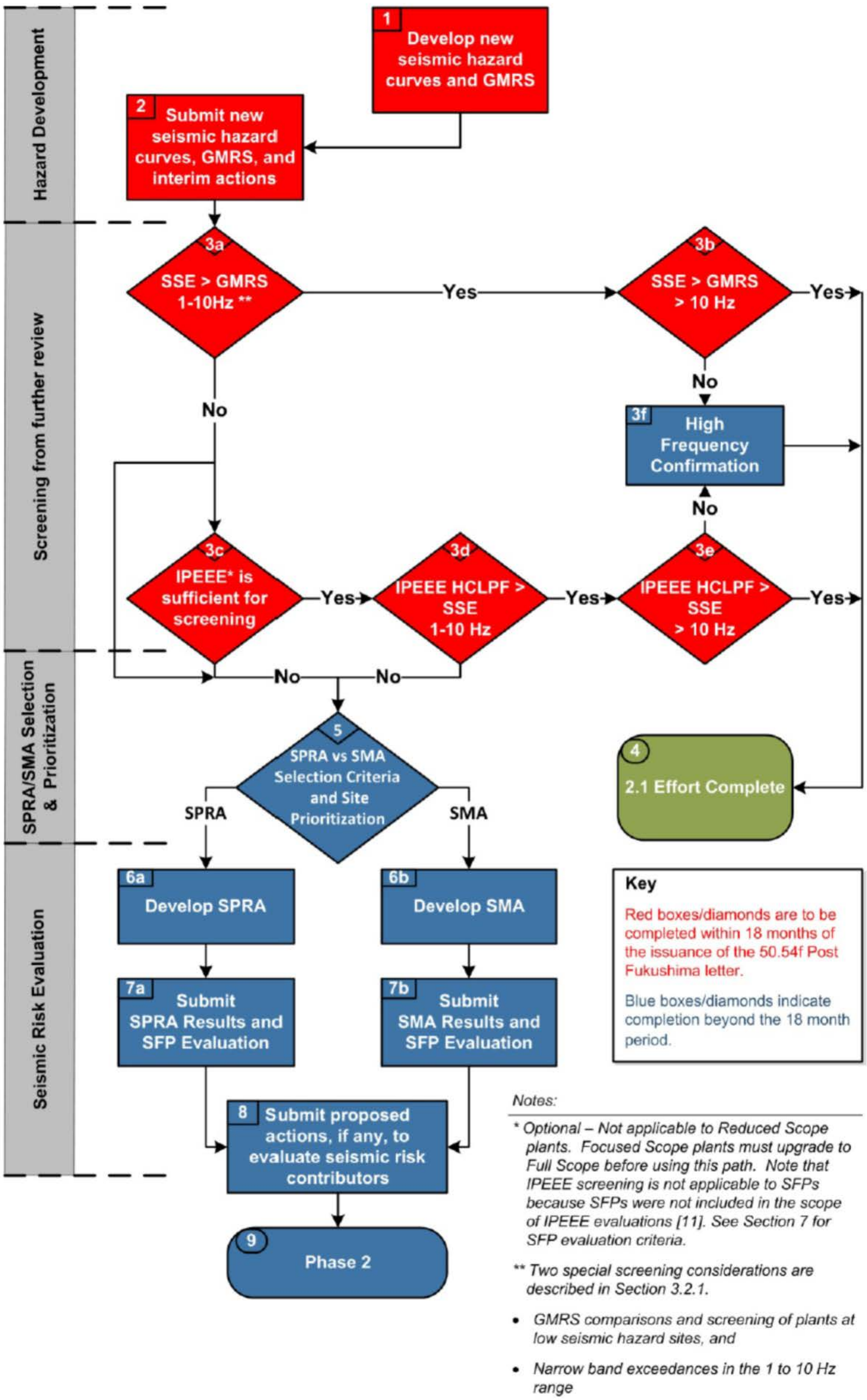

Figure 3. EPRI SPID Flow Chart.

It is recognized that currently, the NRC staff has recommended a plan for closing Near-Term Task Force (NTTF) Recommendation 2.1 on the reevaluation of flooding hazards for operating nuclear power plants.

The NRC staff will likewise use the Commission direction to inform interactions with the industry on guidance to address seismic hazard reevaluations, which is currently following an approach similar to that described below for flooding. The flooding-related action plan identifies two primary activities and one related activity that defines the overall agency response to flooding issues. The two primary actions are:

1. Ensure licensees develop and implement mitigating strategies that are able to address reevaluated flooding hazards, and
2. Complete the flooding hazard reevaluations and close the flooding portion of the 50.54(f) letter, including:
    - Developing a graded approach to identify the need for, and prioritization and scope of, plant-specific integrated assessments, and
    - Developing criteria and guidance to support decision-making related to plant-specific regulatory actions.

The NRC staff will develop probabilistic methods for assessing flooding hazards for future license applications and other NRC activities. The need to complete this activity in the near-term requires that the NRC staff and licensees work efficiently to reach closure on the reevaluation of flooding hazards for each site. The action plan identifies steps that will be taken to reach this closure.

It is noted that the above approach is in agreement with what we are proposing here, in both seismic and flooding context.

2.2 Industry Application Problem Statement

The primary purpose of this report is to define the multi-hazard industry application problem, outline an approach for demonstrating early results with industry participation, and define long term goals.

In 1997, the NRC was issued a commission directive to move towards “Risk Informed” policies. Although the impacts on the Code were limited, and the implementation has been inconsistent, a number of NRC regulatory guides (RGs) and NRC actions and protocols have been issued in response and risk-informed decision-making is now found throughout NRC approaches and protocols. This leads to both opportunities and technical challenges. Although the policy was set to move the decision-making framework forward, the tools and methods available have changed incrementally since that time.

More recently, as a result of the findings and recommendations of the NRC’s Near Term Task Force (NTTF) 2011 report, significant effort is being put into more fully implementing SPRA tools within a regulatory framework. The NRC’s Office of Nuclear Regulatory Research is currently performing a Level 3 SPRA to identify areas in which research may be necessary. If the NRC chooses to implement NTTF Recommendation 2.2, periodic reevaluation of NPPs for external events (e.g., fire and flood) will be required. Based on current activities, External Event PRA would be expected to be a key part of that requirement. The NRC is also actively considering approaches for addressing seismically-induced fire and flood in safety assessments. Because PRA-based implementations if seismically-induced fire and flood not currently exist, it is possible, and perhaps likely, that the NRC will implement a conservative approach that does not make use of risk-informed tools such as those being developed within this program longer-term.

2.2.1 External Hazards with a Multi-Hazard Analysis Problem Definition Approach

The early demonstration that IA2 will solve includes two external hazards, seismic and flood. The flooding at the NPP is caused by both seismically-induced failure of an adjacent levy and seismically-induce internal flooding as a result of pipe breaks within the NPP. The early demonstration will assess the impact of the seismically-induced flooding using the RIMM process. Elements of the process include development of a generic NPP at a generic site, and generic levy and seismic hazard. The problem will assume multiple seismic events that produce ground motion at the generic site. These ground motions will be used to assess the probabilities of SSC failures at the NPP and the adjacent levy. Based on the probabilities of failure on piping systems and of the levy flooding, analysis will be run in those locations. Figure 4 visually shows the problem definition.

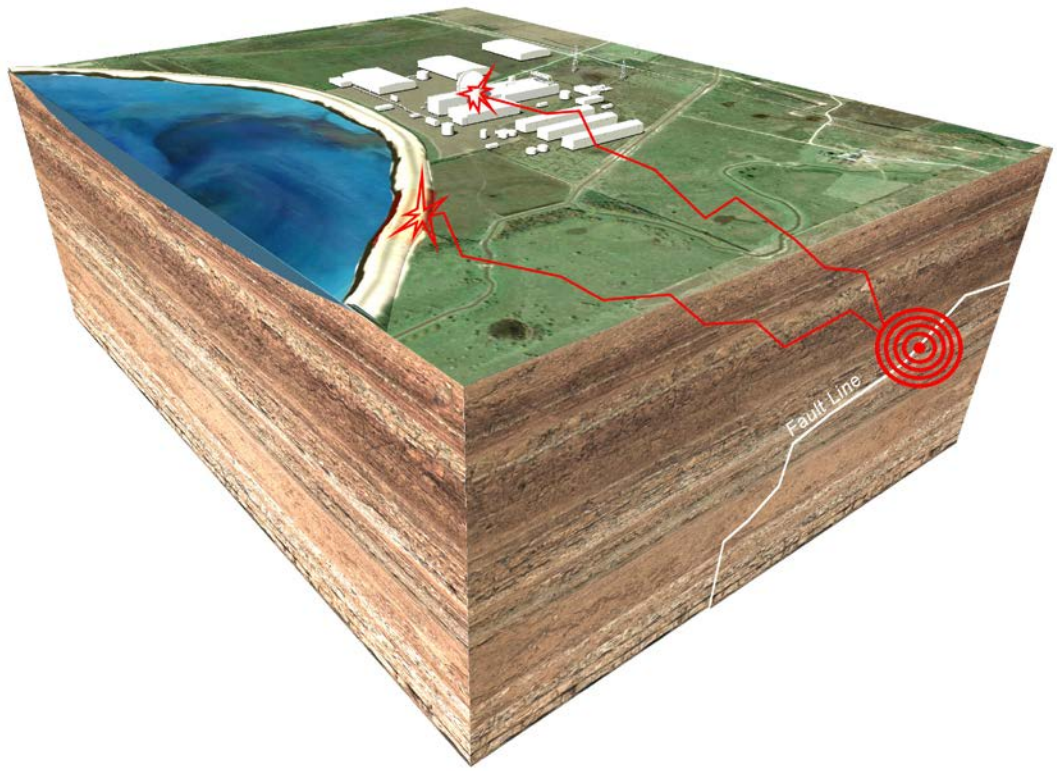

Figure 4. Illustration of the Industry Application External Hazard Analyses Problem Scope.

2.2.2 Seismic Analysis

Nonlinear soil-structure interaction (NLSSI) seismic analysis will be run to determine NPP response during multiple earthquake scenarios. NLSSI will also be performed to calculate dynamic response of the levy. Ground motion input for the NLSSI analysis will be developed from site-specific seismic hazard curves. Hundreds of scenarios, fit to the seismic hazard curve, will be run to determine probability of failure of internal safety class systems and the levy. These probabilities of failure of piping systems and the levy will then drive the assessment of the impact resulting from these secondary flooding phenomena. Discussion of the tools that will perform these analyses is described in the next section.

2.2.2.1 Plant and Site Selection

INL is seeking industry support in applying advanced research and development (R&D) methods and tools to evaluate external hazard risk and decision-making. The seismic portion of the industry application will focus on understanding the benefits of using advanced SPRA methods and tools to perform calculations for actual nuclear power plants (NPP). For the initial activity INL will apply advanced NLSSI to a realistic NPP and system and a realistic soil site.

INL recognizes the importance to industry of separating operational nuclear power plants (NPPs) from research and development activities. Therefore INL is proposing to partner with a NPP owner and use the data provided by the owner such that R&D results will not impact the plant license.

This will be accomplished by using an existing soil site, a NPP not physically sited on the selected soil site, and seismic hazard information (one east coast, and one west coast seismic hazard) that is not related to either the soil site or the NPP, as illustrated in Figure 5.

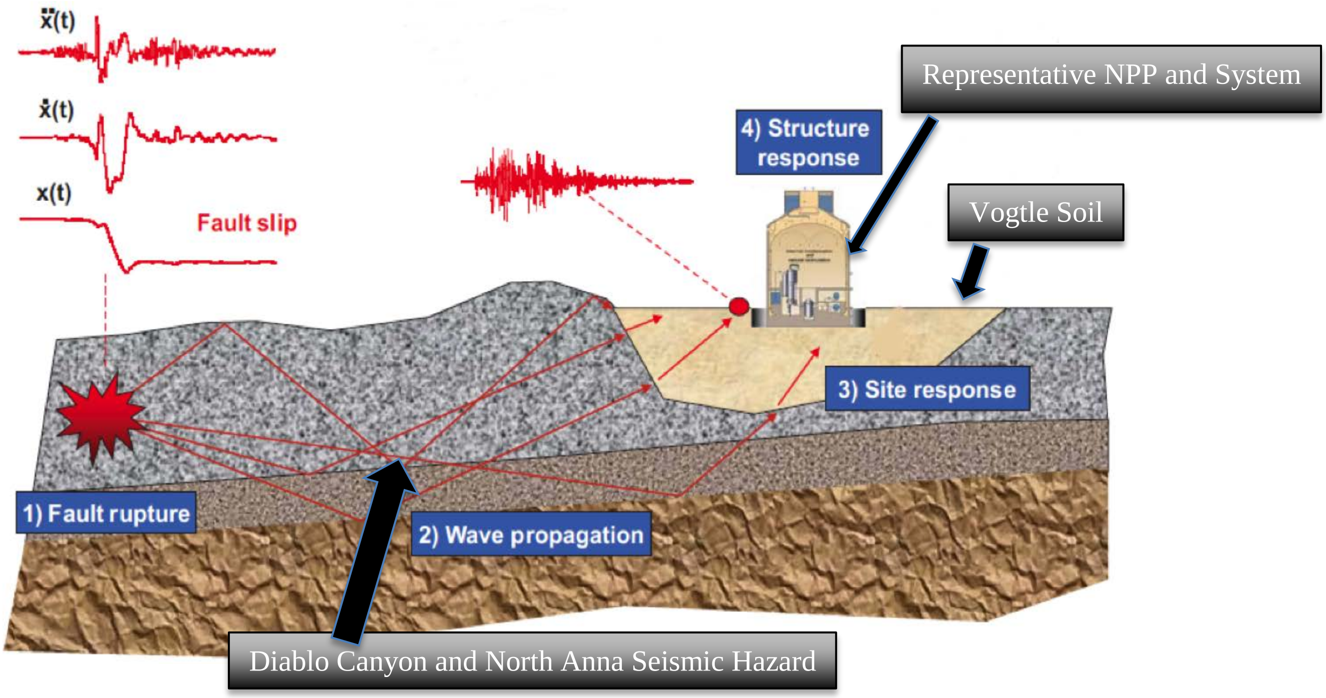

Figure 5. Representative Seismic Model for an Industry Application.

One option for the industry application is to use publically available soil site information for a NPP such as Vogtle, shown in Figure 6. A next step would be to identify industry partnerships to assemble data of a representative NPP and system. Representative ground motions for both a west coast and east coast site would be used to drive the problem. Figure 7 shows potential surface ground motion response spectra (GMRS).

2.2.2.2 Tools and Methods Selection

Probabilistic seismic hazard assessment (PSHA) and seismic probabilistic risk assessment (SPRA) approaches have been applied and improved for several decades and are now considered to be relatively mature in terms of their conceptual development and application. Unfortunately, the tools currently available for SPRA (of which PSHA is a part) are relatively inflexible and were developed principally for internal event probabilistic risk assessments. As a result, currently available tools are now significantly limiting the development of more advanced SPRA methodologies. Development of “next generation” seismic risk assessment tools and methods, which are built upon and expand the RISMC tool kit, would lead to significant improvements in industry’s ability to address regulatory requirements and make the most of regulatory opportunities (e.g., risk-informed relief) related to seismic hazard.

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 some instances the current SPRA approach has large uncertainties, 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). Therefore INL is developing advanced SPRA methods and tools to reduce uncertainties and provide best estimate risk numbers.

