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Revision 0

Regulatory Gaps and Challenges for Licensing Advanced Reactors Using Seismic Isolation

Annie Kammerer
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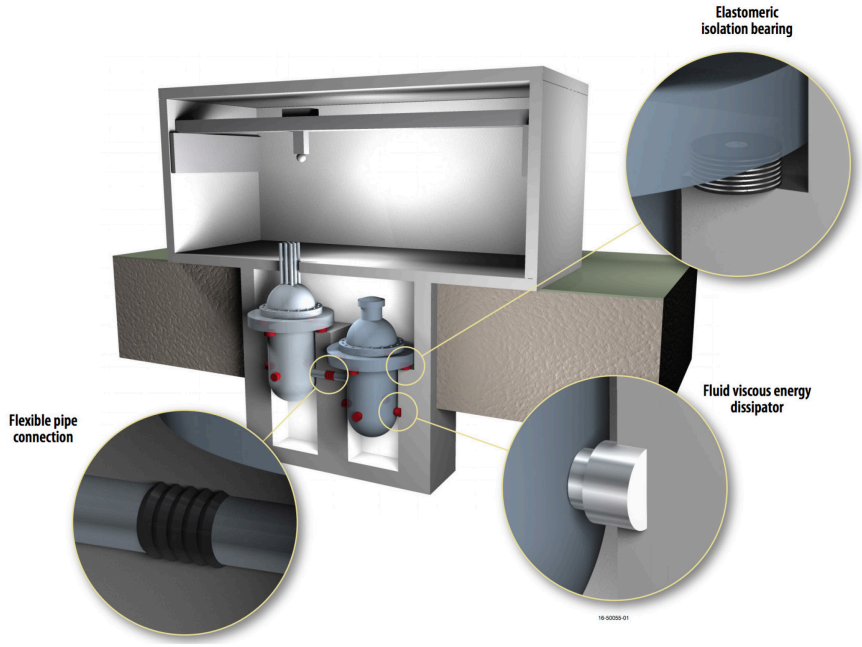
March 2016



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Regulatory Gaps and Challenges for Licensing Advanced Reactors Using Seismic Isolation

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EXECUTIVE SUMMARY

Over the last decade, particularly since implementation of the certified design regulatory approaches outlined in 10 CFR 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," interest has been increasing in the use of seismic isolation (SI) technology to support seismic safety in nuclear facilities. In 2009, the United States (U.S.) Nuclear Regulatory Commission (NRC) initiated research activities to develop new guidance targeted at isolated facilities because SI is being considered for nuclear power plants in the U.S. One product of that research, which was developed around a risk-informed regulatory approach, was a draft NRC nuclear regulatory commission (NUREG) report that investigated and discussed considerations for use of SI in otherwise traditionally-founded large light water reactors (LWRs). A coordinated effort led to new provisions for SI of LWRs in the forthcoming standard ASCE/SEI 4-16, "Seismic Analysis of Safety-Related Nuclear Structures." The risk-informed design philosophy that underpinned development of the technical basis for both of these documents led to a set of proposed performance objectives and acceptance criteria that was developed to serve as the foundation for future NRC guidance on the use of SI and related technology.

Although the guidance provided in the draft SI NUREG report and ASCE/SEI 4-16 provides a sound basis for further development of nuclear power plant designs incorporating SI, these initial documents were focused on surface-founded or near-surface-founded LWRs and were, necessarily, limited in scope. For example, there is limited information in either the draft NUREG report or ASCE/SEI 4-16 related to nonlinear analysis of soil-structure systems for deeply-embedded reactors, the isolation of components, and the use of vertical isolation systems. Also missing from the draft SI NUREG report are special considerations for licensing of isolated facilities using the certified design approach in 10 CFR 52 and a detailed discussion of seismic probabilistic risk assessments (SPRAs) for isolated facilities.

To identify and address limitations in the initial guidance, Idaho National Laboratory (INL) has initiated several projects focused on further developing the technical and licensing underpinnings for facilities using SI technology. These efforts include a 2014 workshop focused on SI, development of new structural analysis tools and methodologies, and development of this report to identify and describe regulatory gaps and challenges related to licensing of advanced reactors using SI. Nearly all of the gaps and challenges identified in the report fall outside the scope of current efforts (including those at INL). This report provides information for developing a roadmap for future activities related to SI of advanced reactors. Although design optimization and commercial aspects related to the use of SI have been identified in Coleman and Sabharwall (2014) and elsewhere as possible issues or areas of opportunity, only topics that may impact efficient and successful licensing are addressed in this report.

Because efforts to date related to regulatory guidance development (e.g., the draft SI NUREG report) have principally considered designs similar to those currently being licensed, the existing literature (some of which is discussed in this report) is reflective of traditional LWR designs. All of the regulatory gaps and challenges that apply to LWRs also apply to advanced reactors; and often the LWR case provides a simplified example as compared to the range of cases found in advanced reactors. Advanced reactor designs often lead to additional gaps and challenges not faced in LWR design. Although this report discusses advanced reactors broadly, the exact set of challenges and potential solutions for any particular reactor design is technology-specific.

SI offers potential significant economic benefits for advanced reactors because the isolation system can be used to reduce the site-dependent seismic demands below pre-qualified levels in the certified design. Site-independent SSCs such as reactor vessels and steam generators would be analyzed, designed and qualified just once. Their protective isolation systems would be site-specific.

Advanced reactors will often be designed and constructed very differently from LWRs, regardless of whether or not they employ SI and damping devices. The key technical advances in civil and structural engineering needed to deploy advanced reactors are: (1) development of performance-based seismic design and assessment procedures for non-LWR reactors; (2) development and deployment of analysis methodologies suitable for computing the response of deeply-embedded power reactors, including nonlinear time domain, soil-structure-interaction analysis; and (3) development, prototyping, and deployment of two-dimensional and three-dimensional isolation systems suitable for components ranging in size and complexity from diesel generators to reactor vessels. Advances in Items 1 and 2 are needed regardless of whether seismic protective measures (i.e., such as those noted in Item 3) are deployed. Not covered in this report are other important technical advances in civil and structural engineering needed for economical deployment of advanced reactors such as development and deployment of modular construction strategies used to minimize "one-off" field work, schedule delays, and construction cost.

The identified gaps and challenges are addressed throughout this report and summarized in Section 7. High-impact/high-value topics that should be addressed in the short term, and that must be satisfactorily completed before substantial progress on other tasks can be made, include the following topics listed below. The number in parentheses refers to the subsection number in this report where the topic is discussed. These topics are also highlighted in Table 4.

- Verification and validation of models and tools for nonlinear soil structure interaction of embedded facilities (2.1.3)
- Guidance for ground motion selection and modification for design and SPRA (2.1.5)
- Procedures for SPRA of nonlinear soil-isolator-structure systems (2.3.1, 2.3.2)
- Component isolation systems (2.3.3)
- Certified design process for deeply-embedded advanced reactors (3.1, 3.2)

ACKNOWLEDGEMENTS

The Advanced Reactor Technologies program at INL commissioned this report. Input was provided by INL's seismic research group, which is conducing research on nuclear applications using non-linear (time domain) soil-structure-interaction analysis.

This work benefitted from input by a broad research review panel on an earlier project on light water reactors. The earlier project that led to the draft NRC SI NUREG report and efforts to develop non-linear time domain soil-structure-interaction tools for use in analysis of structures using SI technologies. The earlier research project was led by Dr. Robert Budnitz (Lawrence Berkeley National Laboratory). The research review group is composed of Dr. Robert Kennedy (RPK Structural Mechanics Consulting), Dr. Nilesh Chokshi (NRC), Dr. Donald Moore (Southern Nuclear Operating Company), Dr. James Johnson (James J. Johnson and Associates), Mr. Antonio Godoy (James J. Johnson and Associates and International Atomic Energy Agency [retired]), and Prof. Jonathan Stewart (University of California, Los Angeles). Dr. Robert Kennedy (RPK Structural Mechanics Consulting) chairs the review group.

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ACRONYMS

1D	one-dimensional
2D	two-dimensional
3D	three-dimensional
6DOF	6-degree-of-Freedom
BDB	beyond design basis
CGD	commercial grade dedication
CFR	Code of Federal Regulations
CMS	conditional mean spectra
CRADA	cooperative research and development agreement
CS	conditional spectra
CSDRS	certified seismic design response spectra
DBE	design basis earthquake (ground motion)
DOE	Department of Energy
EPRI	Electric Power Research Institute
FIRS	foundation input response spectra
FP	friction pendulum
g	acceleration due to gravity (9.81 m/s^2)
GMRS	ground motion response spectra
GMSM	ground motion selection and modification
HTGR	high-temperature gas reactor
INL	Idaho National Laboratory
ISG	Interim Staff Guidance
ISRS	in-structure response spectra
ITAAC	inspection, testing, analysis, and acceptance criteria
LBNL	Lawrence Berkeley National Laboratory
LWR	light water reactor
MAFE	mean annual frequency of exceedance
MOU	memorandum of understanding
NLSSI	nonlinear soil structure interaction
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Commission (report)
PGA	peak ground acceleration

PRA	probabilistic risk assessment
PSHA	probabilistic seismic hazard analysis
RG	Regulatory Guide
SI	seismic isolation
SI-CSDRS	seismically-isolated certified seismic design response spectrum
SI-FIRS	seismically-isolated foundation input response spectrum
SIDRS	seismic isolation design response spectra
SMR	small modular reactor
SPRA	seismic probabilistic risk assessment
SRA	site response analysis
SRP	Standard Review Plan (NUREG-0800)
SSC	structure, system, and component
SSE	Safe Shutdown Earthquake (ground motion)
SSI	soil-structure-interaction
UHRS	uniform hazard response spectrum
U.S.	United States
V/H	vertical to horizontal (ratio)

1. INTRODUCTION

1.1 Advanced Reactors

Advanced nuclear reactors will be fundamentally different from existing large light water reactors (LWRs) in many key safety-related aspects: reactor technology and operation; passive safety systems; and design and construction of structures, systems, and components (SSCs). Importantly, safety-related SSCs in the advanced reactor concepts being supported at this time by Department of Energy (DOE) research and development are principally being housed in reinforced concrete structures that are deeply embedded, which poses a series of challenges in terms of seismic analysis, design, construction, and risk assessment. The Idaho National Laboratory (INL) Advanced Reactor Technologies Program commissioned this report.

Consider Figure 1, which is a section cut through a sample high-temperature gas reactor (HTGR). The embedment depth of the HTGR is a very substantial fraction of the overall height of the building. If measured in terms of seismic reactive mass, the embedded portion of the facility is likely greater than 90%. Deep embedment poses challenges for seismic analysis, design, and construction, and will require development of new seismic analysis and probabilistic risk assessment (PRA) procedures, all as discussed in this report.

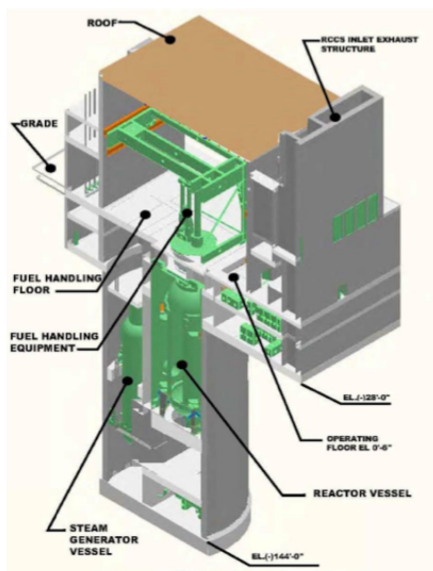

Figure 1. Cutaway view of deeply-embedded high-temperature gas reactor (INL 2011).