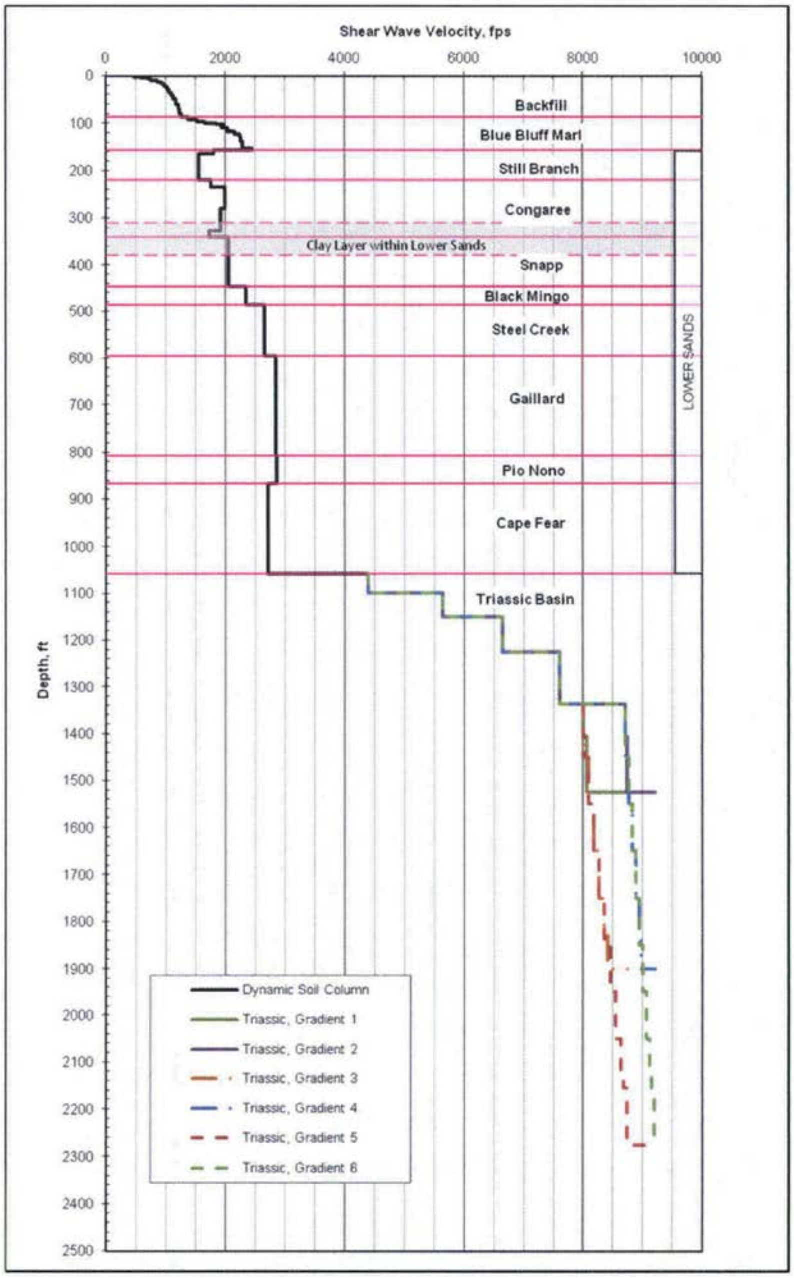

Figure 6. Shear Wave Velocity Profile, Dynamic Soil Column at Vogtle NPP.

The first assumption in SPRAs that may introduce large uncertainty is that NPP response scales linear with increasing ground motion. Initial R&D using nonlinear soil-structure interaction (NLSSI) has shown, for some NPP sites, that this assumption may produce overly conservative in-structure response numbers that are used to calculate core damage frequency. Using realistic models for these NPPs will remove uncertainty associated with NPP response.

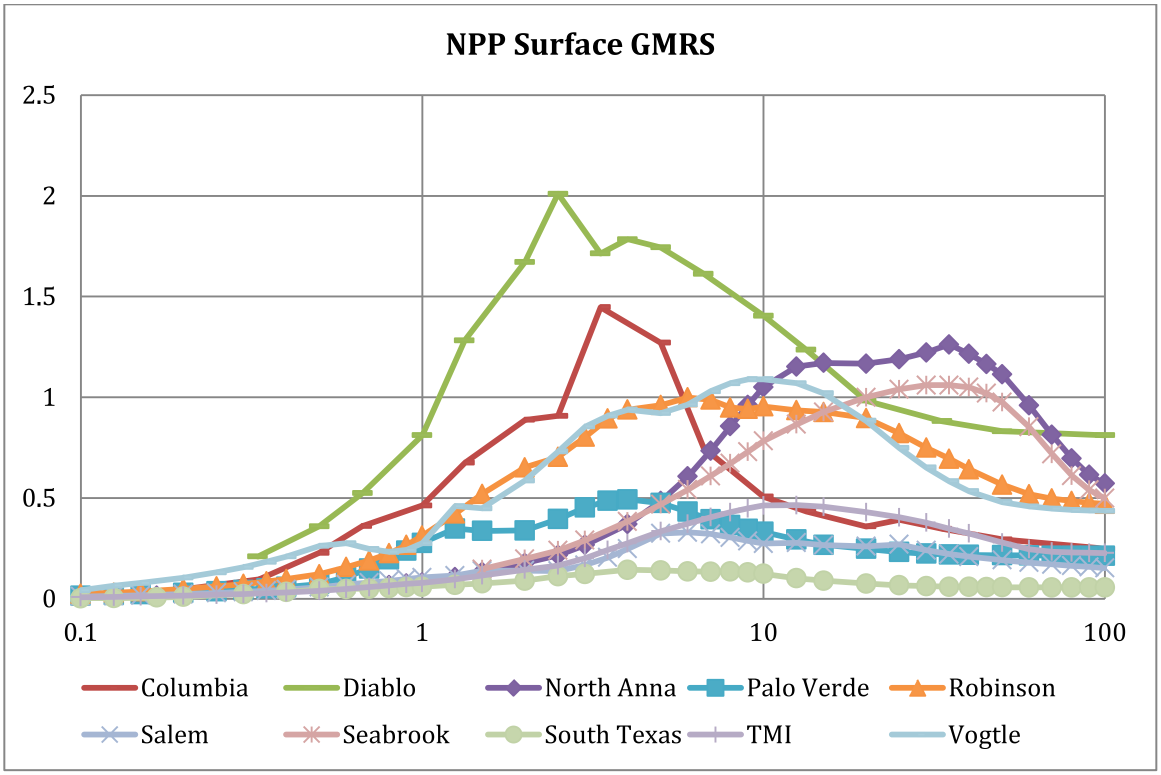

Figure 7. NPP Site Specific Surface Ground Motion Response Spectra (GMRS).

An example of a recent NLSSI analysis shows that the generic NPP (Figure 8) in-structure response is different when calculated using linear and nonlinear SSI codes. The curves in Figures 9 and 10 show a comparison of linear and nonlinear SSI calculations at two different locations in the generic NPP (locations are identified in Figure 8). Figures 9 and 10 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. These plots show the maximum acceleration values on the response spectrum versus the applicable multiple of the site specific Design Basis Earthquake (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.

2.2.3 External Flooding Hazard Analysis

There is also a recognized and growing need for tools and methods to assess risk from other natural phenomena, most notably flooding. Although flooding is an area of focus for the US NRC and industry, in part as a result of the Fukushima accident, probabilistic hazard and risk assessment tools and methods for flooding are relatively crude. Although many elements of the tools and methods developed for seismic hazard and risk assessment can be applied to flooding assessments, significant challenges remain. In particular, the way in which floodwaters and seismic loads impact a NPP once they reach the site is fundamentally different. As a result, flood risk assessment methods must also incorporate robust time-domain physics-based modeling that can provide insight and information on realistic accident sequences, accident progression, and other inputs to both margins-based safety assessment tools and probabilistic risk assessment tools.

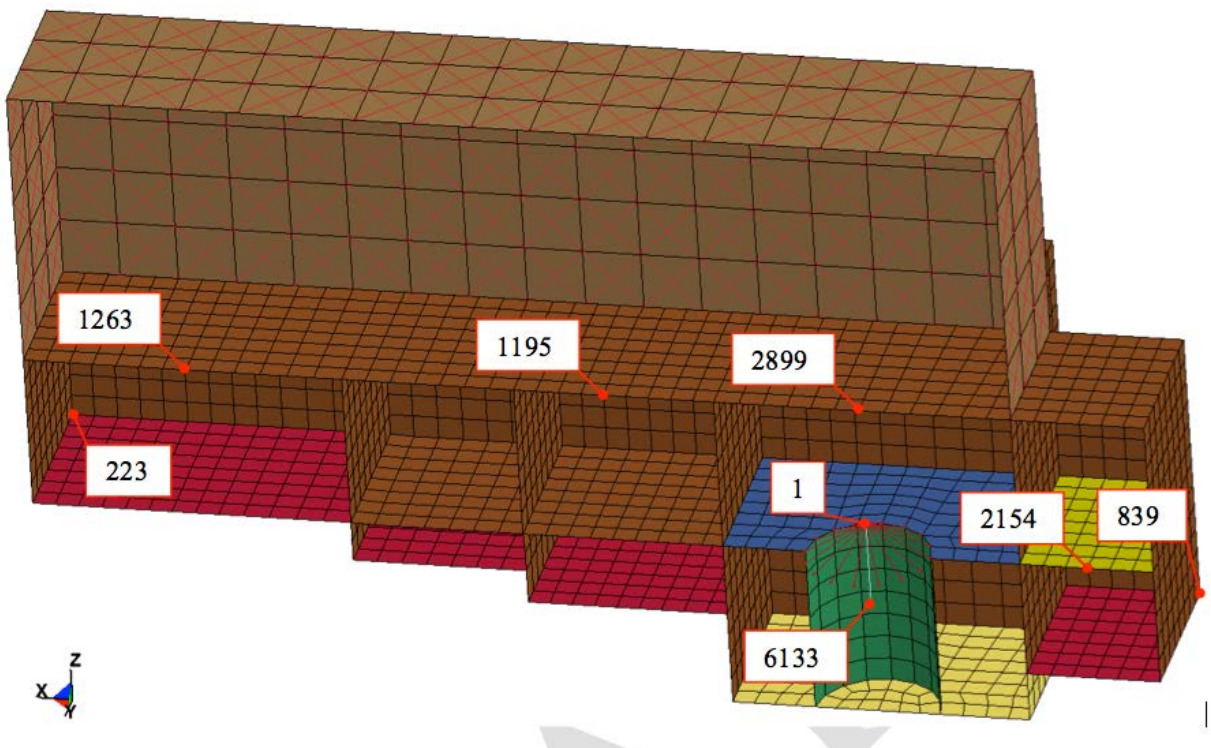

Figure 8. Structural Model of a Generic NPP.

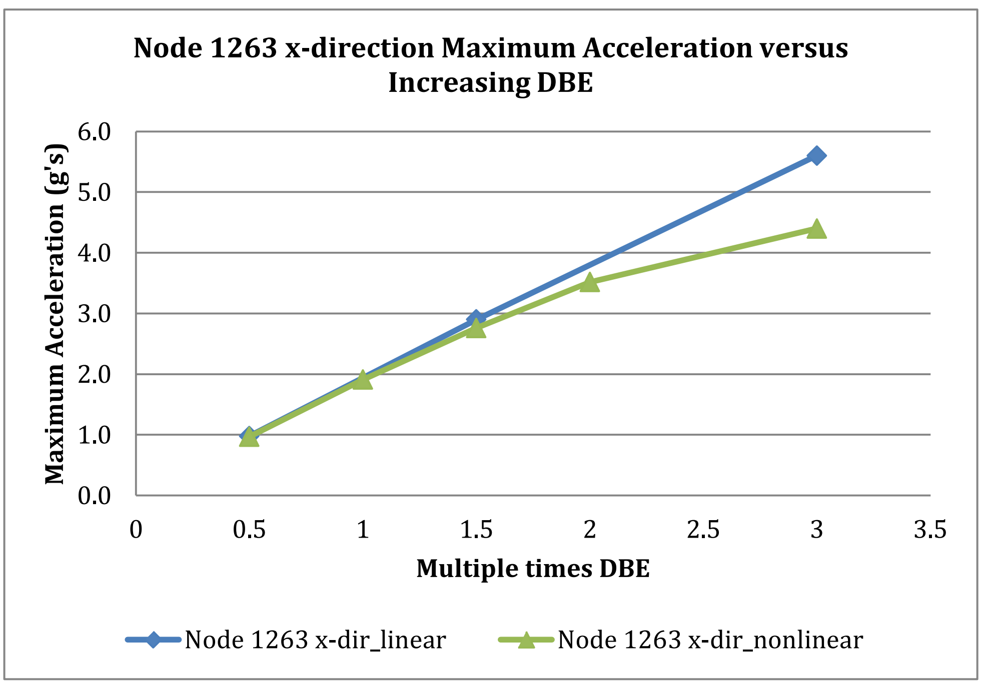

Figure 9. Maximum Response Spectrum Acceleration at Increasing Levels of Ground Motion at INL Site at In-Structure Location, Node 1263.

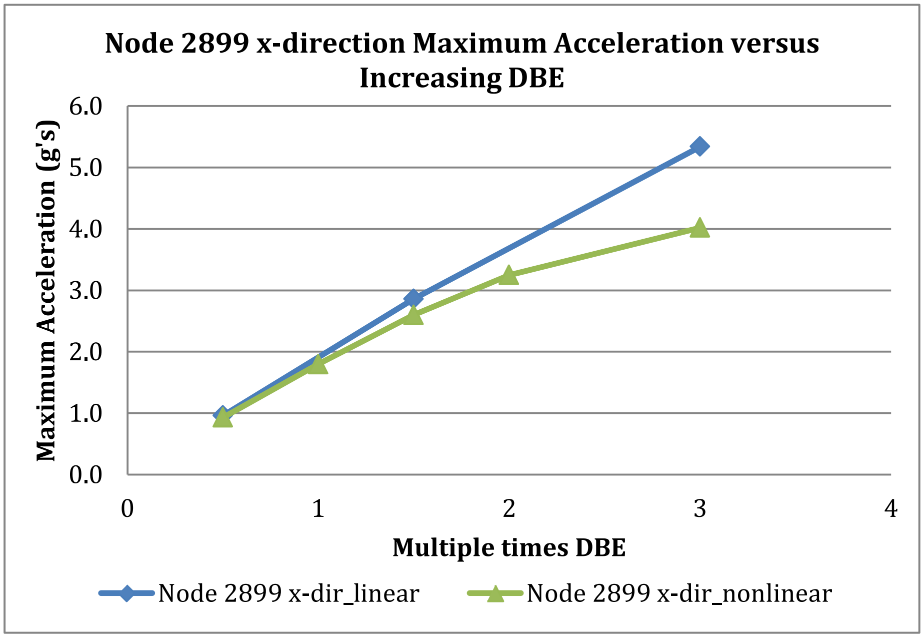

Figure 10. Maximum Response Spectrum Acceleration at Increasing Levels of Ground Motion at INL Site at In-Structure Location, Node 2899.

2.2.3.1 Realistic Plant and Site Representation

When simulating accident scenarios as part of RISMC, we may require multiple physics-based modules that must be run for one or more scenarios directly as part of the analysis. A subset of these simulation modules might be run "offline" and their results stored in whatever format is native to that particular application for retrieval during the analysis. Alternatively, we may be able to translate these mechanistic calculations into what are called “emulators” (or reduced-order models) wherein the emulator mimics the more complicated analysis but is able to run orders of magnitude faster. Let us describe a possible approach that would be used for a realistic plant representation to better understand how physics-based simulation is used in in Industry Application.

First, we need to construct a model representing the topography of the site (and surrounding areas) and various structures at the NPP. An example of this 3D model is shown in Figure 11. Then, as part of the simulation, we are going to represent a flooding event (which occur stochastically and with different magnitudes) and look at implications to the on-site structures and follow the path of the water.

For a given flood that is simulated in the virtual NPP model by the RISMC Toolkit, we query the results of the physics related to the water. The simulation then continues by translating the physics-based mechanistic calculation into an impact in the accident scenario (see Figure 12). For example, if the structure is cracked due to hydrostatic pressure, this state would be applied to the component in the model (perhaps it is a wall or a pipe) using another stochastic model (in this case, a cracking model). Once the component state is specified, then the scenario would continue since the cracked component may experience a dislocation (the crack grows) or further damage. If the component were a pipe containing water, then we might experience additional flow out of the pipe at the point of the crack.

While the special interactions are being represented in the 3D environment, the accident scenario generator continues since water flowing from the leak may (later in time) fail collateral components (say a pump in the same room). Further, there may be other components in that room that are sensitive to the water, for example the pump motor controller which is an electronic component.

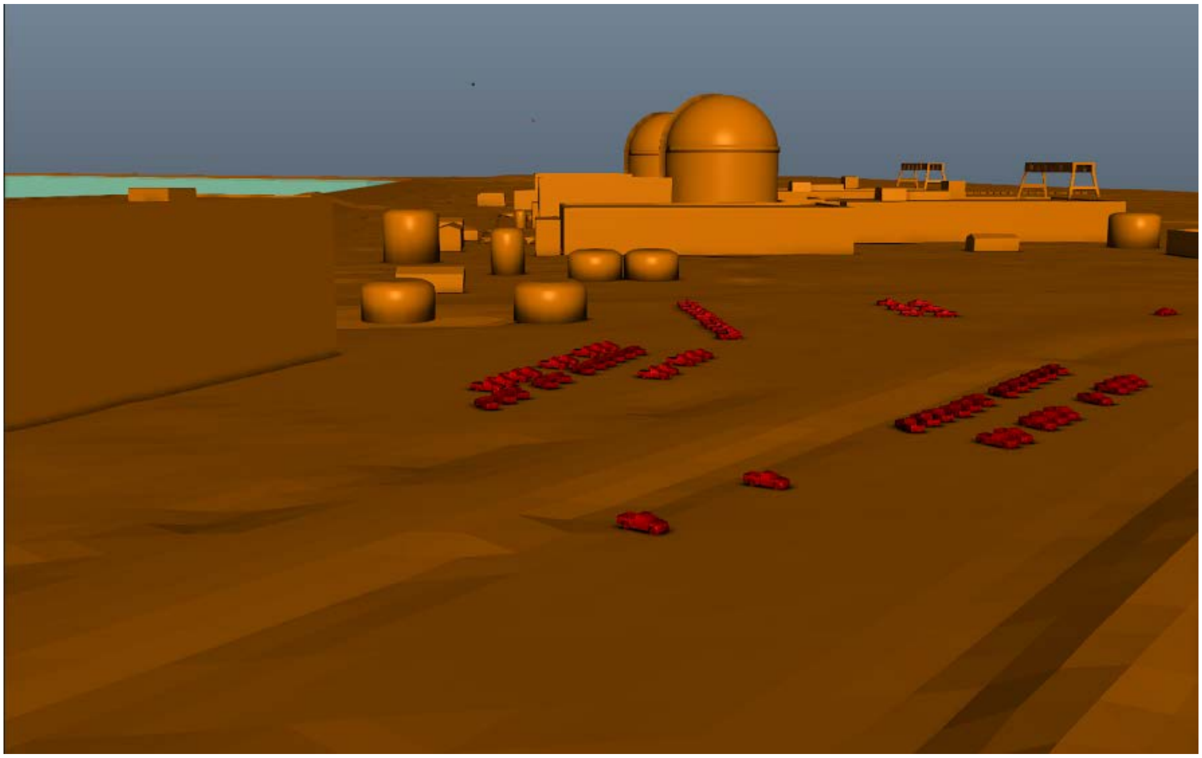

Figure 11. Example of Site Topography and 3D Models to Be Used for the Flooding Simulation.

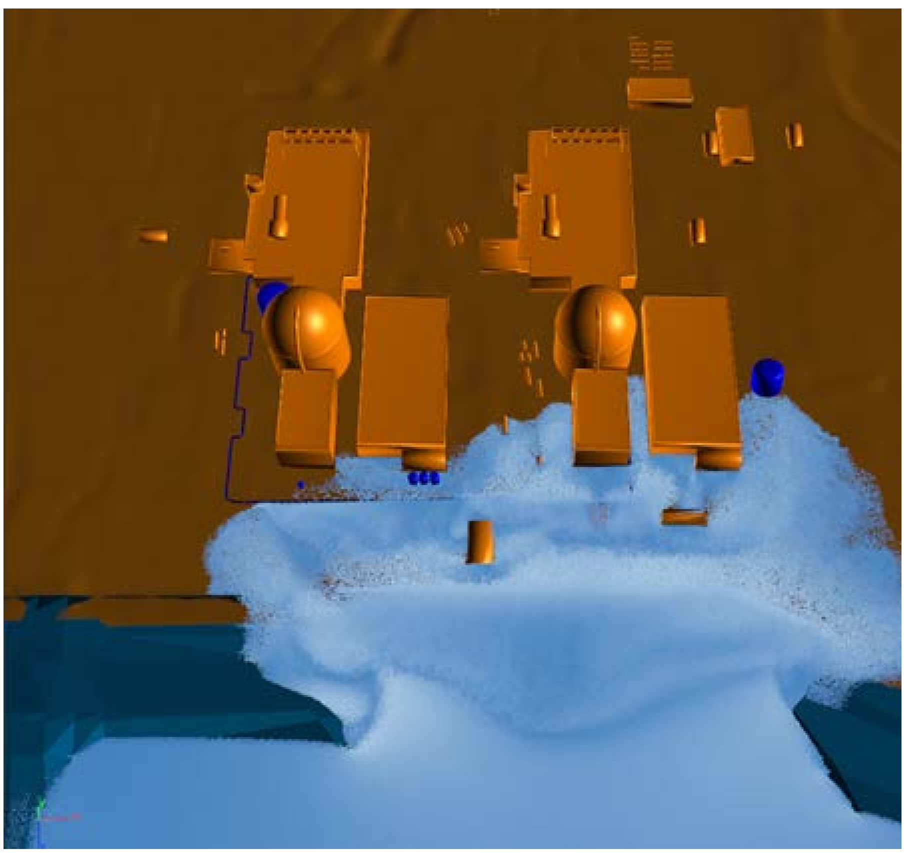

Figure 12. Visualization of a Flooding Simulation.

3. INDUSTRY APPLICATION EARLY RESULTS

In this section we show separately, preliminary results for each of the external events considered for the industry application, seismic and external flooding. Combining the tools developed for each event with a risk-informed methodology encompassing multi-hazard analysis will assist the stakeholder(s) in decision making for future plant modifications or improvements, if necessary.

3.1 Seismic Initial Results

Early SPRA results that compare a traditional SPRA calculation with an advanced SPRA calculation are discussed below. The focus is on implementation of NLSSI into the SPRA calculation process when calculating in-structure response at the area of interest. The NLSSI initial calculations are presented in INL/EXT-15-35687. Two specific nonlinear effects included are localized soil nonlinearity and gapping and sliding. Other NLSSI effects are not included in the calculation. The commercial software program, LS-DYNA, is used for the NLSSI analyses.

The results presented document initial model runs in the linear and nonlinear analysis process. Final comparisons between traditional and advanced SPRA will be presented in future work.

As discussed previously in Section 2.2, we use a generic Nuclear Power Plant (NPP) structure and a generic system and perform linear and nonlinear probabilistic response analysis using:

- Linear SSI (CLASSI, frequency domain)
- Nonlinear SSI (LS-DYNA, time domain)

Component fragilities are developed per EPRI TR-103959. Probability of system failure is calculated for an emergency cooling pump system. 30 spectrum matched time histories at a return period of 1e-4 are used to compute the linear probabilistic SSI results. These results are then scaled linearly for increasing levels of ground motion.

Nonlinear SSI results will be generated by developing responses at three ground motion scale factors (note the linear SSI used just one and then assumes that ISRS scales linearly with increasing ground motion). The same fragilities calculated for the linear analysis are used (note this will be changed in future analyses). ISRS are generated at each ground motion level. The capacity distributions are independent of ground motion level.

In both the linear and nonlinear SSI the probability of system failure is computed by convolving system conditional failure probability with the seismic hazard.

3.1.1 Generic Structure

The study considers a generic 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. Additional details on this study are given in INL/EXT-14-33222. 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 pre-stressed concrete containment structure and reinforced concrete internal structure. The structure and its stick model representation are shown in Figure 13.

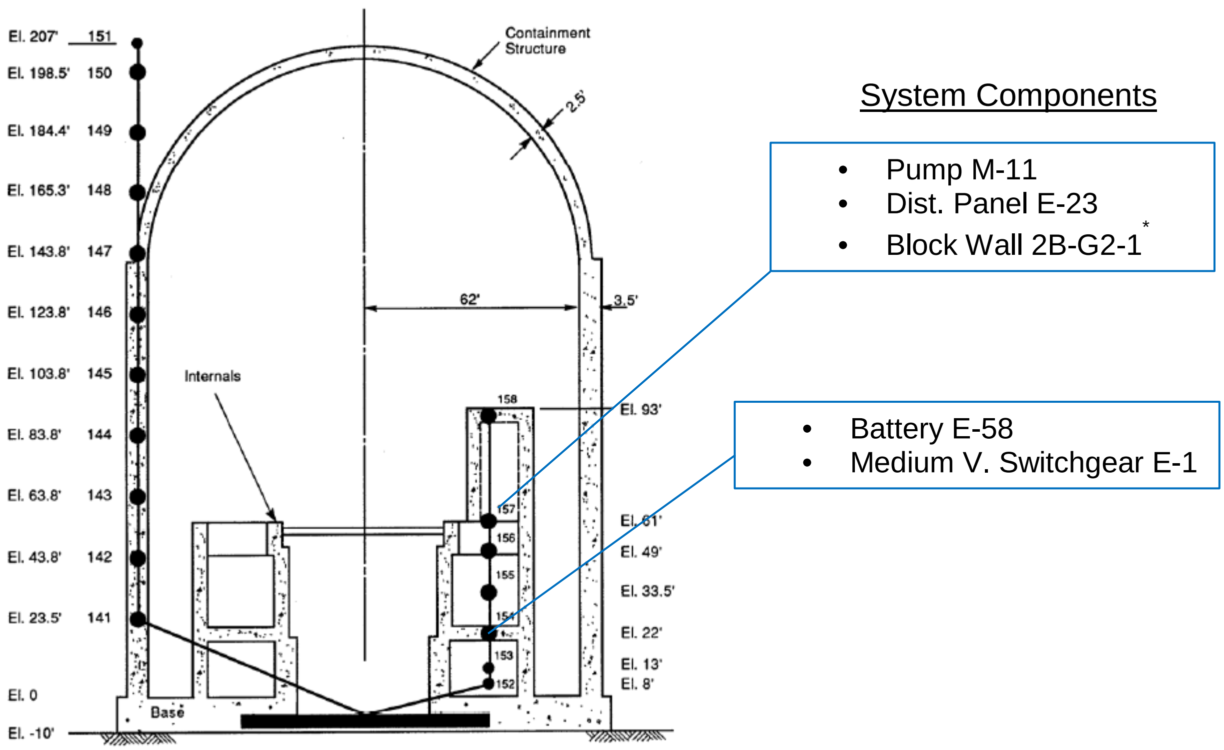

Figure 13. Generic NPP and System.

Probabilistic response analysis was performed using the Latin Hypercube Sampling (LHS) approach for thirty simulations and models run in the linear computer program CLASSI. Preliminary fragility results based on the linear (Traditional) SPRA analysis are provided in Table 1.

Table 1. Linear SSI Fragilities.

Component	Floor	A_m	β_c	HCLPF
Pump 670-M-11	EL 61’	2.70g	0.45	0.95g
Battery 670-E-59	EL 22’	1.14g	0.28	0.59g
Dist. Panel 670-E-23	EL 61’	1.60g	0.59	0.40g
Block Wall 2B-G2-1	EL 61’	0.60g	0.28	0.31g
Switchgear 670-E-1	EL 22’	1.90g	0.47	0.64g

Preliminary nonlinear analysis results (Figure 14) show that at low levels of ground motion the linear and nonlinear models produce similar results (Figure 15).

Preliminary NLSSI analysis demonstrates that a functional NLSSI model has been developed that includes 1) local soil nonlinearities at the foundation and 2) geometric nonlinearities. This NLSSI model will be run multiple times at increasing levels of ground motion to generate in-structure response spectra that will be input into the advanced SPRA calculations. Comparison of traditional SPRA and advanced SPRA results will be provided in future work.

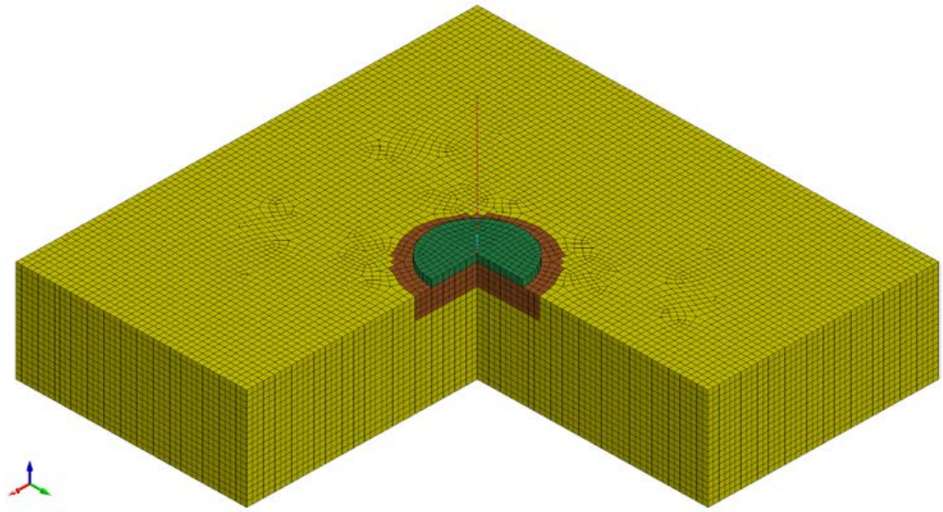

Figure 14. Section View of The Nonlinear Model Illustrating The Nonlinear Soil (Brown), Linear Soil (Yellow) and Basemat (Green).

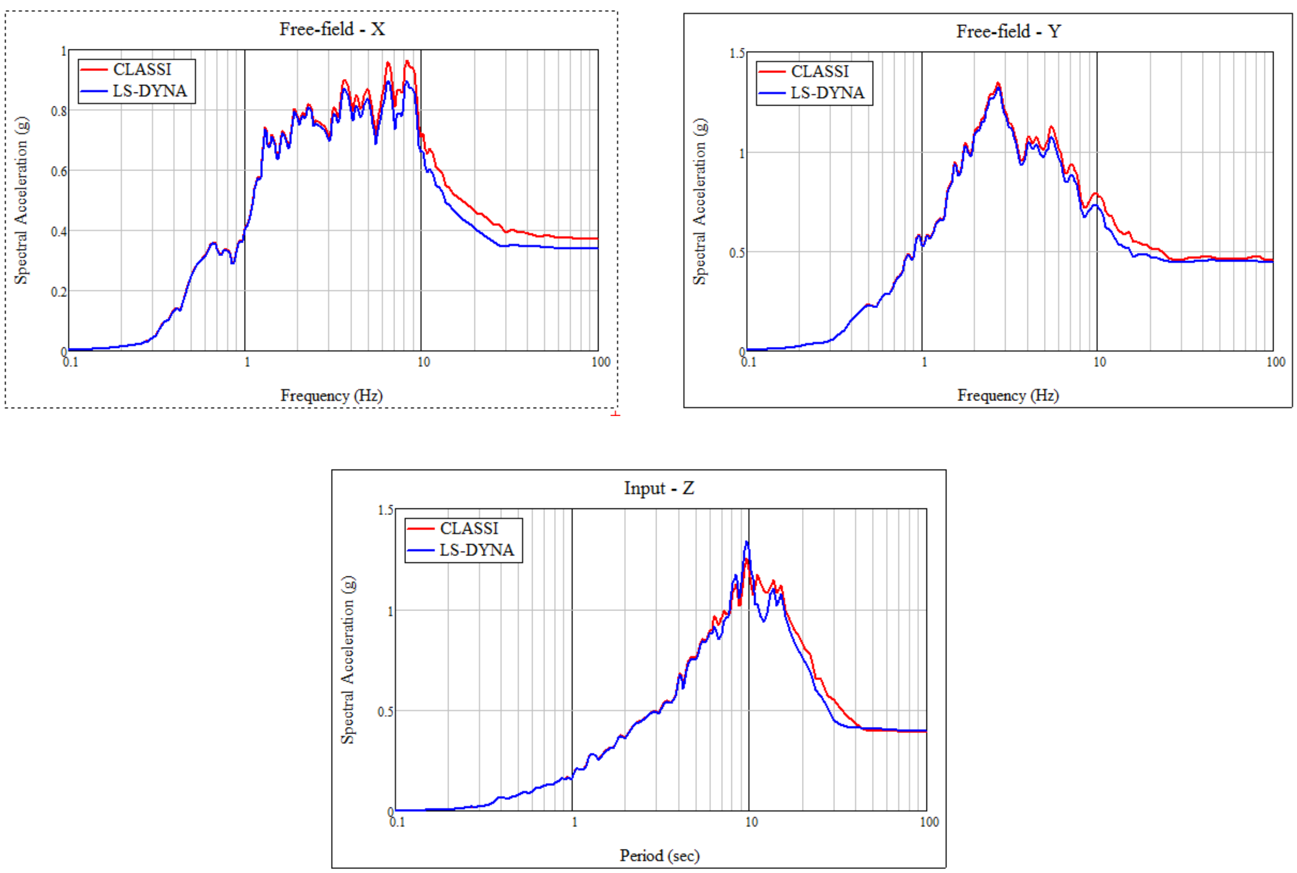

Figure 15. 5% Damped Response Spectra of The Free-Field Input Acceleration in CLASSI and The Free-Field Acceleration Calculated Using LS-DYNA.