Seismic isolation (SI) is a viable, practical technology for protection of safety-related SSCs in nuclear facilities. SI is expected to be deployed as a result of its benefits related to increased earthquake safety and/or reduced construction cost. Methods of analysis and design for LWRs have been proposed and implemented in soon-to-be-published standards and guidance by the American Society of Civil Engineers (ASCE) and United States (U.S.) Nuclear Regulatory Commission (NRC), respectively. The clear focus of the standards and guidance is the horizontal isolation of near-surface-founded LWRs, due in part to the interest shown by industry in the mid-to-late 2000s to deploy SI in nuclear power plants (NPPs) in the U.S. The seismic input was assumed to be at one level across the footprint of the reactor building (i.e., at the foundation level of the isolated LWR shown in Figure 2). Some of the tools, techniques, and hardware developed for SI of the LWR shown in Figure 2 will be applicable in principal to deeply-embedded nuclear facilities. However, new methods of seismic analysis, design, construction, and risk assessment will be required for advanced reactors. These new methods bring with them regulatory gaps and challenges that are identified in the remainder of this report. Importantly, many of the gaps identified in this report will apply to deeply-embedded reactors, regardless of whether or not SI is employed.

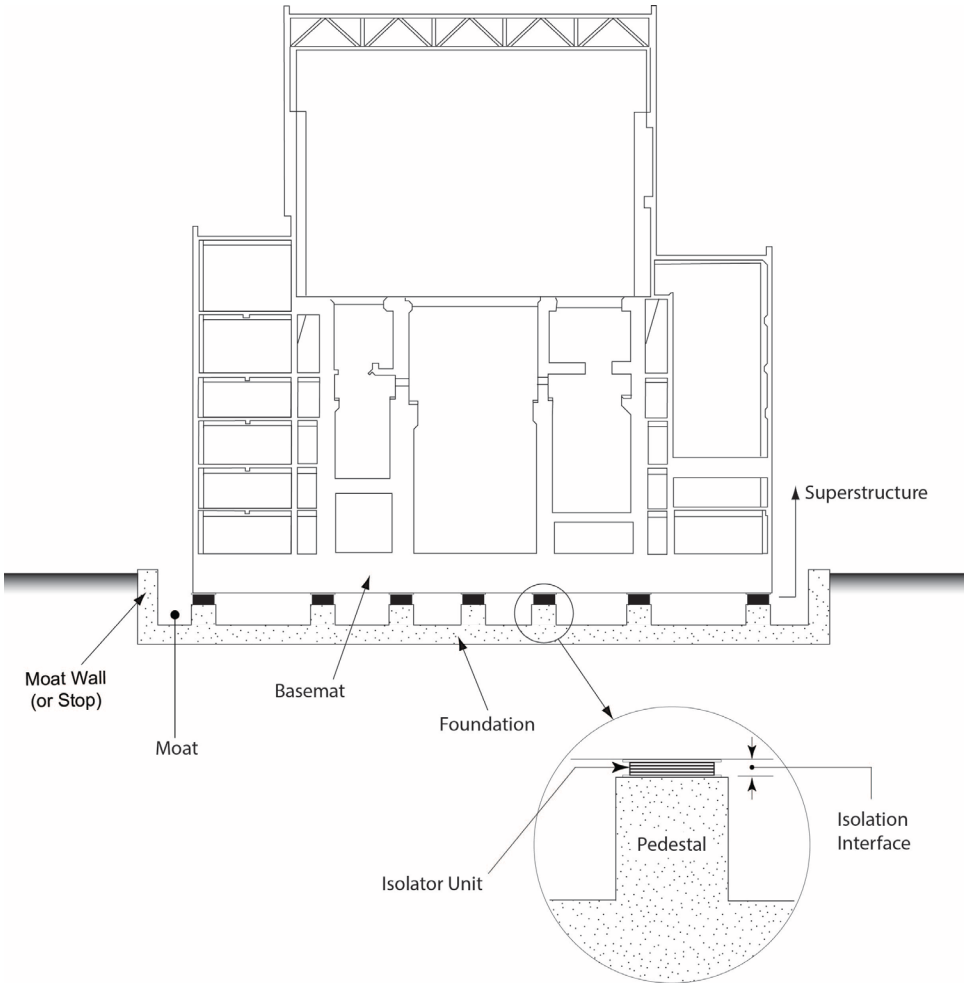

Figure 2. Seismic isolation of light water reactor (Kammerer et al. forthcoming).

1.2 Recent Advances in Application of Seismic Isolation to Nuclear Power Plants

SI has been shown to protect buildings and bridges from the effects of intense earthquake ground motions. Although various techniques for isolating a structure from the effects of earthquake shaking have been known for centuries, the global use of modern seismic (or base) isolation devices has greatly expanded in recent decades. Seismically-isolated structures now number in the thousands around the world and the technology has been shown to meet the performance needs of important non-nuclear structures. SI has been used in the design and construction of NPPs in France and South Africa and was recently used to isolate new emergency response centers at NPPs in Japan.

As a result, over the last decade, interest has been increasing in using SI technology to support seismic safety in nuclear facilities. Because SI is being considered for NPPs in the U.S., the NRC initiated research activities in 2009 to develop new guidance targeted at isolated facilities. That work, which was focused around a risk-informed regulatory approach, resulted in a draft NRC SI nuclear regulatory (NUREG) report (Kammerer et al. forthcoming) that identified and discussed considerations for use of SI in surface-founded and near-surface-founded LWRs. The risk-informed design philosophy, which is described in the draft SI NUREG report, led to a set of recommended performance objectives and acceptance criteria that could serve as the foundation for future NRC guidance on the use of SI and related technology.

The draft SI NUREG report also provides introductory background information on seismic isolators and isolation systems, a discussion of the history and performance of seismically-isolated structures during earthquakes, and a summary of SI provisions in relevant existing codes and standards. The draft SI NUREG report also discusses a series of proposed recommendations that have resulted from the NRC's research program on SI.

The SI guidance incorporated into upcoming revisions of ASCE/SEI 4 (ASCE forthcoming) and ASCE/SEI 43[1] is also risk-informed, drawing from performance statements in ASCE/SEI 43-05 (ASCE 2005) regarding unacceptable performance in design basis shaking and 150% of design basis shaking. As with the draft SI NUREG report, the isolation provisions of ASCE/SEI 4-16 and the forthcoming edition of ASCE/SEI 43 are focused on Seismic Design Category 5 structures, particularly NPPs. Because the draft SI NUREG report and ASCE/SEI 4-16 were developed concurrently through coordinated efforts, the outcomes of research focused on the technical bases for incorporating SI into the NPP design was incorporated into both documents. The gaps and challenges related to advanced reactor designs are generally the same for both documents. At the same time, research on the application of SI to nuclear facilities was being supported or conducted by the Electric Power Research Institute (EPRI), the University at Buffalo, the University of California, Lawrence Berkeley National Laboratory, Idaho National Laboratory, and international organizations.

1.3 Summary of the Draft Seismic Isolation NUREG Report

Two of the three categories of topics addressed in this report specifically involve SI in NPP facilities[2]. The most appropriate starting point for discussions on SI is the draft SI NUREG report that has been developed by the NRC. Because it is not yet published, a summary of the subject matter in the draft SI NUREG report is provided in Table 1. Some of the information provided in the draft SI NUREG report is directly applicable to advanced reactors, as noted in the table.

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[1] ASCE/SEI standards are designated by the standard number (e.g., 4 and 43) and the year of publication (e.g., the 16 and 05 in ASCE/SEI 4-16 and ASCE/SEI 43-05, respectively). Because ASCE/SEI 4-16 is in the final stages of publication, it has been listed in the references with this identifier. By contrast, the upcoming revision of ASCE 43 is earlier in the revision process; and so a year of publication is not yet known.

[2] The third category addresses gaps and challenges associated with deep foundations.

Table 1. Summary of topics in the draft seismic isolation NUREG report and their applicability to isolated advanced reactors.

Section	Topic	Content
1	Introduction to SI	Provides background on SI, including potential benefits. The information is generally applicable to advanced reactors.
2	Brief history of SI	Provides information on history of SI as applied to nuclear and non-nuclear facilities. The information is generally applicable to advanced reactors.
3	Basics of SI	Provides information on (3.1) seismic design practice, (3.2) types of seismic isolators used for base isolation, (3.3) construction of seismic isolators for base isolation, (3.4) base isolation of nuclear facilities, and (3.5) floor and equipment isolation. Also provides an approach (3.3) for demonstrating acceptability of new isolator units. The information is focused on LWRs, but is applicable, in part to advanced reactors. Section 3.3, may be of particular interest for advanced reactors.
4	Mechanics of SI	Discusses mechanical properties of the three types of isolators that are used in U.S. practice and known to be appropriate for isolation of NPPs. The information applies to advanced reactors using these isolator types.
5	Guidance on analysis and design of SI in U.S. codes and standards	Describes information on the use of SI available in U.S. codes and standards used for civil engineering design.
6	International guidance for SI applied to NPPs	Describes available international guidance on the use of SI. The discussion of content of international guidance is specific to the countries of origin.
7	Modeling and analysis of NPPs using SI	Addresses (7.1) analysis of isolated structure and (7.2) modeling of isolator units. Section 7.1 generally applies to advanced reactors with configurations similar to LWRs in terms of their structural design and properties. Configurations with different characteristics (e.g., deep foundations) will require a different modeling and analysis approach. Section 7.2 applies to advanced reactors using available SI technology in a conventional configuration. Some advanced reactor designs may benefit from new isolator designs or new approaches to SI applications.
8	Proposed performance criteria	Discusses proposed performance and acceptance criteria for NPP designs using SI. Discusses a risk-informed approach that addresses the isolator units, isolation system, superstructure, moat, stop, and umbilical lines. Generally, applies to advanced reactors with structural engineering approaches similar to LWRs and for which risk objectives related to core damage frequency apply. Structural systems fundamentally different from LWRs may require additional or different considerations, depending on the technology-specific design. Advanced reactors for which risk objectives tied to core damage are inappropriate will require a new approach.
9	Additional considerations for design, construction, and operation.	Discusses wide range of design, construction, and operations topics, including (9.1) additional design considerations (long-term changes in mechanical properties, basemat and foundation design, other external events, accident conditions and emergency response, moat cap design, and peer review); (9.2) additional manufacturing and construction considerations (quality control and quality assurance, testing of prototype and production isolators, and

Section	Topic	Content
		construction assurance); and (9.3) operational considerations (in-service inspection and replacement, additional seismic monitoring equipment, monitoring of foundation deformations, requirements for safety related equipment, and operating temperature).

Because the mandatory language contained in ASCE/SEI 4-16 is similar, though not identical, to the guidance in the draft SI NUREG report due to its common origins, the relationship between the two documents is not always clear. The differences arise from the focus in ASCE/SEI 4-16 on DOE objectives that were originally described in DOE-STD-1020-2002 (DOE 2002) and further developed for design of NPP SSCs in ASCE/SEI 43-05, which is referenced by ASCE/SEI 4-16. By contrast, the draft SI NUREG report is intended to provide a risk-informed approach that meets NRC risk objectives. The principal difference lies in the definition of the beyond-design-basis (BDB) ground motions used for design and evaluation activities. The SI provisions of ASCE/SEI 4-16 use the BDB definition in Section 1.3 of ASCE/SEI 43-05, namely, 150% of the design basis earthquake (DBE) ground motion, as documented in Huang et al. (2009b, 2013). The draft SI NUREG report defines the BDB ground motion using an acceleration response spectrum that envelopes (1) a uniform hazard response spectrum with ordinates having a mean annual frequency of exceedance (MAFE) of 1×10^{-5}, and (2) a spectrum with ordinates equal to 167% of the ground motion response spectra (GMRS).[3] This definition of the BDB ground motion is used to achieve a NPP annual core damage frequency of less than 1×10^{-6} for the accident sequence associated with failure of the isolation system. Both the draft SI NUREG report and ASCE 4-16 attach a 10% or less probability of unacceptable performance of the SI system when subjected to the BDB ground motions. This report is principally focused on the gaps and challenges related to applying guidance in the draft SI NUREG report (and associated NRC guidance) to advanced reactors in support of NRC licensing, which is the focus of this report

1.4 Background of INL Activities Related to this Report

The guidance provided in the draft SI NUREG report and ASCE/SEI 4-16 provides a solid technical basis for further development of NPP designs incorporating SI. However, these initial documents were focused on surface-founded or near-surface-founded large LWRs and were, necessarily, limited in scope. For example, there is little information in either document related to vertical isolation systems, the isolation of individual systems or components, and the isolation of deeply-embedded reactors and other types of nuclear facilities (e.g., SMRs, HTGRs, or fast reactors).[4] Also missing from the draft SI NUREG report are special considerations for licensing of isolated facilities using the certified design approach in 10 CFR 52 and special considerations for performing seismic PRAs (SPRAs) for isolated facilities.