3.2 External Flooding Initial Results

3.2.1 Primary Flooding Model

All simulations are done on hypothetical facilities for example purposes only. For this example problem, we are using the terrain with a dike holding an above grade cooling pond for a NPP. The terrain map was generated using a previously developed tool that queries altitude data based on existing topography to generate a polygon model. The web based terrain mapping tool can be used to generate a low detail map of most land areas on the earth (see Figure 16).

Well known seismic and geological data from a location has been selected, which will make the facility and earthen dam retaining the reservoir susceptible to seismic events. Internal building features are again from other locations or facilities.

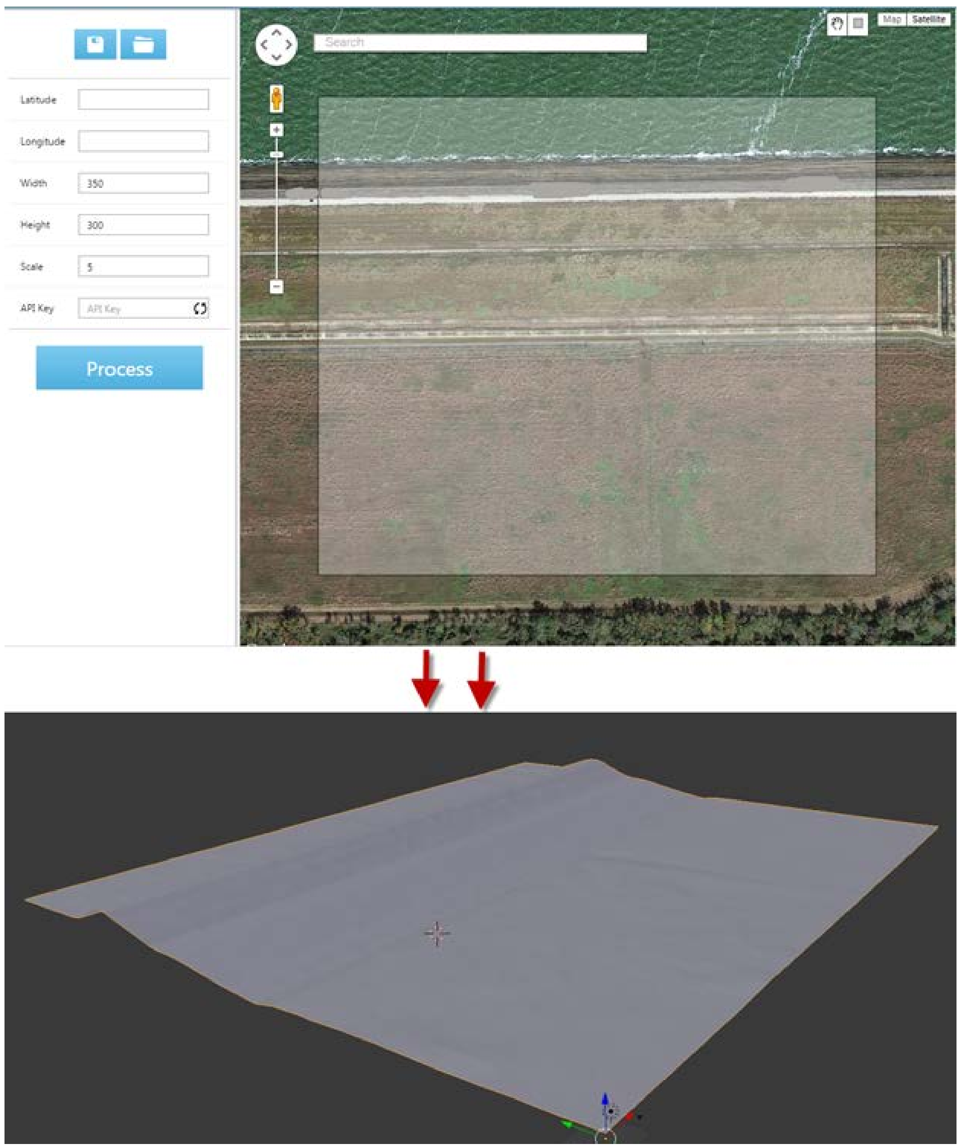

Figure 16. Terrain Map Construction Using Web-Based Mapping Tool.

There are two options to incorporate the seismic triggered flooding into the simulation:

- Option 1. Use a static boundary terrain and dam model for the simulation. Then use seismic data in sampling to determine size and location of the seismic event. With the event data, use formulas and software to calculate effects on the dam such as breaches and an erosion model for fluid output flow rate. Existing software algorithms such as WinDAM can provide estimated flow rates over time. Finally, the 3D simulation with dynamic fluid partial emitters can simulate the output from the breach over time for the given location.
- Option 2. Construct the dam in the simulation from dynamic particles which hold back a reservoir of water particles. Next, sample for a seismic event and simulate that event in the 3D environment. Then let the 3D simulation determine any fracturing, breach, or deformation of the dam. Erosion and debris from the erosion are simulated through physics and become part of the resulting interactions.

The second option is more ideal because it would be a simpler interface and fewer steps, but requires more coding, integrated features, and validation methods. Using first option will allows for testing of the main process and reduces overall complexity by breaking it into separate pieces. For this example the first method is being developed.

3.2.2 Secondary Flooding Model

Internal flooding can be caused by two ways, seepage/entry through penetration locations, and/or a pipe rupture caused by the seismic event. To construct a model for this example, a 3D scanner was used to generate a point cloud representation of a room with similar components and features of what could be in a NPP, but is not from a NPP (See Figure 17).

The second option is more ideal because it would be a simpler interface and fewer steps, but requires more coding, integrated features, and validation methods. Using first option will allows for testing of the main process and reduces overall complexity by breaking it into separate pieces. For this example the first method is being developed.

In order to use the model for simulation, this point cloud must be converted into a sealed polygon mesh and critical components identified and configured. Using a combination of point cloud and polygon processing methods, existing structures can be scanned and much of it auto converted into more usable models by reducing data size, identifying objects, and validating the mesh. This reduces the manual and expensive labor time needed to generate viable models for simulation (see Figure 18).

When the simulation is run, flooding information from the main simulation is saved and then used in this simulation to generate flow rates through penetration points. Alternatively, a pipe rupture will also be able to be triggered by seismic activity causing flooding from within.

3.2.3 Component Failures

Two types of component failures can occur for the demonstration simulations; water contact and debris impact force. In the simulation model there are numerous free form items such as cars, crates, or barrels. These items move according to the simulated physics as they interact with the fluid particle flow. When debris comes in contact with a component, the force exerted on that component is calculated and returned to the PRA methods used. Water contact and pressure, such as on doors, is also monitored and the data returned.

Figure 17. 3D point Cloud Model to Be Converted into a Polygon Mesh and Used in Internal Flooding Simulation.

Figure 18. Polygon Mesh Constructed From The Point Cloud Shown in Figure 17.

3.2.4 Current Development

3.2.4.1 Debris movement and impact measurement.

The simulations use Smooth Particle Hydrodynamics (SPH) based physics solver for fluid movement and interaction with objects. This allows for simple force transfer from the fluid particles to object particles. SPH works by obtaining approximate numerical solution of the equations of fluid dynamics by representing the fluid with particles, where the physical properties and equations of motion of these particles are based on the continuum equations of fluid dynamics. Physical quantities are estimated by interpolating existing fluid quantities using the neighboring particles (see Figure 19). SPH has an advantage over grid based simulation methods because SPH guarantees conservation of mass. Second, it calculates weighted pressure automatically from neighboring particles rather than solving linear equations. Recent SPH development has allowed for incompressible particles and increased accuracy in viscosity and vortexes.

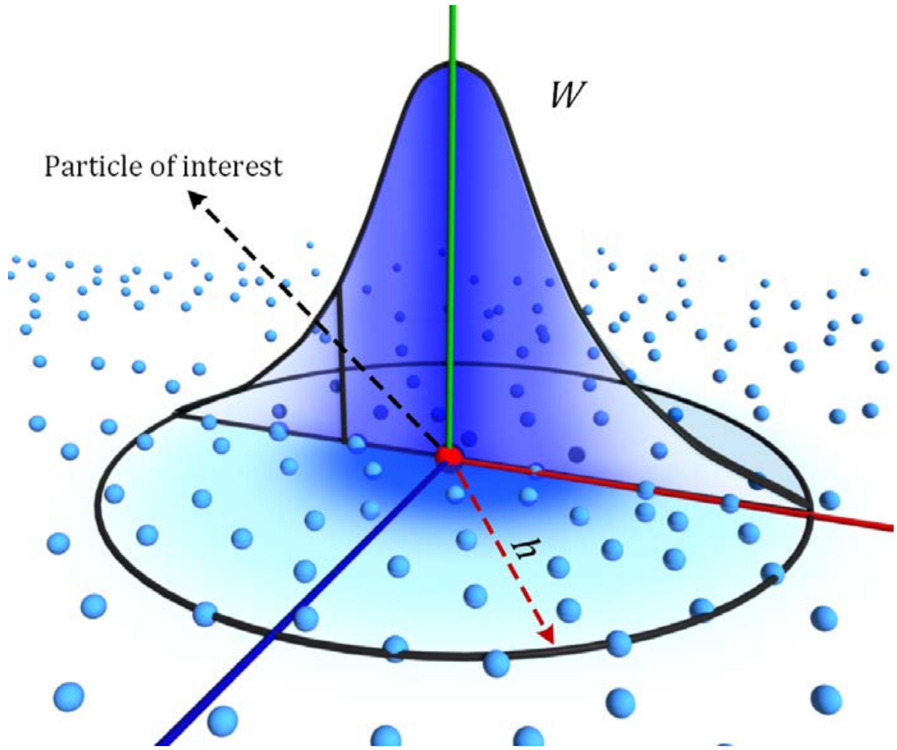

Figure 19. Illustration of SPH Approximation for a Field Variable for the Particle of Interest.

In the testing simulations rigid body interaction occurs when the flood water from the dam failure impacts vehicles and other debris. Current development is being done to calculate the maximum and average force from both debris and water impacts. Object movement depends on the properties of the object set in the simulation. Just as in real world interactions, if the force applied from the fluid, overcomes the given mass of the object, it will shift and move accordingly (see Figure 20). This data is then fed back into the PRA model to be used in determine component failures. Data from both simulated seismic effects and 3D flood effects are used to determine component failure. For example a retaining wall or door affected by a seismic event will have a degraded failure model and thus less able to protect against a subsequent flood impact.

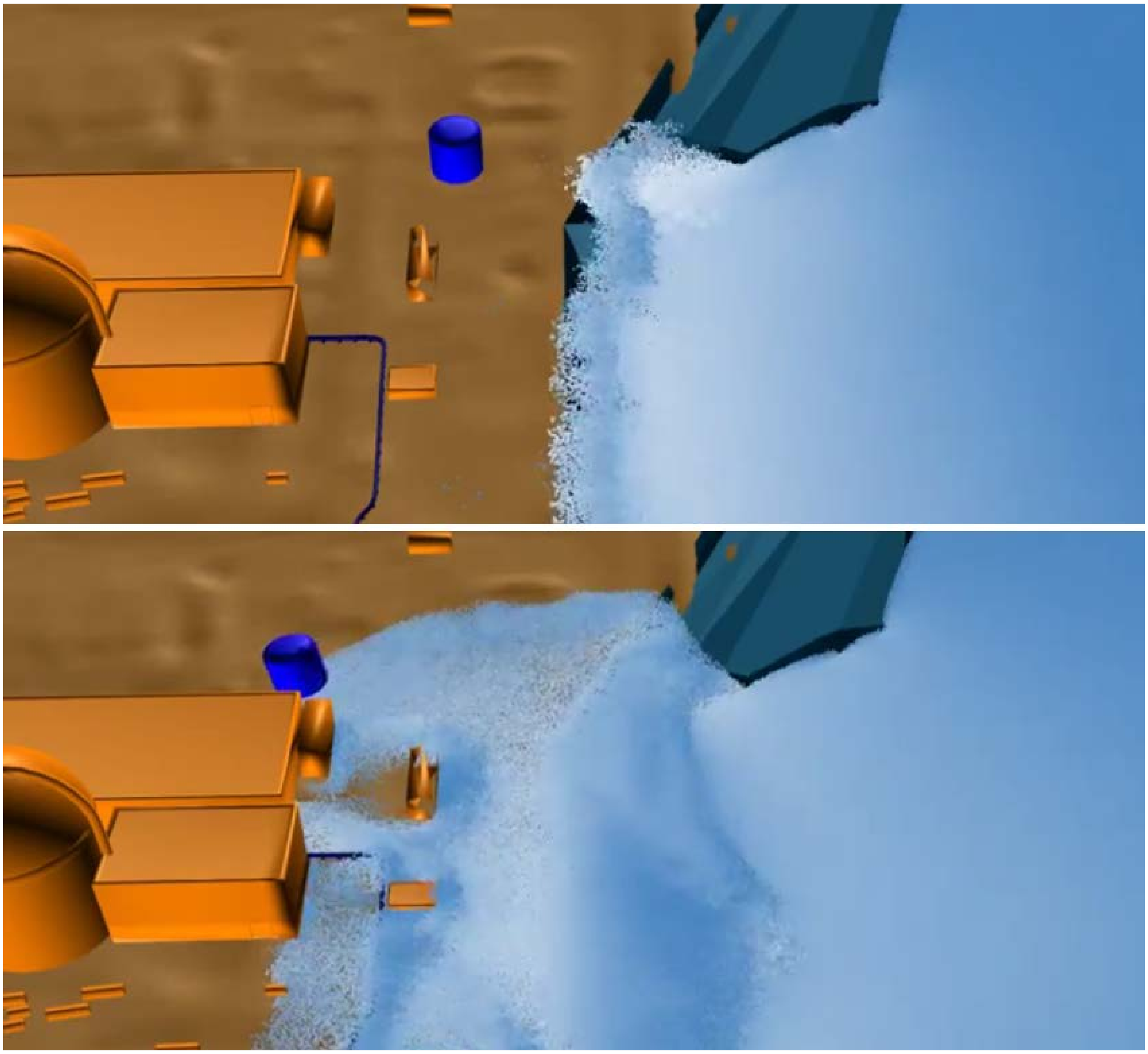

Figure 20. Example of Rigid Body Object Movement From Fluid Forces.

3.2.4.2 Dynamic particle emitter to simulate an erosion model.

Work is currently being done to produce a dynamic particle emitter to match a dam overtopping model. This enables us to simulate a large scale event without simulating an initial large body of water, which reduces runtime and simplifies the setup process. The particle emitter will be able to change shape over time and automatically adjust its flow rate for a give body of water height. By coupling the particle emitter to an erosion model, simulation time can drastically be reduced because the large number of particles making up the source body of water does not need to be simulated. Without using this method, a simulation with no breach would require the same time as a large breach, whereas with it, simulation time is a function of the number of particles created over time to simulate the breach (see Figure 21).

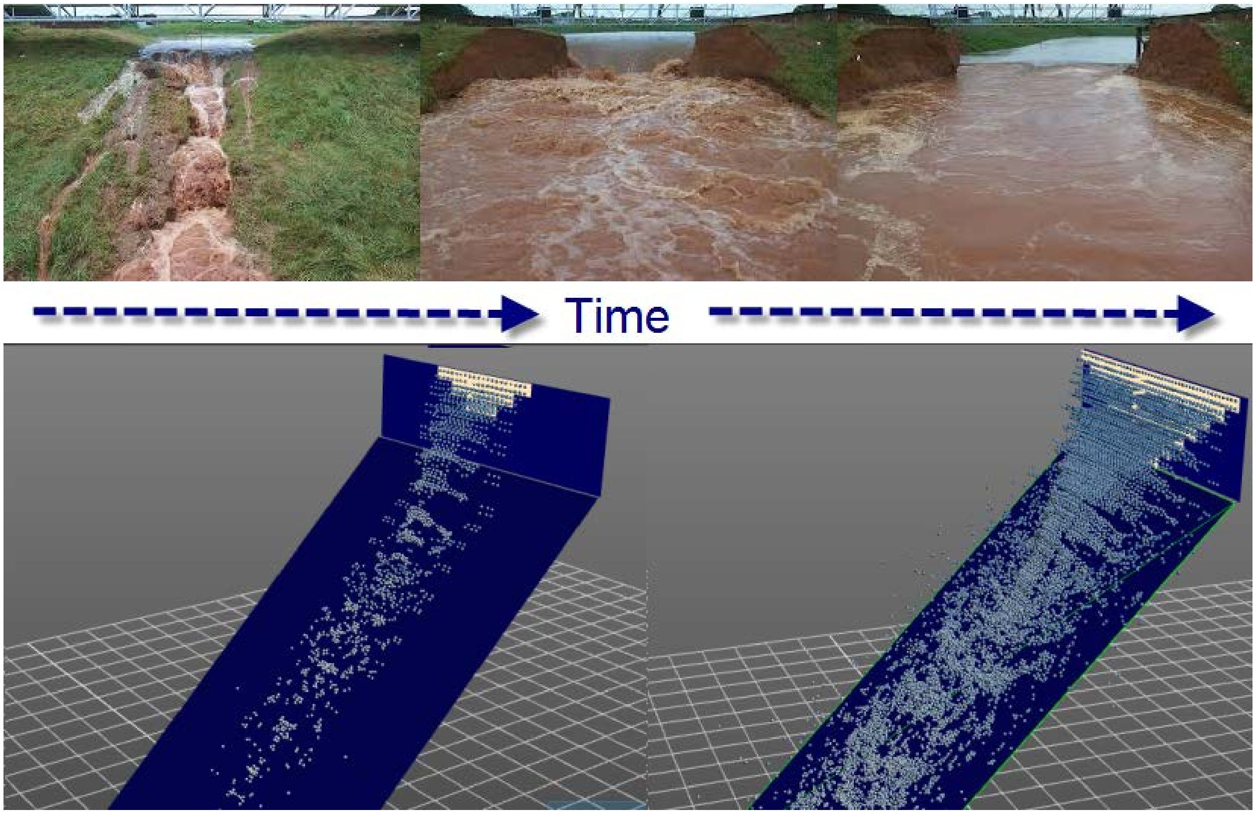

Figure 21. Dynamic Particle Emitter to Simulate a Dam Breach Flow Over Time.