To identify and address the limitations in the existing guidance, INL has initiated several projects focused on further developing the technical and licensing underpinnings for facilities using SI technology. In August 2014, INL sponsored a working meeting that was attended by representatives from DOE, national laboratories, industry, EPRI, NRC, and academia. As described in the resulting INL report INL/EXT-14-33234 (Coleman and Sabharwall 2014), topics related to SI included (1) general background on the state of practice in the U.S.; (2) limitations associated with implementing procedures outlined in ASCE/SEI 4-16 for SI solutions for the entire NPP; (3) potential isolation solutions for

[3] See RG 1.208 (NRC 2007b) and ISG DC/COL-ISG-020 (NRC 2010a) for more information on use of GMRS in licensing of new NPPs.

[4] Although these elements or categories are not discussed in detail, the SI NUREG report clearly states that there is no technical reason that the principles and proposed recommendations contained in that document cannot be extended to these and other cases. It notes, however, that additional considerations, constraints, and recommendations may be required.

systems and components; and (4) gaps associated with developing standardized technologies, methods, and numerical tools for these solutions. Additionally, the working meeting highlighted those SSCs that could benefit from SI technology. Issues related to implementation of SI were discussed and this helped to identify areas and needs for research and development.

Coupled with this effort, INL and the Multidisciplinary Center for Earthquake Engineering Research at the University at Buffalo are currently working with TerraPower, LLC, through a cooperative research and development agreement (CRADA)[5] to develop a methodology for evaluating the potential benefits of SI and to support advancements in the associated numerical modeling tools and approaches. A focus of the CRADA is to couple a nonlinear soil-structure-interaction (NLSSI) methodology with advanced numerical models of isolator units. The CRADA research builds upon the research developments supported by the NRC (for example, the development of advanced numerical models for elastomeric and sliding base isolators (Kumar et al. 2014, 2015a, 2015b, 2015c) that were verified and validated using American Society of Mechanical Engineers protocols). The advancement of the methodology under the CRADA is currently being developed at INL with funding provided by DOE and the National Nuclear Security Administration.

INL also continues to dialogue with the NRC's Office of Nuclear Regulatory Research. At a recent meeting between INL and NRC technical staff, SI was identified as an important area of mutual interest, particularly given that much of INL's current research builds upon earlier work by the NRC.

1.5 Objectives and Organization of this Report

Nearly all of the regulatory gaps and challenges related to deployment of SI at an NPP identified in this report are outside the scope of current technology development efforts. These gaps and challenges related to seismic design of isolated and non-isolated advanced reactors and their likely internal components (such as those identified in Figure 1). To achieve its goal, this report addresses three general categories of topics: (1) regulatory gaps and design challenges associated with SI of near-surface-founded reactors, (2) regulatory gaps and design challenges associated with deeply-embedded reactors, and (3) regulatory gaps and design challenges associated with isolation of SSCs contained within a deeply-embedded advanced reactor. Research and development activities will be required to close or overcome all three sets of gaps and challenges; and the products of the work conducted should be moved quickly into regulatory space prior to commercial deployment. All of the gaps and challenges identified can ultimately be addressed through new guidance (in some cases after work to develop a clear and supportable technical basis is conducted), as summarized in Section 0.

Although standard design and commercial aspects of SI have been identified (in INL report INL/EXT-14-33234 and elsewhere) as possible issues or areas of opportunity, only topics that impact NPP licensing are addressed in this report. This report describes regulatory gaps and challenges that have been identified at this point in time; additional issues may be identified as design and licensing activities of reactors using SI progresses. In this report, a *gap* is defined as a topic that is not addressed in current regulatory guidance. By contrast, a *challenge* is defined as a topic for which the existing guidance creates ambiguities or conflicts. In some cases, the challenges arise because the existing guidance assumes the use of a specific tool (e.g., SASSI) or methodology or because the guidance is focused on a specific NPP technology (e.g., LWRs). Some challenges relate to terminology and arise as a result of the more complex foundation system associated with a base-isolated superstructure.

[5] The point-of-contact for the CRADA is Justin Coleman, an author of this report.

In the U.S. regulatory framework, the Code of Federal Regulations (CFR) generally provides high-level requirements targeted on performance objectives that are to be demonstrated as adequate and reliable with respect to the intended safety function. Indirect deterministic methods (with a probabilistic basis) and prescriptive details have been used for LWRs to achieve probabilistic performance objectives. Advanced reactor technologies will focus on achieving safety performance objectives more directly, placing less reliance on deterministic methods and prescriptive details being used currently.

Because most of the efforts to date related to development of regulatory guidance (e.g., the draft SI NUREG report) has focused on designs similar to those currently being licensed, the existing literature (some of which is discussed in this report) is reflective of traditional LWR designs. All of the regulatory gaps and challenges that apply to LWRs also apply to advanced reactors. Often the LWR case provides a simplified example as compared to the varieties of configuration and risk-based safety approaches found in advanced reactors. Advanced reactor designs may lead to additional gaps and challenges not faced in LWR design. Although this report discusses advanced reactors broadly, the exact set of challenges and potential solutions for any particular reactor design will be technology-specific.

Section 2 of this report describes regulatory challenges and gaps associated with tools and methodologies applied to the seismic design and evaluation of safety-related SSCs. Section 3 discusses regulatory challenges and gaps associated with the certified design licensing process. This section also provides discussion aimed at clarifying legacy terminology that is ambiguous when applied to NPPs using SI technology as part of foundation design. Section 4 addresses facility configurations and applications not addressed in the draft SI NUREG report. Section 0 presents a proposed approach to seismic qualification of the site-specific isolator units consistent with ASCE/SEI 4-16 and the draft SI NUREG report. Section 0 addresses considerations for licensing commitments for construction and operations specific to an isolated NPP. Section 0 summarizes this report and identifies key issues for consideration and possible resolution.

As an aid to the reader, Table 2 maps the three categories of topics (as discussed in Subsection 1.1) to sections in the remainder of this report. Some topics and discussions relate to more than one category. The categories are summarized as follows:

- Category 1 addresses regulatory gaps and design challenges associated with SI of surface-founded or near-surface-founded large LWRs
- Category 2 addresses regulatory gaps and design challenges associated with deeply-embedded advanced reactors
- Category 3 addresses regulatory gaps and design challenges associated with isolation of SSCs contained within a deeply-embedded advanced reactors.

Table 2. Mapping of Topic Categories 1, 2, and 3 to subsequent sections of this report.

Section and Title		Category		
		1	2	3
2.1	Soil-Structure-Interaction Tools and Methods			
2.1.2	Identified Challenges with Legacy Codes	×	×	—
2.1.3	Nonlinear Soil-Structure-Interaction Tools and Methods	×	×	×
2.1.4	Consistency of Site Response Analysis and Soil-Structure-Interaction Analysis	—	×	—
2.1.5	Ground Motion Selection, Modification and Horizon Definition	×	×	×
2.2	Design and Analysis Considerations for the Stop	×	—	×
2.3	Seismic Probabilistic Risk Assessment			
2.3.1	Seismic Probabilistic Risk Assessment Calculations for Isolated Surface-Founded Nuclear Power Plants	×	—	—
2.3.2	Seismic Probabilistic Risk Assessment Calculations for Deeply-Embedded Advanced Reactors	—	×	×
2.3.3	Seismic Probabilistic Risk Assessment for Deeply-Embedded Reactors with Isolated Components	—	—	×
2.3.4	Human Reliability in Isolated Facilities	×	—	×
2.3.6	Screening Tables for Isolated Nuclear Power Plants	×	—	×
3.1	Base-Isolated Nuclear Power Plants	×	—	×
3.2	Terminology and Approach for Licensing Isolated Nuclear Power Plants Using the Certified Design Process	×	—	×
3.3	Challenges and Considerations Related to Vertical Loading of Certified Designs with Base Isolation Systems	×	—	×
3.4	Clarification of Legacy Terminology			
3.4.1	Design Basis Earthquake	×	×	×
3.4.2	Foundation	×	×	×
3.5	Application of 0.1g Minimum Spectrum	×	×	×
4.1	Use of Isolators Exposed to Radiation and Other Uncommon Environmental Conditions	×	—	×
4.2	Isolation of Components, Equipment, and Distributed Systems	×	—	×
5.1	Overview of Proposed Approach to Seismic Qualification	×	—	×
5.2	Testing Requirements	×	—	×
5.3	Commercial Grade Dedication Implementation and Documentation Requirements	×	—	×
6.1	Considerations for Inspection, Testing, Analysis, and Acceptance Criteria	×	—	×
6.2	In-Service Inspections, Testing, and Operations	×	—	×
6.3	Technical Specifications and Descriptions of Seismic Isolation Equipment for Licensing Documentation	×	—	×

2. REGULATORY CHALLENGES AND GAPS RELATED TO TOOLS AND METHODOLOGIES

2.1 Soil-Structure-Interaction Tools and Methods

2.1.1 Historical Perspective and Legacy Codes

Equivalent linear SRA and soil-structure-interaction (SSI) analysis are commonplace in the nuclear power industry at the time of this writing. Such analyses were made possible by studies in the late 1960s and 1970s by a number of expert geotechnical engineers, including Bielak, Idriss, Luco, Lysmer, Ostadan, and Seed, and their collaborators and graduate students (e.g., Lysmer and Kuhlemeyer 1969, Seed and Idriss 1970, Luco and Westman 1971, Veletsos and Wei 1971, Luco and Contesse 1973, Veletsos and Verbic 1973, Veletsos and Meek 1974, Bielak 1975, Bielak and Christiano 1984). Wolf (1985) provides a detailed treatment of the early, seminal studies on SSI analysis.

The underlying principles in the computer codes SHAKE and SASSI, which are widely used for SRA and SSI analysis, respectively, can be traced to the articles identified above, with emphasis on the work by Idriss, Lysmer, Ostadan, and Seed at the University of California, Berkeley. The codes SHAKE and SASSI are described in Lysmer et al. (1999), Ostadan (2006a, 2006b), Ostadan and Deng (2011), and Schnabel et al. (2012) and the reader is referred to these documents for more information.

Equivalent linear analysis, which utilizes secant stiffness and equivalent viscous damping, is likely sufficiently accurate for the purposes of design and risk assessment if soil strains are small and there is no localized relative movement between the foundation and the supporting soil. There is no consensus on the definition of *small* as related to soil strains, but 0.1% would appear to be reasonable (Coleman et al. 2016, Bolisetti et al. 2014, Bolisetti and Whittaker 2015, Coleman et al. 2016). Larger strains may be expected for weaker or more flexible soils at sites of nuclear facilities in the U.S., given shaking at the design basis level and greater, including at the BDB ground motion levels considered in an SPRA.

Although its use is widespread, neither the CFR nor regulatory guidance requires that SASSI, or any other equivalent linear analysis tool, be used for SSI analysis. However, as a result of the long and substantial history described above, the use of SASSI is assumed in a number of current NRC guidance documents. Examples are found in guidance documents issued in the late 2000s in preparation for new license applications. NRC Interim Staff Guidance (ISG) DC/COL-ISG-017 (NRC 2009) was developed to supplement the guidance provided to staff in Sections 2.5 and 3.7 of NUREG-0800 (NRC 2007a). NUREG-0800 is commonly referred to as "the Standard Review Plan" for large LWRs, or "SRP." The guidance in DC/COL-ISG-017 points the reader to Brookhaven National Laboratory Report N6112-051208 (BNL 2009) for guidance on the implementation of the regulatory positions in the ISG. The report contains a detailed discussion of issues and stepwise procedures to implement the technical positions outlined in the ISG; however, the report is specifically focused on technical details that only apply to the use of SASSI. As a result, DC/COL-ISG-017 creates a regulatory challenge for NPPs using SI technology, as well as for isolated and non-isolated advanced reactor designs. Similarly, DC/COL-ISG-017 also defines the SSI not in general terms but as, "a process that is driven by a set of deterministic criteria described in the SRP, Sections 3.7.1 and 3.7.2." It goes on to describe SSI as a series of steps that are, essentially, a description of past practices that (on the surface) appear to limit development or acceptance of advanced SSI techniques.

2.1.2 Identified Challenges with Legacy Codes

Recent documentation issued by the NRC for the mPower reactor (www.generationmpower.com) appears to recognize the special challenges and needs related to SSI analysis approaches for advanced reactor designs. The NRC has stated that the confidence in the ability to implement current SSI methodologies for deeply-embedded NPPs is uncertain. Specifically, the NRC's "Design-specific review standard for mPOWER iPWR design," (NRC 2013) Section 3.7.2, recommends performing a number of

sensitivity studies to quantify the important parameters that affect the response of deeply-embedded structures. Some of the aspects unique to deeply-embedded structures that the NRC identified as needing further investigation are:

- Geometric nonlinear behavior such as separation and sliding between the soil and structure
- Nonlinear dynamic effects in the soil such as dynamic changes in hydrostatic pressure and shear strain (which changes the energy dissipation)
- The extent to which non-vertically propagating shear waves may be more important for deeply-embedded structures than for those with shallow embedment.

These concerns relative to seismic analysis of the mPower design are representative of all deeply-embedded NPP designs, which include a number of advanced reactors. Some aspects (e.g., gapping and sliding) may also be shown to be important for surface-and near-surface-founded structures as well.

2.1.3 Nonlinear Soil-Structure-Interaction Tools and Methods

Nonlinear behavior in SSI has a number of sources, including nonlinear deformation in the soil and/or supported foundation and structure, relative horizontal movement along the soil-foundation interface (termed *sliding*) and separation of the foundation from the soil on either the horizontal or vertical face of the foundation (termed *gapping*). Gapping and sliding may result from combinations of horizontal and vertical shaking and will generally be non-uniform along the soil-foundation interface. Efforts to appropriately model gapping and sliding, and to investigate their potential impact on SSC response and facility risk, have been undertaken at INL as part of a research program focused on advanced SPRAs. In addition, geometric nonlinearities in SSI have been observed in experiments (e.g., Mason 2011) and numerical simulations (e.g., Bolisetti et al. 2015, 2016).

An accurate treatment of large shear strains in soil (involving degradations in strength and stiffness), gapping and sliding, and nonlinear kinematic SSI will require the use of nonlinear time domain codes, such as the NQA-1 (ASME 2015) certified commercial codes ABAQUS (Dassault Systèmes 2015), ANSYS (ANSYS Inc. 2013) and LS-DYNA (LSTC 2013). A nonlinear time domain code for performing NLSSI simulations called MASTODON[6] is being developed using the NQA-1 certified MOOSE framework. The open-source code OpenSees (Mazzoni et al. 2009) and NRC-ESSI (Jeremić et al. 2013) are also available for this purpose, but have not met NQA-1 requirements. In the discussion below, we focus on commercially-supported codes that meet NQA-1 requirements and have formal version control, both of which are generally required for applications in the nuclear industry in the U.S. These codes, once verified and validated for SRA and SSI analysis, could explicitly address the three NRC-identified bulleted items of Subsection 2.1.2.

In recent years, the need for SSI tools capable of assessing behaviors accurately over a wide range of ground motion intensities has been increasingly recognized. Technical needs related to SPRA (which requires accurate modeling of behavior at the design basis and BDB ground motion levels) and implementation of technologies such as SI and nonlinear damping devices has spurred investigation, development, and early deployment of numerical tools for nonlinear (time domain) SSI analysis of NPPs (e.g., Jeremić et al. 2009, 2013; Willford et al. 2010; Bolisetti et al. 2014, 2015, 2016; Coleman et al. 2016). These studies and the nuclear-community interest in nonlinear time-domain analysis prompted writing of the non-mandatory Appendix B in the forthcoming ASCE/SEI 4-16. The provisions in this appendix are written at a high level in recognition of the state-of-knowledge and

[6] https://earthquake.inl.gov/SitePages/Mastodon.aspx

state-of-acceptance of nonlinear time domain SSI analysis in the nuclear community.[7] More specificity and design examples are needed to enable straightforward regulation and widespread acceptance.

The following are those technical research, development, and deployment activities that are currently needed to address the above discussed gaps and challenges. Most are also directly applicable to needs associated with the use of equivalent linear SSI analysis using SHAKE and SASSI.[8] All are needed to support deployment of deeply-embedded advanced reactors by ensuring that design tools can accurately estimate seismic demands and appropriately support design and risk assessment activities. To enable widespread and speedy dissemination, all research-oriented products would be made available via the INL Seismic Research Group website: http://seismic-research.inl.gov/SitePages /Home.aspx.[9]

1. **Develop one-dimensional (1D), two-dimensional (2D), and three-dimensional (3D) nonlinear constitutive models for soils at the sites of NPPs in the U.S. Identify, catalog, and document existing soils models.** Identify new models for development. Write open source code in C++ (http://www.cplusplus.com/) or a similar programming language that is suitable for inclusion in MOOSE (see http://www4vip.inl.gov/research/moose-applications/) and the commercial codes listed previously. Implement each model in MOOSE after verification and validation activities are completed.
2. **Verify 1D, 2D, and 3D nonlinear constitutive models. Develop, solve, and document sample problems**. Follow the verification procedures set forth in ASME (2006) and Oberkampf and Roy (2010), develop a procedure specific to soils suitable for others to follow at a later time.
3. **Plan and execute 1D, 2D, and 3D static and dynamic element-level tests on soil samples**. Plan and execute cyclic, dynamic 2D and 3D simple shear tests at different confining pressures (similar to the Kammerer et al. (2002) tests on undrained Monterey sand) to support validation of new and existing models for analysis at sites of nuclear facilities. Document, curate, and archive tests, test results, instrumentation, and metadata to allow validation of models by others.
4. **Validate 1D, 2D, and 3D nonlinear constitutive models using experimental data**. Validate constitutive models using test data. Follow the validation procedures set forth in ASME (2006) and Oberkampf and Roy (2010); develop a procedure specific to soils suitable for others to follow at a later time.
5. **Verify accuracy of computer codes for SRA and SSI analysis**. Develop trial problems for linear SRA and SSI analysis and solve using finite elements and constitutive models available in commercial and open source codes; investigate utility of alternate shape functions, procedures for discretizing the soil column, and required time step for implicit analysis; use simplified inputs such as wavelets (e.g., Ryan 1994, Bolisetti and Whittaker 2015) and simple soil columns that enable closed form solutions. Follow the verification procedures set forth in ASME (2006) and Oberkampf and Roy (2010).
6. **Plan and execute tests in a 1D geotechnical laminar box**. Plan and execute a coordinated series of tests of alternate soil columns, with and without surface-founded and fully embedded structures, with dry and saturated soils, in a 1D geotechnical laminar box. Place only those soils with verified and

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[7] Although nonlinear SSI tools have not been widely used in the nuclear industry, codes such as ABAQUS, ANSYS, and LS-DYNA are widely used in the building and bridge industries.

[8] SHAKE and SASSI are legacy codes in the nuclear community and have been used for decades. As such, they have not been subject to formal verification and validation of the type described here. It is hoped that the utility (range of application) of these legacy codes will be assessed using the datasets generated in the experiments described here.

[9] This INL Seismic Research Group website is currently used to disseminate research product.

validated constitutive models in the box. Instrument the laminar box with transducers capable of measuring the response of the soil column and the inter- and intra-earthquake-shaking evolution of soil mechanical properties. Input low- and high-intensity ground motions to the base of the soil column to generate linear and highly nonlinear response of the soil columns, suitable for validation of both equivalent linear and nonlinear SRA and SSI codes. Measure basemat and sidewall pressure histories. Document, curate, and archive tests, test results, instrumentation, and metadata to allow validation of models by others. Execute tests in an NQA-1 accredited laboratory.

7. **Validate accuracy of computer codes for SRA and SSI analysis**. Validate commercial computer codes using test data. Follow the validation procedures set forth in ASME (2006) and Oberkampf and Roy (2010); develop a procedure specific to soils and the 1D geotechnical laminar box suitable for others to follow at a later time.

8. **Plan and execute gapping and sliding tests.** Plan and execute a series of earthquake-simulator tests (http://seesl.buffalo.edu) of simple blocks of varying mass distributions and densities, installed atop a shallow soil deposit, with the explicit goal of triggering gapping and sliding. Input 1D, 2D, 3D, and 6-degree-of-freedom inputs or varying intensities. Instrument the soil and the blocks to monitor total and relative responses, bearing pressures beneath the blocks, and changes in material properties (soil only). Document, curate, and archive tests, test results, instrumentation, and metadata to allow validation of models by others. Execute tests in an NQA-1 accredited laboratory.

9. **Validate accuracy of computer codes for gapping and sliding**. Validate computer codes using test data. Follow the validation procedures set forth in ASME (2006) and Oberkampf and Roy (2010).

2.1.4 Consistency of Site Response Analysis and Soil-Structure-Interaction Analysis

SRA is a necessary precursor to SSI analysis, regardless of the type of analysis (e.g., linear, equivalent linear, nonlinear) or the number of components of earthquake input (1, 2, or 3). In NLSSI, for a given type of analysis and a given soil column, and each input acceleration time series, in-column and surface free field spectra generated by SRA should be shown to be identical to those generated by SSI analysis with no structures present. Differences in spectral response should be investigated and explained; possible reasons for differences may include (1) layer thickness, (2) integration time steps, (3) explicit versus implicit analysis, (4) damping formulations, (5) model discretization, and (6) shape functions.

Current NRC guidance related to ensuring the consistency of SRA and SSI analysis is found in DC/COL-ISG-017 (NRC 2009). Similar to other aspects of the ISG discussed in Subsection 2.1.1, the discussions and guidance on this topic are focused around use of SASSI and, as a result, the specific guidance provided does not apply to NLSSI tools; however, the objectives that underlie the discussion in the ISG apply equally to any SSI approach, namely:

- The soil model components of the SRA and SSI analyses must be consistent
- The rock motions, foundation motions, and free field motions at the surface or the hypothetical surface used for the GMRS must be consistent given the overall model.

Generally, demonstrating consistency of the above items using the NLSSI modeling approaches discussed above is straightforward. Additionally, Section 4.2 of NRC Regulatory Guide (RG) 1.208 (NRC 2007b) states:

> *"Often vertically propagating shear waves are the dominant contributor to free-field ground motions at a site. In these cases, a one-dimensional equivalent-linear analysis or nonlinear analysis that assumes vertical propagation of shear waves may be appropriate. However, site characteristics (such as a dipping bedrock surface, topographic effects, or other impedance boundaries), regional characteristics (such as certain topographic effects), and source characteristics (such as nearby dipping*

seismic sources) may require that analyses are also able to account for inclined waves."

Although Section 4.2 of RG 1.208 raises important issues related to the of analysis of sites for which the traditional assumption of 1D vertically-propagating shear waves may not apply (as discussed in more detail below), regulatory guidance on appropriate tools and methods for 2D or 3D analysis is not currently available. NLSSI provides an approach that could be directly applied in these instances to meet both SRA and SSI analysis needs.

An approach to address this issue could begin with using the new capabilities recently added to MASTODON, developed within the MOOSE framework, to understand whether and how waves other than 1D vertically propagating shear waves impact a variety of reactor structural configurations, with a focus on deeply-embedded advanced reactors. Once it is determined which waveforms significantly impact various NPP designs, the waveforms could be tied back to the sources that cause them (e.g., steeply dipping bedrock or soil layers, significant surface waves, etc.). This effort is directly tied to research on appropriate ground motion selection and modification (GMSM) techniques, as discussed in the next subsection.

2.1.5 Ground Motion Selection, Modification and Horizon Definition

GMSM has been a subject of much research for more than a decade, and there is no optimal solution for all applications. A report by the National Institute of Science and Technology (NIST 2011) provides a large body of information on this important topic, which is not repeated here.