4. INDUSTRY APPLICATION FULL DEMONSTRATION – NEXT STEPS

Phase 3 of the Industry Applications considers the full spectrum of demonstrations with all advanced features of the RISMC toolkit (concurrently in development while early RISMC demonstrations take place), including Verification, Validation, and Data Analysis. These will include applications of RAVEN and RELAP-7, also including other models in the MOOSE framework, as needed.

Below are discussions of a few ideas that will be explored in FY2016 and beyond.

4.1 Advanced SPRA Approach

Seismic probabilistic risk assessment (SPRA) methods and approaches were first developed in the 1970s and aspects of them have matured over time as they were applied and incrementally improved. SPRA provides information on risk and risk insights and allows for some accounting for uncertainty and variability. As a result, SPRA is now used as an important basis for risk-informed decision making for both new and operating nuclear facilities in the US and in an increasing number of countries globally.

The first generation of tools and associated methods were limited by insufficient a) computational capabilities, b) knowledge of earthquake ground motion and the interaction of soil-structure systems, c) tools for nonlinear analysis of soil-structure systems, d) knowledge of component fragilities and their correlations, e) tools for directly analyzing external event probabilistic risk (including the tracking of uncertainties throughout the calculations), f) and tools and methods for calculating uncertainty in systems models. Although existing tools and methods have proven very beneficial over time and arguably have improved nuclear safety in the US, these same tools now represent an impediment to advancement of SPRA approaches.

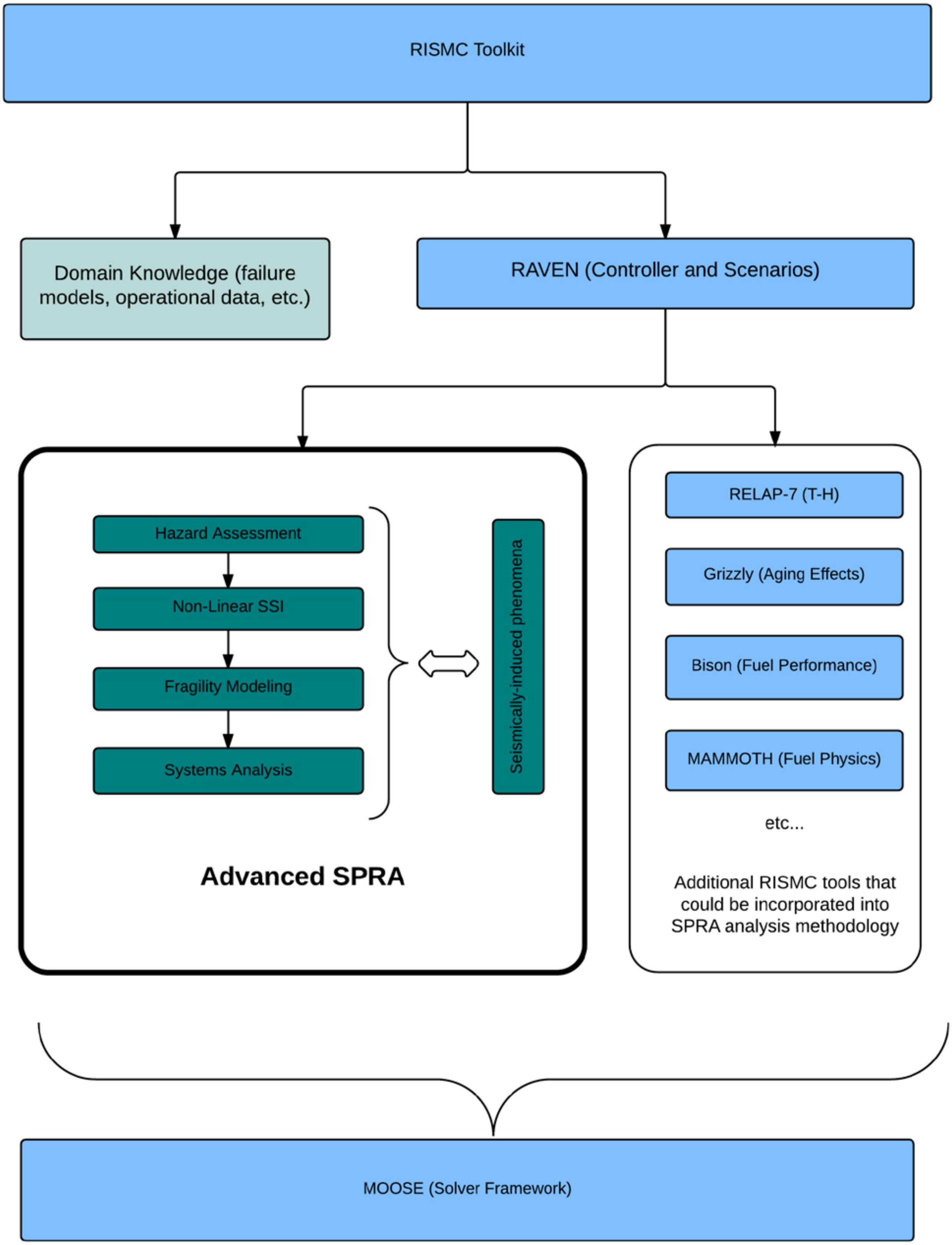

Figure 22. Integration of Advanced SPRA in the Larger RISMC Program.

The Advanced SPRA (ASPRA) development program discussed here will develop a new set of tools and methods within the RISMC program to perform Advanced SPRA. These tools and methods would be implemented within the MOOSE solver framework and would make use of existing and newly developed tools and methods, coupled with the experience and data gained in the past decades, to define and analyze more realistic risk assessment models.

The steps in SPRA are shown in Figure 22, along with their relationship to other RISMC elements. External event PRA is composed of three general elements: hazard assessment, fragility relationships, and systems analysis. SPRA also has the element of soil-structure-Interaction analysis, which couples the rock hazard at the sites to the in-structure motions experienced by the systems and equipment within the NPP. The fragility of the structure itself is also important for assessment of the potential for early release into the environment. The new tools and approaches developed in this project cover the many steps in a SPRA in a more cohesive approach that could reduce interface issues and more accurately track uncertainties throughout the process. The methods developed would move away from the use of peak ground acceleration to incorporate parameters of most significance to response to earthquake ground motions. By tracking uncertainties more seamlessly and rigorously throughout the process, and using physics-based tools to investigate scenarios of interest that have traditionally been left out of SPRA (e.g., seismically-induced fire and flood), the new tools would provide more accurate models with a clearer view of uncertainties.

Development of a set of tools and methods to replace the existing SPRA is the first focus area of a multi-phase project. Focus area 2 would also develop new tools to address two important areas of current research in SPRA, namely seismically-induced fire and flood. Focus area 2 feeds into the tools created in focus area one by developing methods and protocols to use various physics-based dynamic tools available in the RISMC toolkit to investigate issues and uncertainties in the systems model for facilities being analyzed. Activities following these first phase of activities would identify areas in which efficiencies are found and/or further developing methods based on ongoing use of the tools and methods.

4.1.1 Industry Needs

The lack of up-to-date SPRA tools, particularly as coupled with recent post-Fukushima activities, has meant that tools and methods available to industry are not fully able to meet current and future needs. Additionally, available methods have a large number of conservatisms that could be reduced or removed using up-to-date tools and methods. Current methods also often use plant-level performance objectives on a SSC-level, which results in conservatisms in design. This could result in a lower estimate of facility risk.

Current needs generally relate to the response of the US NPPs that have screened into performing additional risk calculation as part of the NRC's 50.54(f) Request for Information letter. The EPRI SPID report covers specific topics in SPRA where there was a lack of guidance or clarity. One of the topics covered relates to when existing structural models can and cannot be used. The issues that have arisen as a result would benefit from new NLSSI tools and methods.

The reduced accuracy of SPRA modeling that results from current tools and methods, the cumbersome and inefficient nature of current tools and methods, and the lack of consistent characterization and tracking of uncertainty throughout the process means that significant improvements could be realized to benefit the ability of industry in a variety of ways. Using these tools, NPP would be able to better manage their risk profiles, to support risk-informed license amendments, to address analysis needs when new hazard information (including external events other than seismic) is identified, and to provide tools to rapidly assess the impact of identified plant issues, such as those examined in the Reactor Oversight Program (ROP) Significance Determination Process (SDP).

A number of long-term challenges also exist. The NRC's Near Term Task Force (NTTF) Recommendation 2.2 would require that reevaluation of hazard and, if necessary, risk is reevaluated periodically (possibly on a 10 year cycle). Additionally, new efforts at the NRC focus on seismically-induced fire and flood and on extending seismic hazard and risk analysis approaches to other external hazards. Recently there has also been a significant increase globally in research and guidance development related to multi-unit or facility-wide risk. Neither the NRC nor industry currently has tools capable of addressing these new elements.

4.1.2 Advanced SPRA Benefits to Industry

The Advanced SPRA tools and methods would address a wide range of short-and long-term needs, in addition to providing opportunities for improved design approaches. The benefits to industry include the following:

- **New tools and methods could give NPP owners more a reliable and comprehensive tools set to better understand and manage risk for a range of external hazards.** The Advanced SPRA project will provide more up to date and computationally appropriate tools and methods that will remove the conservatisms that, necessarily, exist in current analysis approaches. The tools and methods will also provide a more transparent and integrated approach to quantifying uncertainties. Current the tools and methods available are limited to seismic risk, although a need exists for flooding, high-wind, and other external hazards. By providing a tools set that can be broadly applied, NPP owners can more confidently determine the appropriate use of limited resources. The analyses provide more reliable information upon which to base plant modifications or operational changes.
- **New tools and methods could better support regulatory and licensing actions including risk-informed license modifications, relicensing efforts, and assessment of risk impacts that arise as part of the ROP SDP**. State-of-the-art tools increase regulatory confidence and will be developed accounting for technical elements that are of interest to the NRC.
- **New tools and methods could address multi-unit and facility-wide risk.** As a result of events at the Kashiwazaki Kariwa, Fukushima Daiichi, and Fukushima Daiini NPPs, significant effort is being put towards new guidance related to assessing multi-unit risk. Tools and detailed methods to assess multi-unit risk do not currently exist. Because the tools and methods being developed as part of the Advanced PRA project are being built from the ground up, they can be applied to both single unit and multi-risk efficiently.
- **Advanced SPRA tools could reduce conservatisms in design by supporting design approaches that target plant-level performance and risk objectives.** Design approaches currently apply some plant-level objectives at the SSC-level. This is, in part, a result of the cumbersome nature of available tools. More computationally efficient tools could lead to the reduction in conservatisms in design by allowing for SPRA to be more efficiently and effectively incorporated into the design process. This also allows for defense-in-depth to be demonstrated more transparently.
- **New Non-linear SSI tools could reduce conservatisms in design, while more realistically demonstrating beyond-design-basis performance.** Current tools required conservatism in design in order to account for limitations in the tools currently used in the nuclear industry. These tools generally do not accurately capture behavior in the non-linear range and are, therefore, less accurate for beyond-design-basis loads, which must be analyzed to demonstrate margin required by current regulatory guidance.
- **Seismically-induced fire and flood could be incorporated directly into Advanced SPRA through a new set of methods that incorporate time- and physics-based tools in RISMC.** The NRC NTTF recommended addressing seismically-induced fire and flood and the NRC is currently performing research in the area. The tools developed in the Advanced SPRA project are focused on assessing best estimate risk numbers (with associated uncertainty), while the NRC is likely to take a conservative approach. The tools and methods for incorporating “deterministic” analysis could also later be extended to incorporate or analyze other elements of risk, such as aging effects, coping time under certain plant conditions, and human factors.

- **New tools could better address plant facilities that are not traditionally part of a SPRA analysis, such as spent fuel pools and independent spent fuel storage installations.** These plant facilities are coming under increasing scrutiny as a result of the Fukushimi Daiini accident and ongoing NRC research activities.

4.2 Verified and Validated Advanced Toolkit and Methods

The methods and tools under development will be verified and validated using existing data and using the proposed External Hazards Experimental Center (EHEC). This center will be a partnership between INL, other national laboratories, and universities to perform necessary external hazard experiments.

4.2.1 Seismic Analysis Validation Activities

The external hazard experiments will be used to validate physics-based external hazards numerical methods and tools. The partnership will leverage existing capabilities and develop new experimental capabilities where capabilities do not currently exist. The methods, tools, and data, which will be integrated with the intellectual and physical capabilities, are described below.

Existing Data Gathering:

- Gather data on historical and real-time recorded seismic events at nuclear facilities.
- Compare predicted results from existing numerical tools with actual data to identify seismic margins in existing numerical tools and methods.
- Use data to continually evolve numerical tools and methods.
- Provide publically available database of seismic data gathered at NPP sites. This database can be used by researchers to benchmark numerical seismic tools they have developed against recorded events.

Experimental testing to validate numerical tools:

- Gather existing experimental results
  - Experimental 1D vertically propagating shear wave experiments. Results from tests used to validate 1D seismic site response tools.
  - Experimental 1D vertically propagating shear wave experiments coupled with SSI. Results from tests used to validate equivalent linear and nonlinear seismic SSI numerical tools.
  - Large-scale, systems-level seismic testing of coupled soil-structure system. Results from this test used to validate 3D nonlinear SSI analysis.
- Use experimental results to validate numerical tools developed under this seismic capability development program
  - Includes new experimental testing techniques for gathering material property data for cyclic soil behavior. This moves away from traditional torsional shear test to develop cyclic material property data.
- Provide researcher database that documents results of experimental tests. Research can use this to validate their numerical tools.

4.2.2 Flooding Analysis Validation Activities

Flooding is a potential nuclear power plant hazard and the knowledge of how components within the plant fail as a result of flooding is vitally important. Methods currently used to determine component failure resulting from flooding tend to be overly conservative. The methods frequently assume that if a component is in contact with water, the component fails to work. However, it is clear that this is not always the case. It is proposed that a full scale component flood reliability testing facility be used to test component reactions to different types of flooding including water rise events, spray events, and wave impact events. This testing will provide much more accurate risk information than is presently available and will help to validate the flooding models used in risk assessment.

For flooding analysis, Table 2 identifies below the initial components that might be considered for testing.

Table 2. Potential Components to Be Tested.

Potential Components for Testing	
Circuit breaker panels	Programmable logic controller cabinets
Computers	Pumps
Control room panels	Radiation detection equipment
Doors and windows	Radios
Duct work	Sensors and data recording devices
Lighting	Small Generators
Phones	Ventilation equipment
Key piping components such as valves	

In general, there are three specific types of component flooding that should be tested: wave impact testing, rising water testing, and spray testing. Combinations of these types of flooding will also be a possibility for testing. Since not all floods occur with the same type of water, this will be taken into account at this location. River water, sea water, and rain water will all be tested using the same equipment. The wave testing will require the most work because the design must be tested before being fully implemented.

Regarding the technical approach to water rise and water spray testing, the general concept is to accommodate variable rates of water rise while monitoring the component performance. The water spray testing will involve appropriate sprinkler heads found in various plant locations. The flow rates will be variable and the component performance will be monitored during the test. In each case, the ability to perform multiple tests under the same conditions will be emphasized to accommodate collection of reliability data.

The validation testing will test items to failure. For example, testing a radio would require testing to failure while recording the amount of electric current used by the device. This testing will give an idea of what failure modes we should be testing for and if failure is a yes or no test or more of a percentage type of test.

Data from a series of tests can be seen Figure 23 below. Radios were submerged in water and the current being drawn by the radios was recorded. These results show that there is some variability in current in the time leading up to failure, but all of the radios failed at essentially the same water depth, after having a spike in current just before failure.

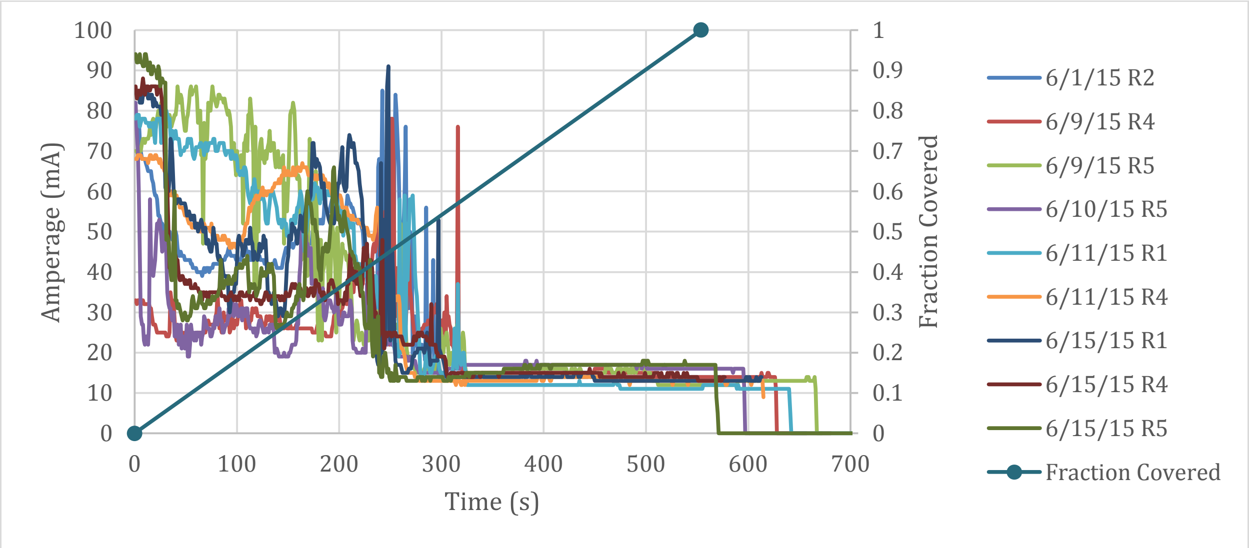

Figure 23. Small-Scale Test Results for Radios with a Slowly Increasing Water Depth.

4.3 Maturity of Toolkit and Methods

The purpose of this section is to discuss how the present industry application (IA2) improves the degree of "risk-informed" of external hazards representation.

A previous report in this series (IA1) proposed a scale of risk-informed analyses that comprised three main attributes: the scope of modeling, the fidelity of scenario modeling within the chosen scope, and the treatment of epistemic uncertainty.

The essence of risk-informed is to create a basis for resource allocation (by licensee and by regulator) that does the best job we know how to do, consistent with our state of knowledge and institutional constraints. The analysis must be geared to supporting conclusions about which scenarios are more important than others, and how much more important, and how beneficial it would be to add preventive or mitigative measures beyond what is already there.

Degrees of Risk-Informed

Following is a first-cut allocation of un-normalized weights to the lineaments of risk-informed.

Scope of Modeling

Formulation of a sufficiently broad issue space to comprise the scenarios that bear on the current decision, and the definition of figures of merit whose quantification brings everything relevant into the decision process. To get a high index in this area, the scope should include not only regulatory acceptance criteria, but also risk metrics, including impact of the subject change on operations. We will not address monetary costs, but we will try to address safety benefits and decrements as well as operational benefits and decrements (probability and severity of accidents, outages, …).

Fidelity of Scenario Set Construction to Reality (including treatment of aleatory variables)

The whole point of being "risk-informed" is to be able to say that some scenarios are more important than others, or to say that a given set of scenarios has high or low absolute importance. Therefore the modeling of the scenarios within the defined scope is of central importance.

Treatment of Uncertainties

To get the highest marks in this area, the treatment of epistemic uncertainty must be not only comprehensive, but open-minded. The decision should not be sensitive to choice of probability distributions on epistemic variables unless those distributions are largely determined by available evidence that has been brought to bear within a Bayesian framework using likelihood models that are appropriately open-minded (e.g., they do not pool trials that are not really exchangeable).

Seismic risk analysis is, of course, already risk-informed, and has always been. Partly because existing seismic risk analyses vary somewhat, it is beneficial in the present context to focus on changes in the level of risk-informed that will result from this industry application.

Table 3 below summarizes how IA2 is expected to enhance the risk-informed of external hazards representation.

Table 3. IA2 Risk-Informed Grading Scale.

"Risk-Informed" Attribute	Point Range Within Attribute	Impact of IA2
Scope of Modeling	0-2	No impact. The scope (what *classes* of scenarios get considered) is unchanged.
Fidelity of Scenario Set Construction to Reality (including treatment of	0-5	The score in this area *will* increase as a result of IA2. The modeling goes into more detail with respect to gapping and sliding. This may change both the absolute and relative importance of some

aleatory variables)		scenarios, potentially changing resource-allocation decisions.
Treatment of Uncertainties	0-4	The score in this area *may* increase as a result of IA2, depending on how the analysis is actually conducted. If the conclusions of the analysis are caveated in a careful way – if, for example, well-chosen sensitivity studies are done to characterize *bounds* on the conclusions of the analysis (rather than simply quoting a judgment-based "mean" or a judgment-based upper bound on risk), then the score in this area will go up.

Discussion

Uncertainty analysis has always been a prominent feature of seismic PRA, and by its nature, it has been scenario-based to some extent. For purposes of discussion, let us say that (referring to the above table) a good seismic PRA would score 2 (max) in scope, 3 or 4 in Fidelity, and 2 in Uncertainty. This would place it near the top of "risk-informed" study types mentioned in IA1.

Fidelity of Scenario Set Construction

"Fidelity of Scenario Set Construction" is one area where IA2 may have a significant impact in risk-informing decision-making. IA2 improves the fidelity of the scenario set to reality, by including effects (gapping and sliding) that have hitherto been neglected in the state of practice. The main reason for the previous neglect of these effects is the difficulty and complexity of addressing them. The benefit of addressing them is expected to be an improvement in the optimality of resource allocation, which (per the above excerpt from IA1 was the main point). Unfortunately, this is not the same thing as "saving money" in the short run; this is believed to be unlikely, but it is conceivable that that risk metrics could go up as a result of these modeling improvements. But even in that case, better decisions in the short run should lead to reduced expected losses in the long run.

We expect a notional increase of 1 or 2 points in the "Fidelity" area.

Treatment of Uncertainty

Depending on how epistemic uncertainty is handled, IA2 may also have some impact in "Treatment of Uncertainties." For generations, seismic PRA has been explicit about state-of-knowledge uncertainty. But the standard treatment assigns explicit probability density functions, which would arguably score 1 to 2. A top score in this area (4 points) would require analysis of reasonable, but very carefully justified, bounds on the impact of gapping and sliding.

5. WORK PLAN AND SCHEDULE

The RISMC program and the plan for the industry application were presented in the previous INL report INL-EXT-14-33186. The demonstration objectives are:

1. Provide confidence and a technical maturity in the RISMC methodology (essential for broad industry adoption)
2. Strong stakeholder interaction required
3. Address a wide range of current relevant issues (see also item (d))
4. Three phase approach:
   - Problem definition (3-6 months) – (on going)
   - Early Demonstration (eDemo) (limited scope) (6-12 months)
   - Complete Application and Validation (Long Term- Methods, Tools, Data) (1-5 years)

The program and program objectives for the Industry Application 2 (IA2) are shown in Figure 24. The eDemo is the first milestone in 2015. The approach for Phases 2 and 3 is described below.

Phase 2 considers a series of demonstrations that are realistic and relevant to the industry stakeholder. In these demonstrations, plant owner/operators actively participate by providing plant information to a given demonstration. Initially, demonstrations are a simplified version (prototype) of an integrated evaluation model. Each discipline is modeled with very simple reduced order models (ROMs). The goal is to identify all the inputs and disciplines involved and compute the approximated value of the outputs to construct a first tier "knowledge database". The "knowledge database" is then analyzed with GP emulators for the purpose of illustrating very complex problems. Later a more realistic and credible solver is used to represent the complete multi-physics demonstration.

As the program enters subsequent phases (Phase 3 and beyond) each discipline (simply represented by early demonstrations in Phase 2) is properly replaced by realistic simulators, therefore improving the fidelity and quality of the "knowledge database". Hence, Phase 3 of the Industry Applications considers the full spectrum of demonstrations with all advanced features of the RISMC toolkit (concurrently in development while early RISMC demonstrations take place), including Verification, Validation, and Data Analysis. These will include applications of RAVEN and RELAP7, also including other models in the MOOSE framework, as needed.

Figure 24 also shows projected cost and schedule for IA2 for the next five years.

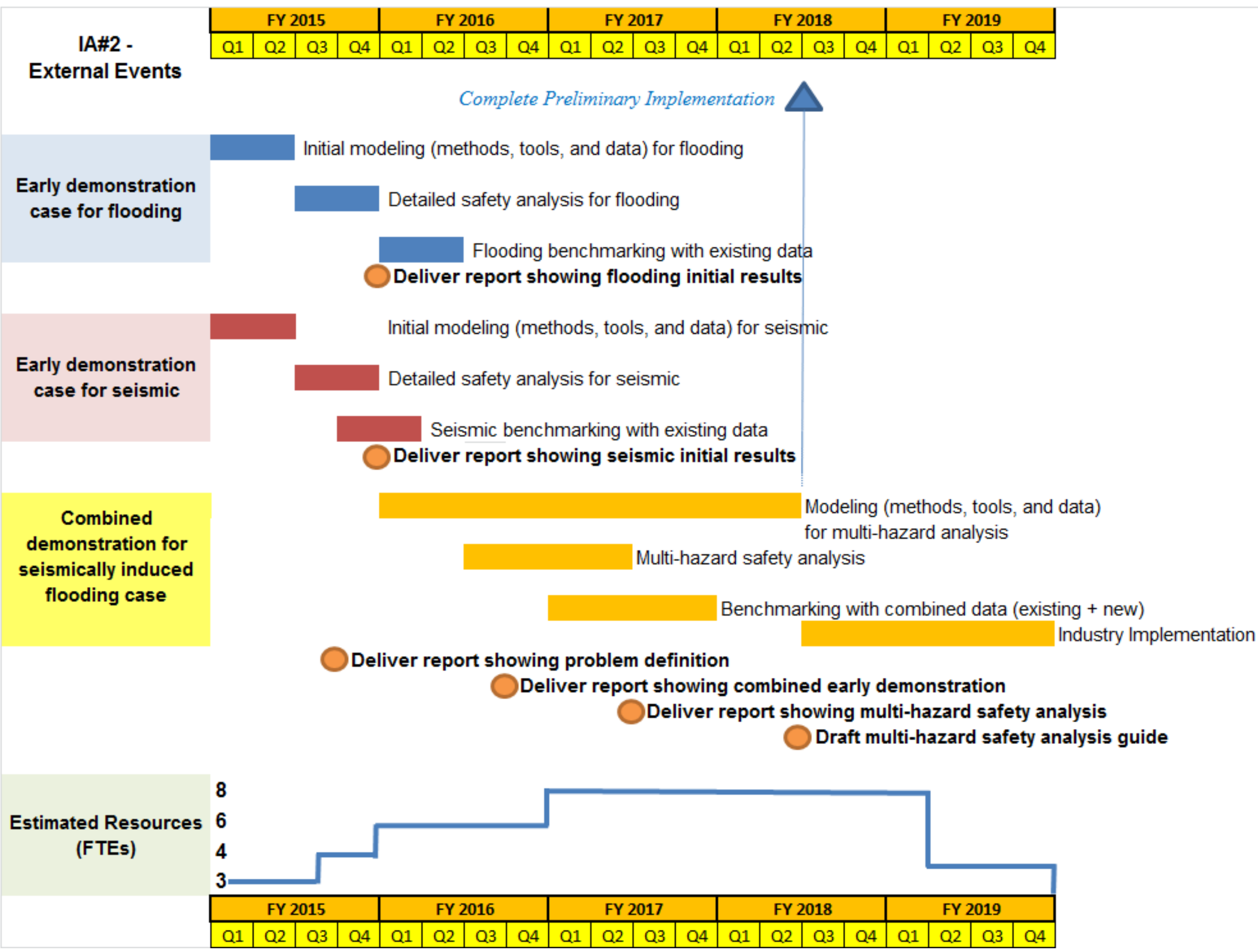

Figure 24. RIMM IA#2 Project Phases.

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APPENDIX –
Development, Needs, and Opportunities for Risk-Informed Approaches for Addressing External Natural Hazard Phenomena

Historical Development of Design Requirements in Part 50 of the Code of Federal Regulations

The earliest seismic hazard and seismic design guidance came from the Atomic Energy Commission (AEC), which was established in 1954 and was the US nuclear regulator prior to the NRC's founding as a result of the Energy Reorganization Act of 1974. Early seismic hazard assessments for the Design Basis Earthquake (DBE) were deterministic (scenario based) with standard spectral shapes tied to a peak ground acceleration (PGA) that was determined based on past earthquakes in the region. This process relied heavily on expert judgment and was based on a small set of seismic recordings from active crustal regions, such as the western US. Design approaches in the early years were conservative (e.g., using very low damping values) with structures expected to remain in the linear range at the DBE. Systems and components were designed using standard practice at the time.

In 1971 the NRC established General Design Criteria (GDC 2), which required that structures, systems and components (SSCs) important to safety be designed to withstand the effects of natural phenomena with "appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding region with sufficient margin for the limited accuracy and quantity of the historical data and the period of time in which the data have been accumulated" (see Appendix A to 10 CFR Part 50). This language codified the consideration of natural hazards (e.g., earthquakes and flooding) in design and further required a consideration of uncertainty in the hazard levels. While this language also implies a consideration only of the historic or recorded information, modern hazard assessment also incorporates types of data that were not available at the time, such as data from paleoliquefaction studies, paleoflooding studies and modern geodetics.

In 1973, Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for NPPs" was promulgated to provide more detailed criteria for evaluation of the suitability of proposed sites and the suitability of a NPPs design basis in light of seismic and geological characteristics. This was followed, in 1996, by new language that incorporates performance-based concepts and completes the current legal basis for seismic design. The new language replaced the deterministic DBE with the Safe Shutdown Earthquake (SSE) ground motion. 10 CFR Part 50 Appendix S states, "Safe-shutdown earthquake ground motion is the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional." Additionally, performance-based requirements for the SSE ground motion were provided in 10 CFR Part 50, which states that "The nuclear power plant must be designed so that, if the Safe Shutdown Earthquake Ground Motion occurs, certain structures, systems, and components will remain functional and within applicable stress, strain, and deformation limits."

The Introduction of the Certified Design Approach

The Part 52 of the Code of Federal Regulations was later enacted to support the licensing of certified NPP designs. NPPs licensed under this part of the code fundamentally meet the Part 50 requirements and objectives in terms of seismic safety. However, much of the plant design is covered under the certified design documentation and is included in a construction and operating license application by reference. The hazard assessment remains site-specific and the resulting ground motion levels are compared to the allowable certified design response spectrum (ground motions). The portions of the NPP not part of the certified plant are designed on a site-specific basis. Floor spectra from soil-structure-interaction (SSI) analyses are used for design of contents.

The Introduction of Risk-Informed Approaches and Decision Making

In 1975, the AEC commissioned WASH-1400, the first probabilistic risk assessment NPPs. The NRC later published the findings in, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear

Power Plants". As a result of the Three Mile Island accident, in 1980, the NRC published NUREG/CR-1250, "Three Mile Island; A Report to the Commissioners and to the Public." The report recommended that "More rigorous and quantitative methods of risk analysis have been developed and should be employed to assess the safety of design and operation," and "The best way to improve the existing design review process is by relying in a major way upon quantitative risk analysis." On the basis of other recommendations in that report the NRC also began to incorporate some beyond the design basis requirements. Later, the NRC encouraged licensees to use the newly developed PRA methodology to search for vulnerabilities through 50.54(f) Requests for Information. These requests are known as the Individual Plant Examination program and Individual Plant Examination for External Events (IPEEE) program. Approximately a third of the NPPs performed SPRA as part of the IPEEE program, making it the first widespread use of SPRA.