RG 1.208, Appendix F, *Criteria for Developing Time Histories*, addresses GMSM, but is now considered out of date. At the time the NRC was developing RG 1.208, a strong technical basis for a more advanced approach for analysis of NPPs did not exist. More recently, the NRC published an update to the SRP, Section 3.7.1. However, there is some controversy regarding the changes related to the requirements for acceleration time series to be used in SSI and it is not clear that the technical basis used for the SRP revision provides the most appropriate approach for isolated facilities, particularly given the findings of recently available research targeted at SI, as discussed below. For this reason, GMSM has been identified as a regulatory gap in this report. This gap exists for both isolated and non-isolated facilities and for both surface-founded and deeply-embedded NPPs.

To date, limited research has been conducted to investigate GMSM for NPP design; and the research that has been conducted has focused exclusively on surface- or near-surface-founded LWRs. The approaches most appropriate for advanced reactors, deeply-embedded reactors, NPPs with height to width aspect ratios different from LWRs, and reactors with risk-significant frequency ranges differ from those of LWRs. As a result, the identified gap related to GMSM is most acute for advanced reactors.

Currently, the state-of-art in SSI analysis utilizes three-component ground motion records applied to the lower boundary of a soil domain (i.e., a rock or hard soil horizon), as shown in Figure 3. These three-component motions are developed using hazard-consistent spectra determined by probabilistic seismic hazard analysis (PSHA). PSHAs for NPPs follow current NRC guidance as described in RG 1.208 and its supporting documentation (e.g., NRC 2007b and NRC 2012). The ground motion inputs used in SSI analysis must be consistent with any SRA performed as part of the PSHA, particularly if the SRA is directly implemented in the ground motion characterization model. This model approach and configuration, which is common for large LWRs, has been used in nearly all of the SSI-related research performed to date. While this configuration may be appropriate for some surface- or near-surface-founded reactors (depending on the geologic profile at the site), there will be additional considerations for deeply-embedded reactors. These considerations including wave incoherency and the relative amplitudes of input motions at the various soil-structure interfaces.

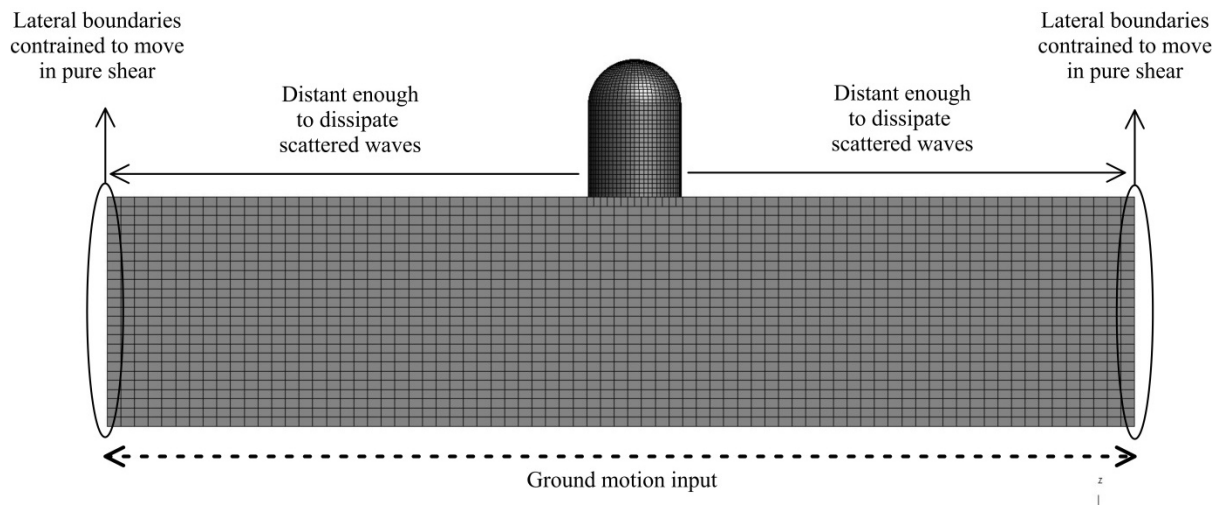


Figure 3. Soil-structure-interaction model (Bolisetti and Whittaker 2015).

Work underway by Bielak and his collaborators at Carnegie Mellon (e.g., Bielak et al. 2003, Xu et al. 2003, Yoshimura et al. 2003, Taborda and Bielak 2011) and the Southern California Earthquake Center (www.scec.org) are substantially advancing knowledge in seismic wave propagation using well-defined seismic sources (e.g., the San Andreas fault) and well-defined geological domains (e.g., the Los Angeles basin), potentially enabling the complete characterization for a scenario earthquake of the seismic wave field around the boundary of the soil domain in close proximity to a nuclear facility in the study region. Longer-term, these numerical simulations of fault rupture and seismic wave propagation may contribute in a meaningful way to PSHA data and tools for certain well-characterized regions in the U.S. However, there are few nuclear facilities in the U.S. located adjacent to well-defined active faults capable of generating large magnitude earthquakes, and even fewer facilities located at sites in geological domains that are defined sufficiently to enable these numerical simulations. Bielak and his colleagues have unparalleled experience and expertise in ground motion simulation and INL will continue to interact closely with his team to stay abreast of developments and any resulting insights that may inform INL research.

INL has recently implemented the Domain Reduction Method as originally developed by Bielak (e.g., Bielak et al. 2003, Yoshimura et al. 2003), in MOOSE. This capability, which has not yet been fully explored, will facilitate an understanding of the relative importance of body and surface waves to the dynamic response of nuclear facilities. Analysis using MOOSE for sample sites and geological domains, and hypothetical fault rupture scenarios, could be used to propagate rupture-induced seismic waves to (1) the 3D boundary of a soil domain that will include the nuclear facilities, and in a subsequent analysis propagate the 3D wave field around the boundary of the soil domain into the facilities; and (2) the center of the lower boundary of the soil domain that will include the nuclear facilities, and in a subsequent analysis apply the three-component time series across the lower boundary of the soil domain and propagate the motions up into the nuclear facilities. Such analyses should be performed for a variety of deeply-embedded advanced reactor configurations to gain a clear understanding of the relative seismic demands imposed by inclined body waves and surface waves.

Although the above discussed shortcomings with the current tools and methods have been recognized for many years, SSI analysis of nuclear facilities will likely continue be performed (for at least the next decade) by imposing a three component time series uniformly across the lower boundary of a soil domain, as shown in Figure 3. Calculations will be performed for ground motions levels with user-specified MAFEs. Uniform hazard response spectra (UHRS) and ground motion response spectra (GMRS) defined at the lower boundary of the soil domain will be generated from seismic hazard curves that are a product of PSHA. Once the appropriate UHRS or GMRS for the lower boundary of the soil domain are available,

three-component ground motions consistent with these spectra must be generated for input to SRA and SSI analysis.

There are at least four procedures for generating three component acceleration time series currently used in the technical community:

1. Spectrally matching ground motions in the time domain to the UHRS.
2. Generating conditional mean spectra (CMS; e.g., Baker and Cornell 2006) and scaling motions to the CMS.
3. Generating conditional spectra (CS; e.g., Jayaram et al. 2011) and scaling motions to the CS.
4. Matching motions to the geomean horizontal UHRS with explicit recognition of the difference in amplitude of the two horizontal components (Huang et al. 2008a, 2009a, 2009b) (which is addressed in ASCE/SEI 7-10 [ASCE 2010]), and matching motions to the vertical UHRS.

The use of the CMS (procedure 1) and CS (procedure 2) requires the analyst to specify a conditioning period, which is typically taken as first mode translational period of the superstructure ignoring SSI. When SSI is considered in development of acceleration time series, the conditioning period will be a function of the degree of nonlinear response of the soil-structure system, which will not be known until the analysis is completed. Importantly, both CMS and CS are geomean horizontal spectra and formal procedures to decompose such spectra into two horizontal spectra that are suitable for selecting and scaling ground motions do not yet exist. Of the remaining two procedures, Procedure 4 is considered the most robust, especially for nonlinear response-history analysis, because it best represents recorded ground motions (Huang et al. 2008a). Kumar et al. (2015b) performed analysis on nonlinear isolation systems (fundamentally no different from a nonlinear soil-structure system) and concluded that procedure 4 was best for generating mean and 90th percentile demand values in an SSI analysis.

As discussed above, a regulatory gap exists due to a lack of definitive guidance on the GMSM of ground motions generally for all SRA and SSI analysis. However, because legacy approaches exist for traditional surface-founded NPPs, the gap is most problematic for facilities with nonlinear soil-structure systems, which includes all NPPs with SI and nearly all advanced reactors.

The following are subject areas requiring technical research, development, and deployment. All three are topics that must be addressed to support the deployment of deeply-embedded advanced reactors. Work in these areas is needed to allow for an accurate estimate seismic demands, which is needed for design and SPRA activities. To enable widespread and speedy dissemination, all research-oriented products would be made available via the INL Seismic Research Group website described previously.

1. **Spectral representation and scaling**. Although procedure 4 above is likely to yield the most realistic results in terms of soil shear strains, isolator displacements and demands on SSCs, this has not been proven. Analysis of example soil-structure systems, intended to represent a range of surface mounted and deeply embedded advanced reactors, should be conducted for ground motions developed using procedures 1 and 4 above, for the purposes of design and seismic performance assessment.
2. **Minimum number of sets of ground motions**. Determine the number of sets of three-component ground motions required to (1) compute design basis demands at the 80th percentile, and (2) perform SPRA.
3. **Guidance development for a future revision of ASCE/SEI 4**.

2.2 Design and Analysis Considerations for the Stop

The need for a physical displacement restraint, or *stop*, is referenced in both the draft SI NUREG report and the SI provisions of ASCE/SEI 4–16. The stop is intended to limit displacement of an isolated superstructure and to limit the deformations in the isolator units and any cross over umbilicals. The minimum unobstructed distance between the isolated superstructure and the stop is set equal to the 90th percentile displacement of the isolated superstructure along each of its horizontal axes for BDB ground motions levels. The 90th percentile displacement will be determined by nonlinear dynamic analysis of a numerical model of the soil-structure system. This model incorporates the proposed mechanical properties of the isolators. Prototype seismic isolators are tested for the maximum horizontal displacement permitted by the stop to confirm that the performance objectives for the isolators are met (see Subsection 5.2 of this report).

The guidance provided in the draft SI NUREG report and the mandatory language of ASCE/SEI 4-16 are applicable to a wide range of superstructures and equipment, ranging from an emergency diesel generator supported by four isolators to a nuclear island supported by hundreds of isolators. The appropriate design details of the stops for these isolated SSCs, which range in weight from tens of tons to hundreds of thousands of tons, will be vastly different. This is one reason that neither the draft SI NUREG report nor ASCE/SEI 4-16 provides detailed guidance on the design of a stop. Other reasons for not providing detailed guidance include (1) the large number of potential configurations of a stop for a given application (e.g., isolation of a nuclear island), (2) the effect of torsional response in the isolated superstructure, leading to impact at a point rather than along an edge, and (3) differences in possible impact velocities depending on facility designs and sites.

An impediment to the implementation of SI in nuclear facilities is the lack of analysis, design and detailing examples for stops. Examples that span a wide range of nuclear structures (e.g., an emergency diesel generator, a stand-alone control building, a turbine building, and a nuclear island) have not been prepared, but would be useful. Although the draft SI NUREG report offers a procedure to identify an impact velocity (i.e., assume the isolation system is displacing cyclically to the 95th percentile BDB displacement at a frequency equal to the inverse of the isolated period [based on post-yield stiffness for nonlinear isolators] and calculate the velocity at the 90th percentile BDB displacement), the BDB data generated by Huang et al. (2009b) and Kumar et al. (2015b) could be mined for more accurate information.

A case for placing a soft stop in series with a hard stop has been advanced for the purpose of dissipating some of the energy transferred at impact. The downsides to this proposal include (1) that the deformation in the soft stop would have to be added to the clearance to the stop to establish the required displacement capacity of the bearings, and (2) construction materials suitable for soft stops and exposure to the environment (e.g., solid rubber blocks and rubber marine fenders) are not particularly flexible. Nonetheless, example soft stops could be developed and designed for selected applications, noting that their utility would be dependent on the flexibility and strength of the supporting elements (e.g., a retaining wall supported laterally by compacted soil).

The following SI technology research and development activities are needed to enable deployment of SI in a commercial NPP. To enable widespread and speedy dissemination, all research-oriented products would be made available via the INL Seismic Research Group website identified previously.