As a result of this history of risk assessment methodologies, in 1997, the NRC was issued a commission directive to move towards "Risk Informed" policies. Although the impacts on the Code were limited, and the implementation has been inconsistent, a number of NRC regulatory guides (RGs) and NRC actions and protocols have been issued in response and risk-informed decision-making is now found throughout NRC approaches and protocols. Although the policy was set to move the decision-making framework forward, the tools and methods available have changed incrementally since that time.

More recently, as a result of the findings and recommendations of the NRC's Near Term Task Force (NTTF) 2011 report, significant effort is being put into more fully implementing SPRA tools within a regulatory framework. The NRC's Office of Nuclear Regulatory Research is currently performing a Level 3 SPRA to identify areas in which research may be necessary and is also actively considering seismically-induced fire and flood implementation in safety assessment. If the NRC chooses to implement NTTF Recommendation 2.2, periodic reevaluation of NPPs for external events (e.g., fire and flood) will be required. Based on current activities, External Event PRA would be expected to be a key part of that requirement,

PSHA and SPRA in the current regulatory framework

Currently PSHA, performance-based engineering, SPRA and risk-informed decision making are currently being used in the NRC regulation of NPPs in a number of ways, including:

- PSHA is the basis of the development of the SSE ground motion defined as part of the design bases for new NPPs (RG 1.208 and NUREG-0800).
- Risk-informed performance requirements are defined as part of the design requirements and design bases for new NPPs (ASCE 43-05 through reference in RG 1.208 and NUREG-0800).
- SPRA is used as a required confirmatory assessment of new NPPs to ensure that NRC risk-objectives are met (NRC Interim Staff Guidance DC/COL-ISG-20
- PSHA is the basis for the assessment of new seismic hazard information used as part of the 2012 NRC 50.54(f) Request for Information activities. As discussed later in this document, the new information was used both as part of screening criteria and as input to SPRA for those plants that "screen in".
- SPRA is being used as a decision-making tool as part of the 2012 NRC 50.54(f) Request for Information activities, as discussed later in this document.
- External Event PRA is used as a decision-making tool in the Reactor Oversight Program and its significance determination process
- External Event PRA is used as a basis for voluntary license amendment applications (RG 1.200 and RG1.174)

SPRA and Performance-Based Methods in Design

In 2007, the NRC issued RG 1.208, "A Performance-Based Approach to Define the Site-Specific Ground Motion," which the first NRC seismic hazard assessment guidance based entirely on probabilistic methods tied to risk-informed objectives. RG 1.208 was developed for use with ASCE 43-05, “Seismic Design Criteria for SSCs in Nuclear Facilities.”

Current guidance targets both performance goals for individual SSCs and plant level risk goals. The SSC performance goals are developed to provide the engineering practitioner a set of criteria to work against such that the plant-level risk objectives are achieved. The NRC to date specified risk-objectives in detail for new or operating NPPs. However, based on past NRC studies and actions, annual Core Damage Frequencies (CDF) of approximately 10-6/yr for new plants and approximately 10-5/yr for existing plants are generally considered to be the point at which risk information requires a closer look by the NRC. Because CDF is for any kind of damage of the core, and does not imply release from the NPP, the Large Early Release Frequency (LERF), which is tied to risk to the population and environment due to core damage plus breach of containment should be at least an order of magnitude lower than CDF.

The Frequency of the Onset of Significant Inelastic Deformation (FOSID), is a criteria for individual SSCs that is set to 10-5 /year for individual SSCs. This objective is met by coupling ground motion with an annual probability of 10-4 to 10-5 with margin in design.

A plant-level HCLPF of 1.67 times the SSE ground motion must also be demonstrated for NPP design. This is a deterministic design criteria that has been shown to meet CDF risk objectives for PWRs and BWRs. Importantly, this is a plant-level criteria, but it is typically implemented at the SSC level as a margins criteria. This leads to overdesign of NPP facilities that could lead to substantial savings for new designs if more efficient and accurate tools and methods for external events PRA are developed. This is particularly true if the tools and methods provide for PRA insights to be fed back into the design process efficiently (as discussed above).

As part of 10 CFR Part 52, licensees are required to perform a PRA and provide a description and analysis of design features for the prevention and mitigation of severe accidents (10 CFR 52.47(23) and 10 CFR 52.79(48)). Because site-specific hazard information and plant walkdowns, which can’t occur until the plant is built, are required to complete a seismic PRA, the NRC published, DC/COL-ISG-20, “Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors.” This guidance describes how to develop an SPRA-based margins assessment through the licensing process. The final SPRA must be completed and accepted by the NRC prior to fuel loading.

Risk Assessment In the Reactor Oversight Program

The NRC’s Reactor Oversight Program (ROP) assesses the safety and security performance of operating commercial nuclear power plants, and responds to any decline in their performance. Within the ROP, another process, called the Significance Determination Process (SDP), is used to determine the safety or security significance of inspection findings. This process provides both an initial screening to identify those inspection findings that do not result in a significant increase in plant risk as well as those that may require a more thorough risk assessment. Depending on the final outcome of the risk assessment and the dominant metric used for increase in risk (i.e., ΔCDF or ΔLERF), a color associated with the risk thresholds (presented in Figure A-1 below) would be determined.

Because the External Event PRA tools and methods currently available to industry and the NRC are not computationally efficient or flexible, and do not easily incorporate natural hazards, the plant models and risk-assessment tools used for this process are, necessarily, simple. Flood PRA models do not exist for most plants. As a result, updated tools and methods that more efficiently and accurately assess changes in risk, could benefit all stakeholders involved in the ROP.

Risk Assessment used for Voluntary Licensing Changes

The US regulatory framework allows NPP operators to request voluntary risk-informed changes to their licensing basis. This is described in NRC RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis.” As an example, SPRA results can be used to demonstrate that specific equipment may be held to less stringent regulatory criteria without causing risk implications for the facility. In this case, the classification of the equipment is changed to match the new maintenance requirements. Figure A-2 below, from RG 1.174, show the risk criteria the NRC uses in terms of both CDF and LERF.

The NRC, in implementing this voluntary process, provides opportunity to NPP owners to find efficiencies by risk-informing their operations and maintenance requirements. This is an area in which new external event PRA tools and methods could benefit industry by identifying and supporting license basis changes that reduce ongoing operational costs.

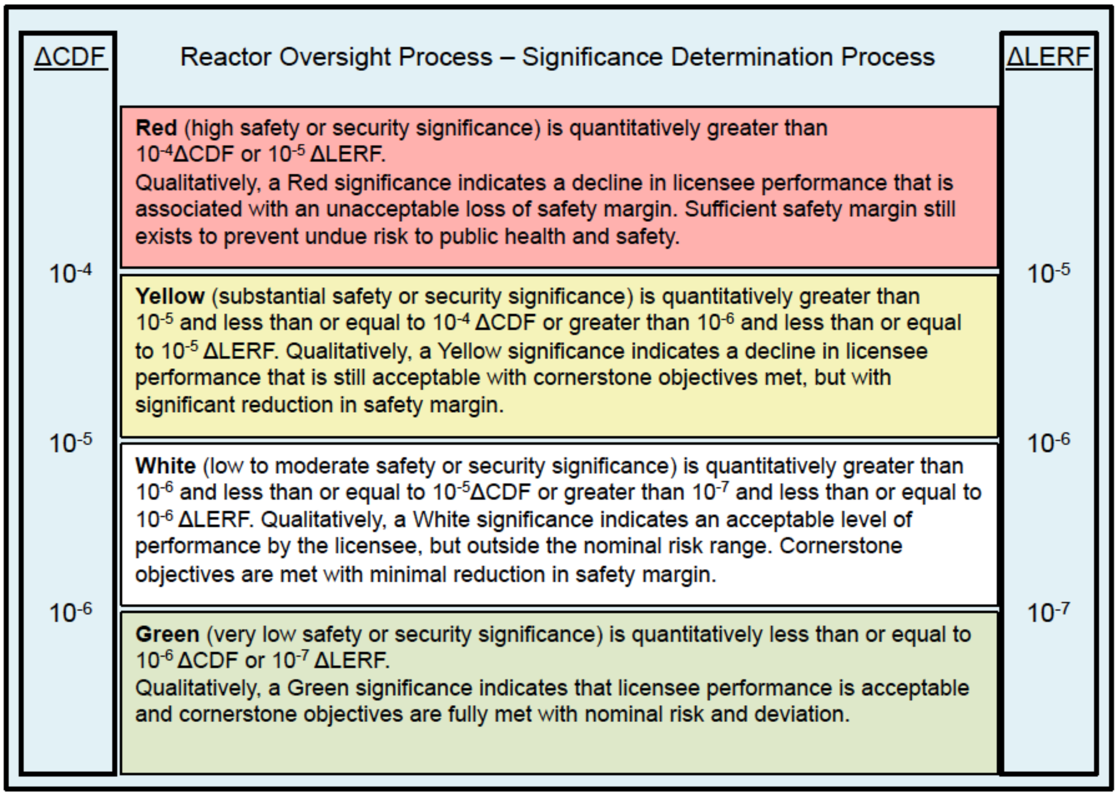

Figure A-1. Example of How Risk Implications Are Used in the NRC's Reactor Oversight Program. (NUREG-2150)

Use of risk-informed approaches in the post-Fukushima recommendation 2.1 program (i.e. response to the March 2012 50.54(f) Request for Information.

The above discussed application of external events PRA are focused on cases where it is the capacity of a piece of equipment or system that has changed. More recently, situations have arisen in which the focus is on risk implications resulting from the consideration of higher assessed hazard levels, either because a natural hazard was not previously assessed (e.g., dam failures or local intense precipitation) or because the assessed values of a hazard has increased (e.g., due to a new scientific understanding of some aspect of the phenomena).

Changes in assessed hazard levels are typically more challenging than changes in SSC capacity from a regulatory perspective because the natural hazard inherent in the region around the facility needs to be considered with respect to an existing facility for which siting and design criteria were previously defined. In cases where estimates of hazard have increased, external events PRA is a particularly valuable tool that can provide key data needed to answer important questions, with the most important being:

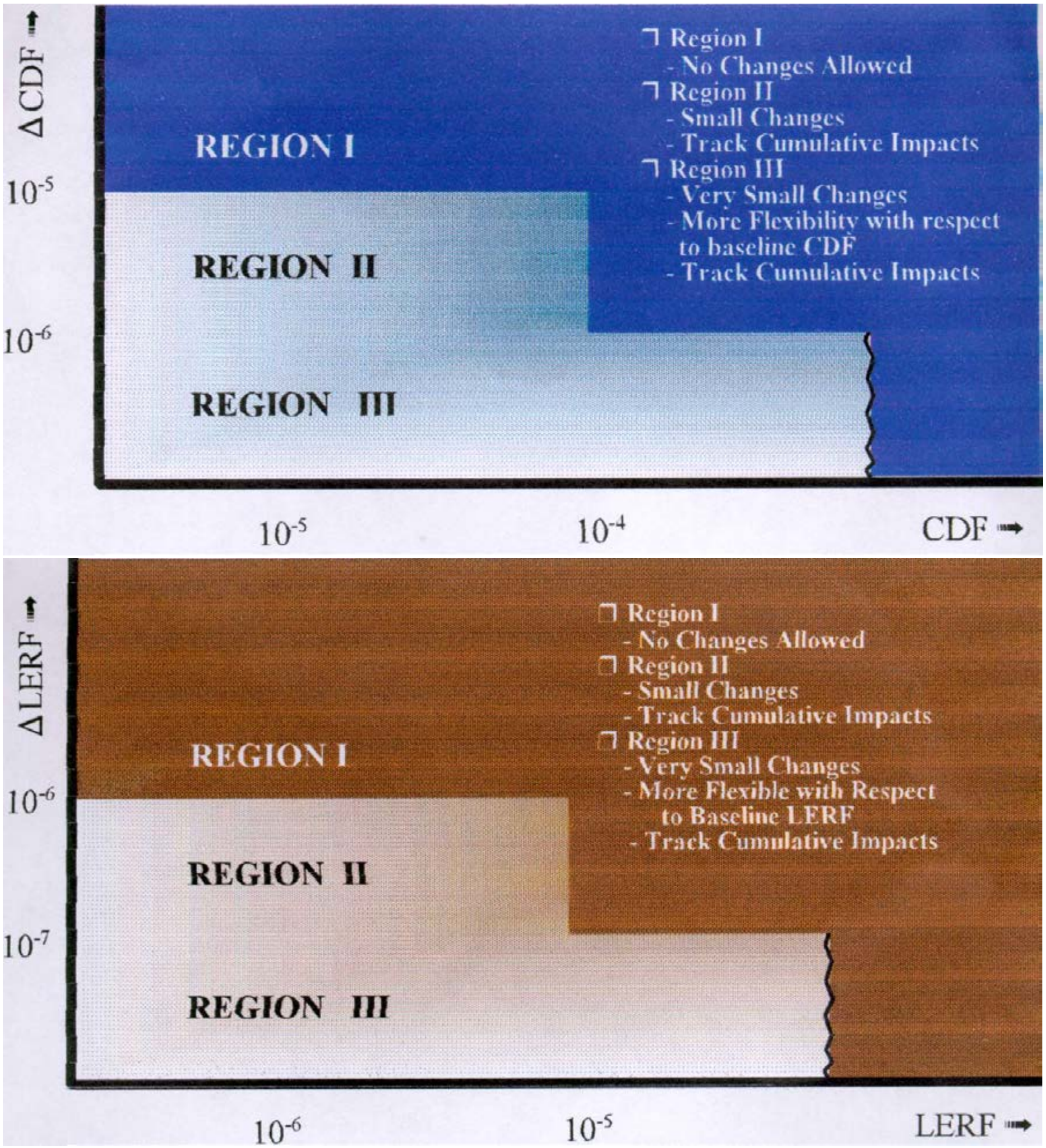

Figure A-2. Risk Criteria Used by the NRC in Terms of CDF and LERF (RG 1.174).

- Given the increase in assessed hazard values, does the risk posed by the facility still fall within acceptable risk tolerance limits?
- If the assessed risk exceeds acceptable limits, what are the key risk contributors? What are the options for effectively mitigating the risk such that the risk can be lowered to within acceptable limits?

The most notable example of how SPRA can fit into a process of hazard reevaluation is found in the recent post-Fukushima activities implemented by the NRC. Following the events at the Fukushima Dai-ichi nuclear power plant in Japan on March 11, 2011, the NRC established a senior-level agency task force referred to as the NTTF, which was tasked with conducting a systematic and methodical review of NRC regulations and processes and determining if the agency should make additional improvements in light of the accident. The NTTF developed a comprehensive set of recommendations, documented in SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," issued July 12, 2011. The NRC Staff enhanced these recommendations through interactions with stakeholders and issued SECY-11-0124, "Recommended Actions To Be Taken without Delay from the Near-Term Task Force Report," and SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned."

The NTTF recommendations 2.1, 2.2 and 2.3 related to seismic and flooding hazard and safety. Recommendation 2.3 was implemented immediately and focused on immediate seismic and flood walkdowns of the facility to confirm that the NPPs current licensing basis was being met. The Recommendation 2.3 activities have, essentially, been completed. Recommendation 2.2, which requires a longer-term rule making activity, is focused on requiring periodic (10 year) reevaluation of natural hazards at operating NPPs. Recommendation 2.1 focused on the reevaluation of seismic and flooding hazard and risk at operating NPPs. It is important to note that, at the time of the Fukushima accident, the NRC was already working on a reevaluation of the seismic safety of US NPPs as a result of the Generic Issue 199 evaluation that had been ongoing for several years. Additionally, the NRC was already actively developing new seismic source characterization and seismic ground motion models for the central and eastern US. The NRC was also in the final stages of developing NUREG 2117, which provides additional practical guidance on conducting hazard assessment studies. The NRC has also been working on Generic Issue 204, "Flooding of Nuclear Power Plant Sites Following Upstream Dam Failure."