1. **Develop trial designs for stops**. Analyze and design stops for a sample advanced reactor, an isolated reactor vessel, and/or other applicable SSCs.
2. **Investigate benefits and implementation approaches for soft stops**. Identify construction materials suitable for use as a soft stop. Identify implications of implementing a soft stop. Extend the trials designs for stops (above) to include soft stops.

2.3 Seismic Probabilistic Risk Assessment

In the U.S., an SPRA must be accepted by the NRC prior to a loading fuel at a new NPP. A growing number of countries require that an SPRA be performed for all NPP designs. Both the U.S. and South Africa have risk-informed targets that must be met and Japan requires an assessment of the "residual risk," which is the risk that comes from ground motions beyond the design level. In addition to the regulatory requirements for evaluation of the NPP as part of licensing, the role of SPRA in informing design activities also continues to evolve. Although, SI has been shown to reduce facility risk levels, when SPRA is performed for an NPP using SI technology, additional accident sequences may need to be considered and assessed. These include consideration of failure of the isolation system (as discussed above), failure of umbilical lines and other crossover SSCs, and the potential for pounding against the stop, as appropriate. The benefits of SI will be quantified in the SPRA in the form of lower seismic motions in the superstructure.

This section of the report identifies regulatory challenges and gaps associated with an SPRA of isolated surface- or near-surface-founded LWRs, deeply-embedded reactors, and deeply-embedded reactors in which components and systems are isolated. Another important area of risk improvement that is not being captured at present relates to the probability of success of operator action (human factors) if SI is employed, as discussed below. Risk metrics for some advanced reactors may be different from those for LWRs, for example, core damage frequency may not be relevant. Below we emphasize *unacceptable performance* rather than *core damage frequency* (for example) in an attempt to be technology-neutral.

2.3.1 Seismic Probabilistic Risk Assessment Calculations for Isolated Surface-Founded Nuclear Power Plants

Traditional SPRA methods that employ a standard spectral shape anchored to peak ground acceleration (PGA) cannot be used for nuclear structures isolated with nonlinear bearings because in-structure response spectra (ISRS) demands do not scale proportionally with shaking intensity.

Examples of risk-based calculations for assessing adequacy of an SI system design and demonstrating the risk-benefits of seismically-isolated surface or near-surface-founded LWRs are provided in Huang et al. (2008b, 2011a, 2011b), Kumar et al. (2015b), and Kumar and Whittaker (2016). The Huang et al. report and papers provide a methodology for the seismic performance assessment (risk calculations) of an isolated NPP equipped with linear (low damping rubber) and nonlinear (lead rubber, friction pendulum [FP]) bearings. The Huang et al. calculations assume a rigid surface beneath the isolation systems, which is extended conceptually below to address compliant nonlinear soil and deeply-embedded reactors. Huang et al. (2008b, 2011b) quantify the benefits of implementing SI for sample, simple accident sequences.

The Huang methodology, as described in the report and papers noted above, utilizes three-component earthquake records that are selected and scaled to a wide range of MAFEs (e.g., from greater than 10^{-4} to less than 10^{-5}). This provides a sufficient range to enable calculations of risk to a MAFE of less than 10^{-7}. Figure 4 presents a sample PSHA-developed seismic hazard curve upon which eight intensities of shaking, denoted as e_i, are identified. A sample selection of scaled ground motions for two intensities of motion is presented in Figure 5.

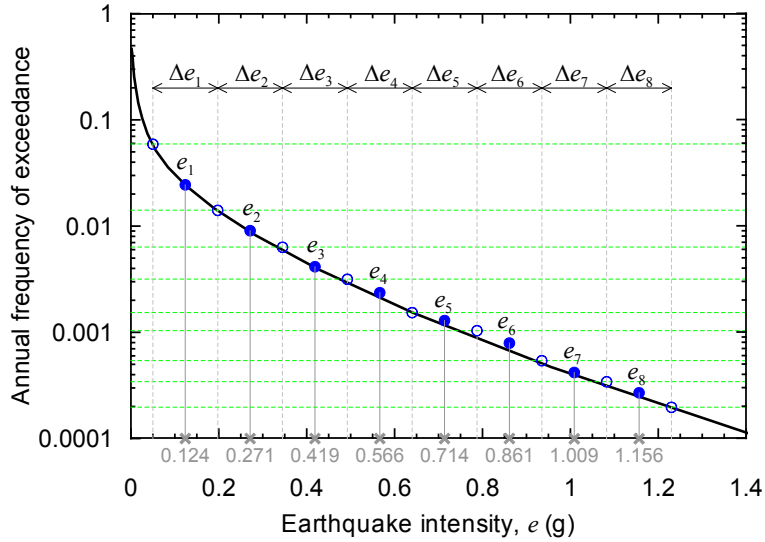


Figure 4. Intensities of earthquake shaking for seismic probabilistic risk assessment calculations (Huang et al. 2008b).

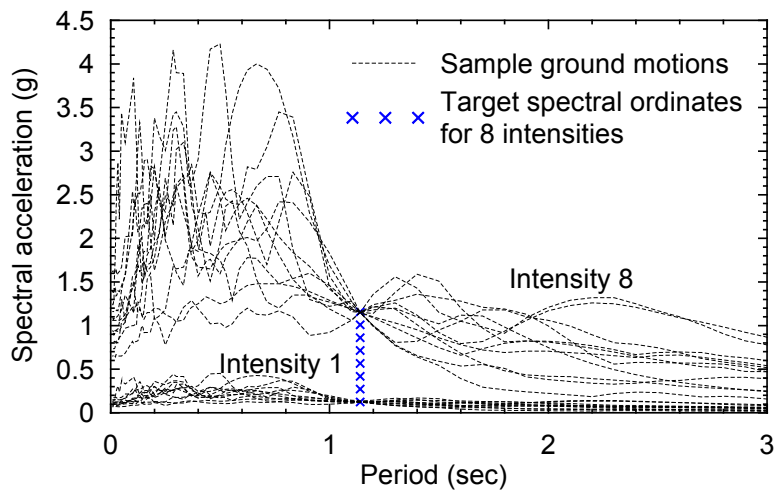


Figure 5. Spectral accelerations for 11 ground motions at intensities e_1 and e_8 (Huang et al. 2008b).

Traditional plant response analysis involves the use of event trees (accident sequences) and fault trees. Structural analysis is undertaken for each scaled earthquake record set at each intensity of ground motion to calculate a matrix of demands on SSCs. Statistical procedures based on Yang et al. (2009) and FEMA (2012) are then employed to expand the size of the matrix from approximately ten entries per intensity to thousands of entries per intensity. The expanded matrix is sufficient to enable Monte Carlo simulation. Monte Carlo simulations are performed at each earthquake shaking intensity using component-level demands and fragility functions (functions that plot the conditional probability of failure versus local demand parameters, such as floor spectral acceleration or story drift) to establish failure probabilities for input into the plant response analysis and enable the calculation of the conditional probability of unacceptable performance (e.g., core damage for a LWR). The products of the conditional probability of failure at each shaking intensity and the mean annual frequency of shaking at that intensity are summed over the entire intensity range to compute the mean annual frequency of unacceptable performance. This peer-reviewed methodology is referenced in the forthcoming ASCE/SEI 4-16.

One possible accident sequence initiator for a seismically-isolated NPP is failure of the isolation system, wherein the isolation system is unable to sustain axial loads imposed by gravity and earthquake shaking at the horizontal (lateral) displacements expected in BDB shaking. As described above, a physical *stop* is introduced in ASCE/SEI 4–16 and the draft SI NUREG report to limit the horizontal displacement of an isolated superstructure to well below its capacity. These requirements for implementing a stop and dynamic testing of prototype isolators to the lateral displacement defined by the stop are intended to ensure the mean annual frequency of failure associated with isolation-system failure is less than 10^{-6} if the recommended NRC performance criteria are used. Other possible accident sequence initiators relate to the failure of the safety-related umbilical lines that cross the isolation interface (i.e., from isolated construction to non-isolated construction).

Kumar et al. (2015b) and Kumar and Whittaker (2016) provide a risk-calculation procedure for SI systems and safety-related umbilical lines that cross the isolation interface. Calculations of risk for the isolated superstructure are not provided in Kumar et al. (2015b) and Kumar and Whittaker (2016) because they can be made following the procedure described above. Fragility curves are developed for the hypothetical isolation systems (and umbilical lines) of NPPs located at eight NPP sites across the U.S. These calculations assume that the performance goals for the isolation system outlined in the draft SI NUREG report and ASCE/SEI 4–16 are satisfied. These goals include 100% confidence that the isolation system (and umbilical lines) will survive DBE ground motion without damage,[10] and 90% confidence that the isolation system will support gravity and earthquake-induced loadings at the 90th percentile BDB earthquake displacement (i.e., the minimum clearance to the stop). For the cases analyzed, the introduction of the stop into the NPP design reduces the seismic risk associated with failure of the isolation system by approximately an order of magnitude.

The following technology research and development activities are needed prior to commercial deployment of SI in advanced reactors. To enable widespread and speedy dissemination, all research-oriented products would be made available via the INL Seismic Research Group website identified previously.

1. **Implement the Huang et al. SPRA-based methodology for assessment of the SI system described above in MOOSE**. An advanced SPRA-based methodology is required to assess the risk of failure of the foundation system in seismically-isolated nuclear facilities. The Huang et al. (2008b, 2011a, 2011b) SPRA-based methodology should be implemented in MOOSE to enable risk assessment of isolated nuclear structures.
2. **Implement isolator-related accident sequences in MOOSE**. The introduction of an SI system will substantially reduce the mean annual frequency of unacceptable performance, and Huang et al. (2008b, 2011a, 2011b) quantified the benefits. The use of isolation will add accident sequences to the SPRA, including those associated with (1) failure of the isolation system, and (2) failure of the safety-related umbilical lines that cross the isolation interface, and these should be generically implemented in MOOSE.
3. **Characterize the assumption of isolator correlations**. The fragility functions of Kumar et al. (2015b) and Kumar and Whittaker (2016) are biased low because the failure of one isolator is assumed here to represent failure of the isolation system. This is a very conservative assumption that assumes that the isolator axial force and horizontal displacement demands are fully correlated, which will almost never be the case. The impact of this assumption on risk calculations should be quantified for sample applications.

[10] The confidence levels for design basis ground motions come from physical testing of 100% of the isolators to be placed into service to the DBE displacements (see Subsection 5.2Section 5.2 for more information).

4. **Perform SPRA calculations for sample isolated nuclear facilities**. The analysis, design, and risk assessment of one or more example isolated nuclear facilities, coupled with the careful documentation thereof, would speed the implementation of SI technology by laying out the process to analysts, designers, and regulators.

2.3.2 Seismic Probabilistic Risk Assessment Calculations for Deeply-Embedded Advanced Reactors

A deeply-embedded reactor will experience seismic loads along its basemat and walls. These loads cannot be characterized by a surface free field acceleration response spectrum for analysis purposes. SRA and SSI analysis for design and SPRA for such reactors will have to employ an alternate horizon control point to define seismic input. This horizon to be used for analysis will likely be located at some depth below the reactor.

The Huang et al. methodology described in the subsection above, developed for SPRA of isolated surface-founded LWRs, is directly applicable for risk calculations for deeply-embedded advanced reactors. Soil above the rock horizon and surrounding the embedded reactor is modeled explicitly (replacing the nonlinear isolators in Subsection 2.3.1) and the three-component ground motions are applied at the rock horizon below the base of the embedded reactor (replacing the motions applied at the underside of the isolation system in Subsection 2.3.1).