In March 2012, the NRC issued a 50.54(f) Request for Information letter to all operating NPPs. Enclosure 1 of that letter, "Recommendation 2.1: Seismic," described the actions related to seismic hazard and risk reassessments for licensees to take in response to the letter. In response to the 50.54(f) letter, EPRI and NRC staff and contractors worked together to further refine the process initially described in Enclosure 1 and to address several specific technical areas where additional guidance would bring greater efficiency and reduce uncertainty in the conduct of the hazard and risk assessment activities. The outcome of that collaboration was EPRI Report 1025287, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," which was endorsed by the NRC. The SPID report (as it is commonly called) provides targeted information and is not intended to be general SPRA implementation guidance, although some of the technical approaches in the report are likely to be used in the future. Figure A-3 of the SPID (reproduced below) lays out Phase 1 of the reevaluation process, which was collaboratively enhanced from that originally provided in Enclosure 1 of the 50.54(f) letter.

In Phase one of the process implemented by the NRC, the seismic hazard was reevaluated for all operating reactors using the guidance for new reactors in place at the time of the evaluation. Those NPPs for which the new hazard assessment, expressed in terms of a RG1.208 Ground Motion Response Spectrum, exceeded the original design basis in the 1 to 10 Hz frequency range "screened into" further safety/risk evaluation. Those NPP for which the GMRS exceeds 1.3 times the design basis in the 1 to 10Hz range are required to perform a seismic PRA.

Plants that screen in to additional risk assessment, but do not exceed the design basis by 30% can choose to do a seismic margins assessment instead. Although this option was provided, and the NRC issued Interim Staff Guidance JLD-ISG-2012-04, "Guidance on Performing a Seismic Margin Assessment In Response to the March 2012 Request for Information Letter," all NPPs that have screened in have chosen to do an SPRA. In phase two of the process (which is being refined at the time of this writing), the NRC will use the results of the SPRA to determine if future regulatory action is needed on a plant-by-plant basis.

Additional screening criteria for high frequency exceedance are also considered in both the 50.54(f) letter and the SPID. Addressing high frequency exceedance let to a separate process, led by EPRI that involved several steps. First, a new shake table-testing program was conducted on potentially high-frequency sensitive equipment by EPRI. This limited the number of types of equipment of concern. Next, a protocol is being developed to assess the impact of potentially sensitive equipment to higher ground motion levels. This protocol is expected to address the inclusion of high frequency sensitive equipment in SPRA.

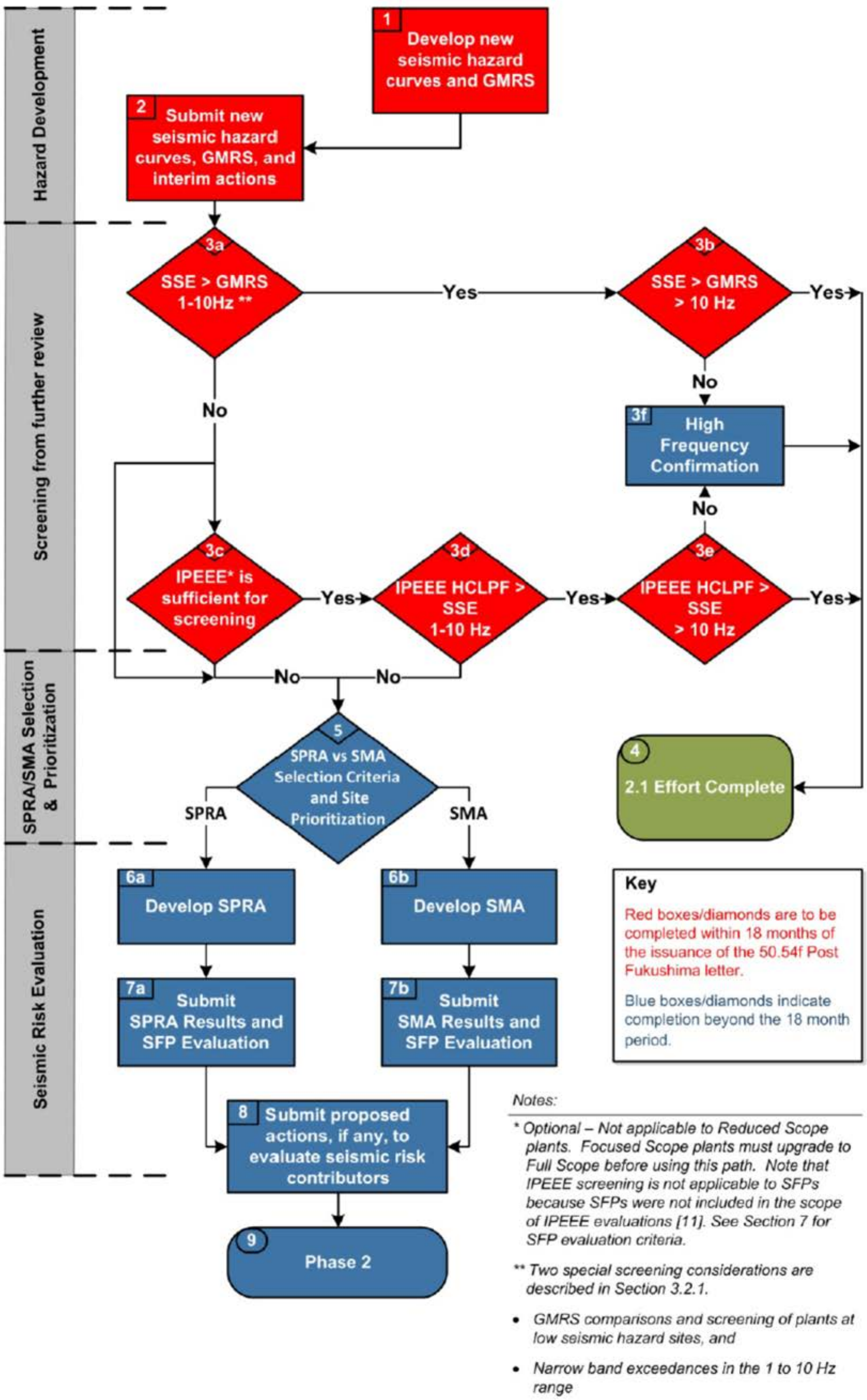

Figure A-3. EPRI SPID Flow Chart.

The NRC 50.54(f) letter discussed the option of NPPs proposing “interim actions” to be taken while any required risk assessment studies were being conducted. In response, the industry proposed an additional activity, which they termed the “augmented approach” and which is described in EPRI Report 3002000704, “Seismic Evaluation Guidance: Augmented Approach for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic.” This report describes and expedited evaluation process that addresses interim evaluations of critical plant equipment. This evaluation was conducted by plants that had screened into additional risk evaluation in a very short time frame. The activity was intended to ensure and demonstrate seismic safety as the NPPs were conducting their risk evaluation studies.

Current Needs in Seismic Hazard and Risk Assessment

Probabilistic seismic hazard assessment (PSHA) and seismic probabilistic risk assessment (SPRA) approaches have been applied and improved for several decades and are now considered to be relatively mature in terms of their conceptual development and application. Unfortunately, the tools currently available for SPRA (of which PSHA is a part) are relatively inflexible and were developed principally for internal event probabilistic risk assessments. As a result, currently available tools are now significantly limiting the development of more advanced SPRA methodologies. Development of “next generation” seismic risk assessment tools and methods, which are built upon and expand the RISMC tool kit, would lead to significant improvements in industry’s ability to address regulatory requirements and make the most of regulatory opportunities (e.g., risk-informed relief) related to seismic hazard.

Current Needs in Flood Hazard and Risk

There is also a recognized and growing need for tools and methods to assess risk from other natural phenomena, most notably flooding. Although flooding is an area of significant concern and focus for the US NRC and industry, in part as a result of the Fukushima accident, probabilistic hazard and risk assessment tools and methods for flooding are in their infancy. Although many elements of the tools and methods developed for seismic hazard and risk assessment can be applied to flooding assessments, significant challenges remain. In particular, the way in which floodwaters and seismic loads impact a NPP once they reach the site are fundamentally different. As a result, flood risk assessment methods must also incorporate robust time-domain physics-based modeling that can provide insight and information on realistic accident sequences, accident progression, and other inputs to both margins-based safety assessment tools and probabilistic risk assessment tools.

At the same time, reevaluation of flooding hazard and risk was also being conducted in response to the NRC 50.54(f) letter. Because flood hazard and risk assessment processes have not been developed over the past decades in the way that seismic tools and methods have, the flood safety reevaluation process turned out to be far more problematic than the seismic reevaluation. The flood hazard assessment process is still deterministic in nature, resulting in large uncertainties, a lower assurance of safety overall, and potentially very conservative review flood levels for some facilities. Because flood hazard assessment was not probabilistic, and tools for flood PRA have not been developed to a generally implementable degree, flood PRA techniques could not be applied as part of the 50.54(f) process. This is a significant shortcoming that new tools and methods can address.

Generally, the recent activities have brought to light a large number of technical challenges and shortcomings in the current set of tools and methods available for assessment of NPP safety in light of natural hazards. New tools and methods could create significant benefit for the nuclear industry, while better demonstrating NPP safety and increasing regulatory assurance.

Risk Information used in Decision Making

Generally, risk information used in regulatory decision making falls into 3 categories:

- Plant-level risk metrics
- Information on risk-significant SSCs
- Information on risk-significant accident sequences.

The most important plant-level metrics are CDF, LERF, ΔCDF, and ΔLERF. Both CDF and LERF can be used to determine whether or not the overall risk profile of a facility is within acceptable limits. Typically, a PRA can consist of multiple levels, connecting the impact of hazards and operational failures to the resulting radiological releases. A Level 1 PSA estimates the frequency of accidents that cause core damage, while a Level 2 PRA evaluates the progression of such core damage accidents to estimate the frequency of accidents that release radioactivity from the nuclear power plant. Based on the Level 2 radioactivity release accidents, a Level 3 PRA estimates the consequences in terms of injury to the public and damage to the environment. In the US framework, surrogates for the radiological consequences of reactor accidents are be implemented via CDF and/or LERF metrics. Additionally, CDF- and LERF-based criteria provide additional risk-insights when compared to a margins-based approach.

The metric ΔCDF is defined as the change in CDF as a result of a change in either capacity or hazard on the risk to an operating NPP. Risk significant SSCs are identified through analysis of the PRA results. Importance measures such as the risk reduction worth, risk achievement worth, Birnbaum importance factor, and the Fussell-Vesely importance measure are used to assess the importance of individual SSCs. Analysis of the risk contribution of individual SSCs has several purposes within a risk-informed framework. An analysis of the risk-significance of SSCs helps to ensure that no single SSC dominates the risk profile, which supports defense-in-depth objectives related to diversity and redundancy. Importance analysis can help identify SSCs with the greatest potential for risk reduction. This is particularly important if a PRA indicates that risk objectives are not met for an NPP or design.

Analysis of the PRA results also provides information on the most significant accident sequences. For new reactors, this information can be fed back into a review design process that considers the overall balance of the plant and whether or not risk-based objectives are fully reached.

Relationship between design, margin and risk approaches

Hazard resistant design (including margins assessments) and external event PRA methods have an important and complementary role in assuring NPP safety. Current design methods ensure adequate performance and protection of SSCs on an individual basis through the application of some margins- or performance-based criteria associated with some benchmark load caused by the hazard of interest. This load may be deterministic in nature or may be specified using probabilistic methods. Currently, in the US regulatory framework flooding design basis is defined using a deterministic approach, while seismic hazard is defined probabilistically. However, the NRC is conducting activities to develop probabilistic flood hazard analysis (PFHA). In any case, the loading level for nuclear design is intended to be an extreme or very rare event that is highly unlikely to be exceeded. Unfortunately, in recent years, the design loading level that was defined for currently operating NPPs has been exceeded in a number of cases, including at the North Anna NPP in the United States.

In traditional engineering approaches, design margin is defined as the ratio of the code capacity of an SSC (either physical or functional) over the specified design load. To calculate design margin, the load must, necessarily, be defined as a specific value (as opposed to the full hazard curve). The design margin does not define the actual additional capacity of the SSC, which may be well beyond the design margin. Some modern techniques also incorporate a characterization of uncertainty in SSC performance in order to quantify the confidence that the SSC will perform its intended function at the specified design level. Although SSCs are traditionally designed to have significant margin, challenges occur when the design basis loading level is shown to be more likely than expected. In these cases, external events PRA can be used to demonstrate that risk objectives are still met, even if the calculated margin is reduced. Additionally, if risk objectives aren’t met, PRA provides the data and insights needed to support decisions related to plant or operational modifications.

Figure A-4 below illustrates graphically how design (or margin) and risk assessment methods differ in the way they consider hazard inputs. It shows the relationship between some parameter (hazard) level and the annual probability of exceedance associated with that particular parameter level. As the parameter levels increase they become more and more rare. There is also some uncertainty in the assessment of hazard, which must be accounted for. The determination of a design load, necessarily, leads to an individual value that can be used for engineering purposes, as denoted by a dot in the figure. If a probabilistic approach is applied, the value associated with the mean (as opposed to median) hazard is typically used. However, even when a design load is developed using deterministic methods, it has some probability of occurrence implicitly associated with it.

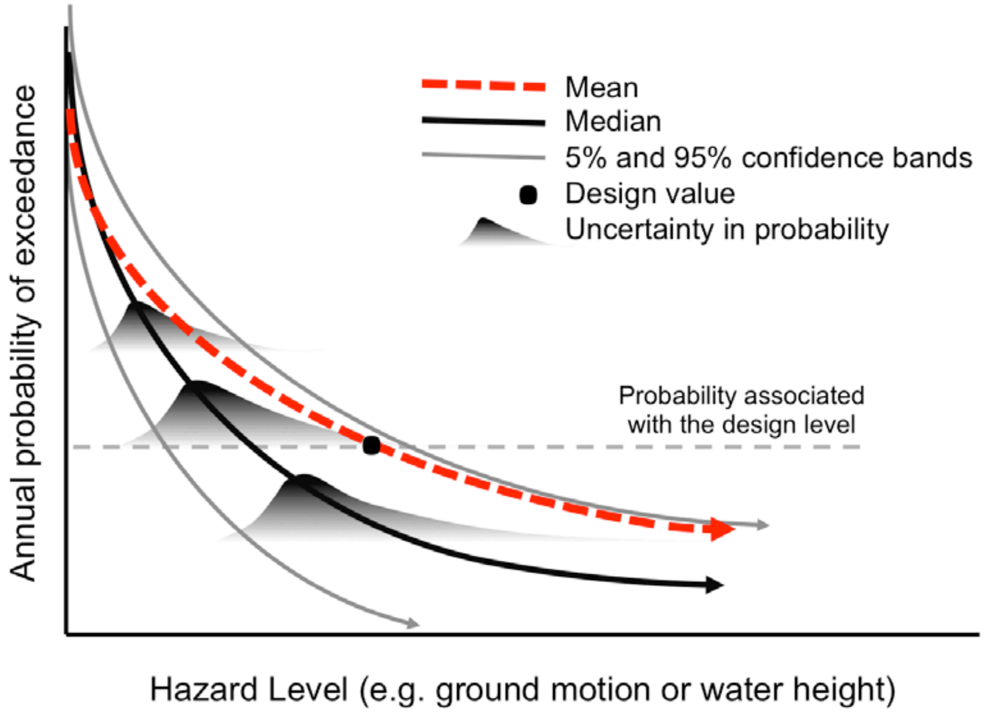

Figure A-4. Illustration of Relationship between Design Level and Total Hazard Curve.

Even when the design basis loading meets objectives in terms of its rarity, there is always some probability that the chosen design level (and even the additional margin added to the load for conservatism) can be exceeded. Risk assessment methods incorporate the full range of hazard levels, along with their associated probabilities and uncertainties, in order to determine overall risk at a plant and gain risk insights. Additionally, risk is a function of both individual SSC capacities and the interaction of SSCs within the plant systems and operations. Elements such as human actions and random (non-hazard-related) failures must also be accounted for when assessing natural hazard risk to nuclear plants. External events PRA, therefore, further complements the traditional design approach and margins assessment methods by assessing beyond design basis load conditions with a consideration of accident sequences and the probability of failure of individual SSCs.

Incorporating risk-informed decision-making into regulatory processes, could lead to greater efficiency and focus on nuclear safety by bringing the consideration of both design and beyond design basis events into a quantitative framework such that decisions to ensure and demonstrate that acceptable risk levels can be made using risk-insights. As discussed above, this is particularly true if new seismic hazard information comes to light. However, because plant-level objectives are often now applied at an SSC level, there is also room for far greater efficiencies in design. Design, margin assessment and risk assessment methods could be more effectively used for new NPPs if incorporated into an iterative process that feeds risk information and risk insights back into the design activities. These efficiencies could be realized even when improving the actual safety and risk profile of the NPP. This is an area in which the next generation of risk assessment tools could have significant untapped benefit.