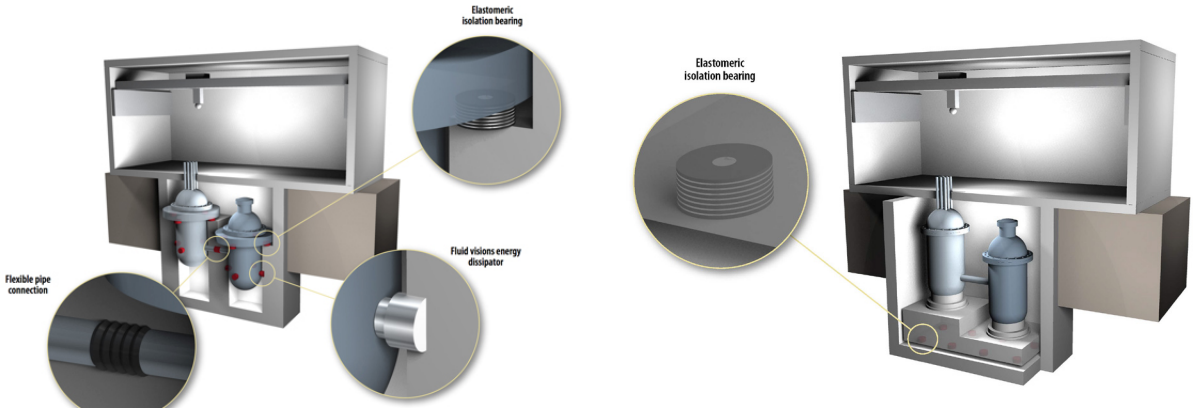
The calculations proceed identically to those described in Subsection 2.3.1. At each intensity of motion, between five and 11 three-component sets of ground motions are propagated through the nonlinear soil-foundation-structure system. Multiple soil columns (e.g., best estimate, lower bound, upper bound) should be analyzed for each three-component set of ground motions to account for variability in the mechanical properties of the soils. Mean seismic demands (e.g., peak accelerations, peak spectral accelerations, floor spectra, story drift) are calculated from the nonlinear response-history analyses. These demands are then input into a systems model to calculate the conditional probability of failure. Again, the products of the conditional probability of failure at each shaking intensity and the mean annual frequency of shaking at that intensity are summed over the entire intensity range to compute the mean annual frequency of unacceptable performance.

The following technology research and development activities are needed to support commercial deployment of deeply-embedded advanced reactors, regardless of whether or not they utilize SI technologies. The activities listed below are predicated on (1) implementation of the Huang et al. SPRA-based methodology discussed in this subsection in MOOSE, and (2) availability of robust nonlinear constitutive models for soil and verified and validated computer codes for NLSSI analysis (see Subsection 2.1.3). To enable widespread and speedy dissemination, all research-oriented products would be made available via the INL Seismic Research Group website identified previously.

1. **Develop trial design of deeply-embedded advanced reactors**. To enable SPRA of deeply-embedded advanced reactors, sample designs or test beds are needed for the civil/structural system and safety-related SSCs. Designs would be accompanied by fragility functions for incorporation of safety-related SSCs. Sites and soil profiles consistent with possible DOE construction of advanced reactors, including INL, should be selected to maximize the benefit of the test bed designs.
2. **Develop accident sequences for the sample reactor designs**. Event trees and fault trees will be needed to develop systems models for the sample reactor and enable SPRA.
3. **Perform SPRA calculations for the sample reactors**. The analysis, design, and risk assessment of sample reactors (one reactor type, multiple sites, multiple soil profiles), coupled with the careful documentation thereof, would laying out the process to analysts, designers, and regulators. Importantly, the SPRA calculations and the disaggregation of the seismic risk would enable the risk-informed design of advanced reactors: achieving the desired level of safety at minimum cost.

2.3.3 Seismic Probabilistic Risk Assessment Calculations for Deeply-Embedded Reactors with Isolated Components

It is unlikely that deeply-embedded reactor buildings will be seismically (base) isolated for two reasons. First, there are significant additional initial construction costs associated with the over-excavation of the site. The additional excavation is required below the basement to enable placement of the foundation, pedestals, and isolators and around the building to provide uninterrupted space for the isolated building to displace. Secondly, there is additional cost associated with construction, monitoring and maintenance of a permanent retaining wall surrounding the embedded building. These additional costs will likely outweigh the benefits in terms of reduced seismic inputs. However, deeply-embedded reactors may benefit significantly from the isolation of safety-related components such as reactor vessels, as measured by increased seismic safety and/or reduced cost. Figure 6 presents cut-away views of applications of seismic protective systems, defined here as a combination of SI devices and energy dissipation devices such as fluid viscous dampers. This idealized advanced reactor is embedded similarly to that shown in Figure 1.



a. Component isolation b. Internal system isolation

Figure 6. Deeply-embedded advanced reactor incorporating seismic protective systems.

The following are technology research and development activities that should be conducted to enable deployment of SI at an NPP. The activities listed below are predicated on the prior completion of the three tasks of Subsection 2.3.2. The sites, soil profiles and civil/structural systems identified in the tasks of Subsection 2.3.2 would be used here.

1. **Develop trial designs of protective systems for components**. Selected safety-related SSCs would be identified for possible isolation. Seismic design *spaces* would be developed for two cases for each SSC. In the first case, the SSC capacity would be based on non-seismic load cases. In the second case, the capacity would be based on the designs of Subsection 2.3.2. Protective systems (isolators and dampers) for each case, site and soil profile would be developed. All designs would be priced to establish the benefit, if any, of the use of protective systems.
2. **Develop accident sequences for the protective systems and safety-related umbilical lines**. Event trees and fault trees will be needed to augment the plant response analysis of Subsection 2.3.2.
3. **Perform SPRA calculations for the sample reactors**. The analysis, design, and risk assessment of the sample reactors (one reactor type, multiple sites, multiple soil profiles), coupled with the careful documentation thereof, would provide a methodology that could be used by analysts, designers, and regulators. Importantly, the SPRA calculations would inform and guide the possible use of protective systems in deeply-embedded reactors.

4. **Development of isolation systems for large systems and components**. The optimal isolation systems and seismic isolators for systems and components of advanced reactors may not be those currently available in the marketplace because current SI technologies focus on horizontal isolation of buildings with a low to moderate aspect ratio. Components and major systems of advanced reactors (and LWRs) have geometries that are often quite different from buildings and many may be supported for gravity loading above their base. The preparation of trial designs above may inform the development of next generation seismic isolators and damping devices, which would then require design, prototyping and testing—following the procedures set forth in ASCE/SEI 4-16 and the draft isolation NUREG report. Next generation protective systems will likely include vertical isolation systems, which will permit 3D isolation and seismic protection when used in conjunction with currently available isolators. Guidance for implementation of isolated components in advanced reactors (and LWRs) would have to be developed for inclusion in a future revision of ASCE/SEI-4 and other guidance.

2.3.4 Human Reliability in Isolated Facility

The approaches currently applied for quantification of reliability of operator action in SPRA have been in use for decades and, until recently, few advances have been developed and proposed for use. Generally, the probability of failure of a manual action in a seismic event is treated as a set percentage (e.g., 50%) of the expected probability of failure for a non-seismic event, which itself has a high degree of uncertainty. The probability of failure includes an assumption that some portion of the NPP operations staff is unable to perform their function due to a blocked access path, injury, or some other factor. Although seismic walkdowns often look for the potential for important pathways to be blocked as a result of physical damage to SSCs during a seismic event, a systematic treatment that includes a thorough investigation of all pathways that may be needed is not universally applied. This is despite the current understanding that a large percentage of accident sequences that are determined to be risk-significant in SPRAs have some element of operator action. Given this current state of practice, the benefit of SI in terms of risk reduction resulting from improved human performance would not be fully incorporated in the SPRA model and results.

Recently, work by EPRI (Presley et al. 2013a, 2013b) has identified a new approach that bases the probability of failure of operator action on plant damage state. This approach, though in the early stages of development and implementation, provides an opportunity to include a more technically-advanced approach to quantify risk in both isolated and non-isolated facilities. INL will follow developments in this subject area and will implement the approach to the extent possible within the Advanced SPRA Program. Because of the importance of operator actions, this advanced approach has potential to significantly improve both the design process (to the extent that SPRA is used to inform design, as discussed below) and the accuracy of assessments that investigate the cost-benefit balance of incorporating SI into a particular NPP design.

2.3.5 Opportunities for Seismic Probabilistic Risk Assessment Feedback to Design

SPRA performed as part of new reactor licensing activities has most commonly been performed towards the end of the design process. Although current performance and risk objectives are set facility-wide, they are most typically applied on an SSC by SSC basis, which leads to inefficiencies in NPP design. Infusing insights from risk calculations into the various stages of a design process should (1) alleviate the need to mitigate unacceptable risks at the end of the design process, (2) enable cost-effective design of safety-related SSCs based on risk-targeted demands, (3) enable the appropriate choice of structural framing systems that support safety-related SSCs, and (4) enable a risk-informed design of isolation systems that effectively balances risk and cost (isolators, safety-related umbilical lines, superstructure and SSCs). This is particularly true for advance reactors with new and innovative designs.

The level of granularity of a design increases as it proceeds, from conceptual design through construction documentation. Conceptual design involves plant layout and decisions regarding framing systems and types and locations of SSCs. Risk calculations involving the most important accident sequences can be performed at this stage to assess the required capacity of key components, such as reactor vessels and steam generators. The cost of safety-related components as a function of their mean strength (a proxy here for incipient failure), which is likely a series of step functions and not linear, could (should) be used to select the framing system (potentially with SI technology) needed to support the components to minimize cost and meet target risk objectives. Such an approach would require a paradigm shift in the design process but should lead to significant cost reductions. The products of a risk-informed, conceptual design would be a viable structural framing system, a trial design and sizing of safety-related SSCs, risk calculations, and cost estimates for construction.

A preliminary design (or design development) advances a conceptual design to the point where framing systems are finalized, SSCs are crisply identified, piping runs finalized, etc. The number of accident sequences that can be identified and quantified is much greater than at the conceptual level, and the conceptual level accident sequences can be refined and updated. Risk calculations are performed in much greater detail, with more accurate demands, accident sequences, and fault trees. A draft preliminary design would be updated on the basis of the preliminary design risk calculations, with attention paid to SSCs and accident sequences with either unacceptable risk or too great a contribution to the seismic risk profile.

Final design (or construction documents) advances a preliminary design to the point where framing is designed component-by-component, SSCs are specified for construction, and documents are prepared for review by the regulator. The corresponding SPRA would be as detailed as those currently undertaken but would include refinements and updates from the calculations prepared in conjunction with the preliminary design. The mean annual frequency of unacceptable performance calculated at the end of the design process should be within 20% of that calculated at the end of preliminary design.

To date, no guidance has been developed to describe an efficient process for incorporating this risk information into design. This process would support licensing efforts by making risk targets an integral aspect of design, along with the more traditional margins approaches.

2.3.6 Screening Tables for Isolated Nuclear Power Plants

The incorporation of currently available SI technology reduces the horizontal response of a structure to horizontal ground motion through the installation of horizontally flexible and vertically stiff seismic isolators between the superstructure and its foundation. Whereas the horizontal accelerations are significantly reduced by the isolation layer, the vertical motions are relatively unchanged.

This leads to a question of the applicability of commonly used screening tables, such as those in EPRI Report NP-6041-SLR1 (EPRI 1991). These screening tables provide agreed-upon ground motion levels at which certain classes of equipment can be screened out of additional fragility calculations based on information in the earthquake experience database associated with the components. The tables list equipment that has been shown to be robust to beyond the screening level during large seismic events in which the equipment was in service in industrial facilities. Because the equipment in the current database was universally housed in non-isolated facilities, the vertical to horizontal (V/H) ground motion ratios (commonly called the V/H ratio) the equipment experienced would be very different than those that would occur in an isolated facility. Therefore, as a result of the significant reduction of horizontal motion in an isolated facility, the V/H ratio will be much higher than for the ground motions represented in the earthquake experience database. Although it is believed that few SSCs are sensitive to vertical motions (with batteries being one known exception), it may be necessary in the future to address the use of the screening tables in SPRA for isolated reactors before the SPRA results are used for design or regulatory purposes.

3. REGULATORY CHALLENGES AND GAPS RELATED TO THE CERTIFIED DESIGN PROCESS

3.1 Base-Isolated Nuclear Power Plants

The current approach for comparing the assessed ground motions at a site (based on PSHA results) against the ground motions used for a certified design is presented in DC/COL-ISG-017 (NRC 2009). This ISG further references NEI (2009) for additional details related to three possible approaches that can be applied, the choice of which depends on the NPP configuration. The possible configurations described reflect the LWRs going through licensing activities at the time of publication. Regardless of the approach utilized, the evaluation fundamentally compares the foundation input response spectra (FIRS), which is based on the site-specific GMRS, to the certified seismic design response spectra (CSDRS), as shown in Figure 7, which is a modified version of Figure 3-1 in NEI (2009).

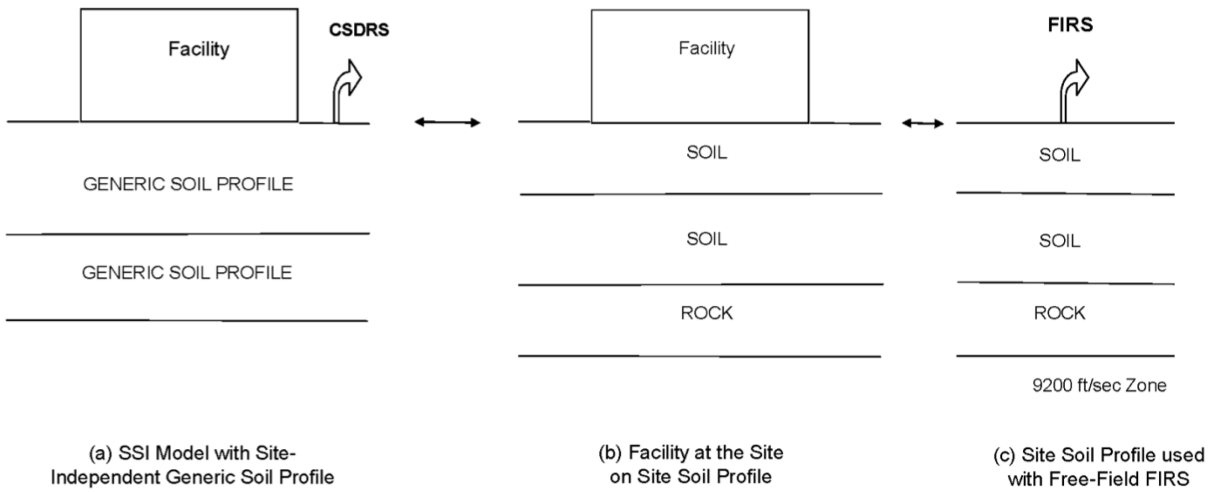


Figure 7. Approach for comparing site-independent certified seismic design response spectrum to site-dependent foundation input response spectrum for surface-founded nuclear power plants with conventional foundations (modified for clarity from Figure 3-1 of NEI [2009]).

For purposes of this comparison, DC/COL-ISG-017 defines the following terms:

- **CSDRS.** *Are site-independent seismic design response spectra that have been approved under Subpart B, "Standard Design Certifications," of Title 10 of the Code of Federal Regulations, Part 52 (10 CFR Part 52), "Licenses, Certifications, and Approvals for Nuclear Power Plants," as the seismic design response spectra for a nuclear power plant using an approved certified standard design.*
- **FIRS.** *The performance-based site-specific seismic ground motion spectra at the foundation levels in the free field are referred to as the FIRS and are derived as free-field outcrop spectra. The FIRS is the starting point for conducting a soil structure interaction (SSI) analysis and making a one-to-one comparison of the seismic design capacity of the standard design and the site-specific seismic demand for a site. The FIRS for the vertical direction is obtained with the vertical to horizontal (V/H) ratios appropriate for the site.*
- **GMRS.** *Are site-specific and are characterized by horizontal and vertical response spectra determined as free-field motions on the ground surface or as free-field outcrop motions on the uppermost in situ competent material using performance based procedures in accordance with Regulatory Guide (RG) 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion".*

Although the above approach comparing the FIRS to the CSDRS has been successfully applied for conventionally-founded LWRs, it cannot be directly applied to facilities with even simple base isolation configurations without clarification/modification. The principal issue is that, in a base-isolated structure, the CSDRS and FIRS are not unambiguously co-located at a single control point (such as the soil surface in the example shown in Figure 7). Instead, the CSDRS location (which is site-independent and defined at the base of the certified portion of the facility) is separated from the most appropriate FIRS location (which is site-specific and needed for overall structural analysis) by the isolation system, as can be seen in Figure 8. The isolation system is designed on a site-specific basis that depends on properties of both the site-specific hazard and the site-independent facility. Although the isolation system is site-specific and is an important contributor to the input motions used for design, it is inappropriate to incorporate it into the GMRS, which is principally an earth-science-based descriptor, as described in NRC RG 1.208 (NRC 2007b). This represents a significant gap in the application of the certified design process to isolated NPPs generally.

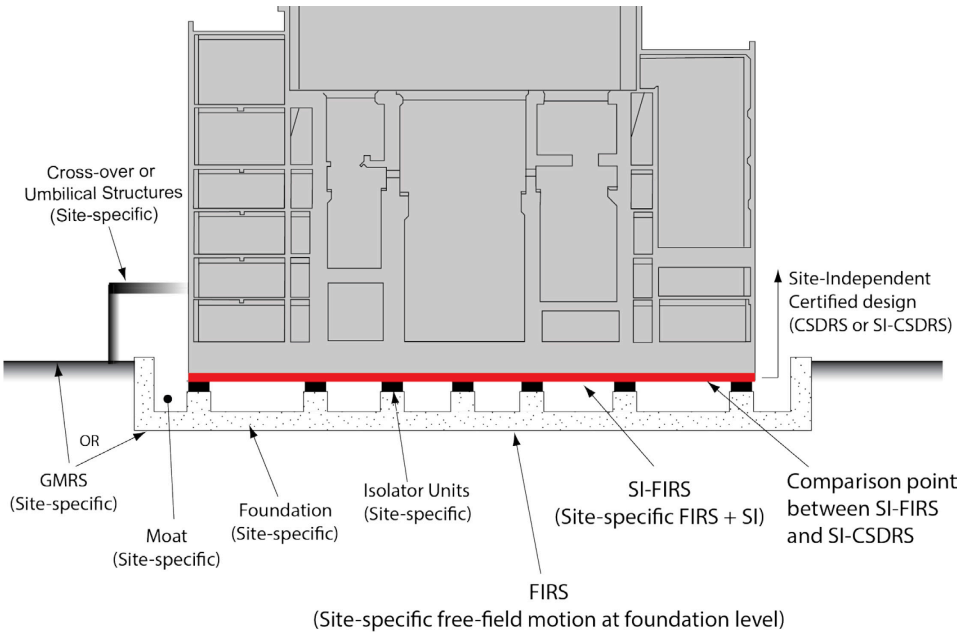


Figure 8. Definitions for evaluation of site against certified design for surface-founded, base-isolated nuclear power plant.

3.2 Terminology and Approach for Licensing Isolated Nuclear Power Plants Using Certified Design Process

A modified regulatory approach, such as that described above, should be identified and developed to close the gap in the certified design process related to SI. This modified approach will require new terminology to bring clarity to the design, review, and licensing process. Figure 8 identifies two new definitions proposed for use in the licensing of surface-founded base-isolated structures using the certified design approach. These proposed definitions are *seismically-isolated foundation input response spectrum* (SI-FIRS) and *seismically-isolated certified design response spectrum* (SI-CSDRS) and are defined in Table 3. SI-FIRS and SI-CSDRS are both defined at the level of the upper basemat and provide the appropriate point for comparison, as shown in Figure 8. The terminology can be used to distinguish between licenses or license applications where a certified design can also be conventionally founded.

Table 3. Definitions of proposed terms for use with seismically-isolated certified designs.

Term	Definition
SI-FIRS	A site-specific motion defined at the underside of the basemat and immediately above the plane of isolation in a base-isolated facility.
SI-CSDRS	A site-independent motion in a base-isolated facility, consistent with Subpart B of 10 CFR 52 and defined at the underside of the basemat and immediately above the isolation system.

Because this issue is newly-identified and the consideration of approaches to address it are in the very early stages, a near-term effort may be needed to further develop the concepts and appropriate supporting documentation to the NRC. This issue was originally identified during application of SI to a LWR, and so the proposed path forward described above may only be applicable to surface-founded or near-surface-founded, base-isolated NPPs with isolators installed in a horizontal plane.

Chapter 12 of ASCE/SEI 4–16 proposes SI design response spectra (SIDRS) for use with a multi-step process to account for the effects of SSI. The SIDRS are site-specific and are generated at the underside of foundation and include the effects of SSI. Accordingly, SIDRS are different from both SI-FIRS and SI-CSDRS.

Many advanced reactor designs are deeply embedded with basemats at multiple levels. Although isolation of the enclosure surrounding the reactor vessel is unlikely for deeply embedded designs for the reasons given in Subsection 2.3.3, it is possible. In such a case, and for a structure with basemats dependent on at multiple levels, neither a single SI-FIRS nor a single SI-CSDRS may provide the clearest and most optimal solution because the isolators at different levels will experience inputs that will be site-specific depending on local characteristics, including the dynamic characteristics of the soil below and surrounding the NPP. If a component within an advanced reactor is isolated (see Figure 6), it may experience out-of-phase seismic inputs at multiple levels. Alternate approaches to SI-FIRS and SI-CSDRS may be beneficial in these cases.

The following are two tasks requiring a combination of study, research, and development.

1. **Analysis of case studies**: Analysis of a variety of case studies, with some possibly including vertical isolation or seismic protection systems (see section below), would help to ensure that new guidance would be appropriately technology-neutral. The tasks identified in Subsection 2.3.3 should be completed before case studies involving SI of components in deeply-embedded reactors are prepared.
2. **Development of a report to NRC**: One approach for resolving the gap for surface-founded, base-isolated NPPs may be to submit for NRC consideration a report that can be used as a basis for further technical and regulatory discussions. This approach, which was taken by the Nuclear Energy Institute to address earlier issues related to the certified design process, led to development of DC/COL-ISG-017, which resulted in reduced regulatory risk and faster licensing reviews. Because the NRC intends to soon publish the draft SI NUREG report, which was developed to form the technical basis for a new RG on SI, activities that address significant gaps (such as the one discussed here) would benefit the timely licensing of isolated facilities in the near future.

3.3 Challenges and Considerations Related to Vertical Loading of Certified Designs with Base Isolation Systems

As discussed above, SI can significantly reduce the horizontal response of an isolated superstructure to incoming seismic motions. The properties of a base isolation system can be specified such that the incoming horizontal shaking level is reduced to below the SI-CSDRS. However, base isolation as commonly applied cannot be used to modify vertical motions entering the superstructure. Further, the typical tools for determining V/H ratios are inappropriate for defining the design spectra of seismically-isolated NPPs. For this reason, designers of seismically-isolated NPPs need to consider carefully the vertical spectrum that is used for a certified design. Although it has not been considered to date, one approach may be to evaluate carefully the equipment within an NPP design that may be sensitive to vertical motions and to develop not one, but a set of certified designs that each utilize the same horizontal design spectra but also incorporate sets of minor modifications, each tied to a specific vertical design spectrum.

Consideration may also be given to exploration of newly developed vertical isolation systems, although the challenges related to applying technologies that do not have the same extensive history of use as rubber and FP bearings require additional activities prior to their use. Chapter 12 of ASCE/SEI 4-16 and the draft SI NUREG report provide a protocol for demonstrating that a new isolation technology is appropriate and its safety can be assessed.

3.4 Clarification of Legacy Terminology

3.4.1 Design Basis Earthquake

In the past, a number of terms were used—generally interchangeably—to represent the ground motions to which a nuclear plant was designed. One such term applied was "design basis earthquake" or "DBE." This terminology was originally used because the ground motions utilized in design were based on a deterministic assessment that considered past earthquakes in a region, or earthquakes thought to be possible given the local seismic sources. These ground motions, defined in the form of response spectra were developed using a standardized spectral shape (see, for example, the shapes in NRC RG 1.60 (NRC 1973) anchored to a site-specific PGA. The standard spectral shapes were developed based on the analysis of the relatively few ground motion recordings that existed at the time. The ground motions determined from the hazard studies were used directly as the design input ground motions for the facility. Therefore, the site-specific ground motions were associated directly with the site-specific design.

In ASCE/SEI 43-05, the DBE is defined as the, "Response spectra used for design. The DRS are equal to the product of the UHRS and the Design Factor and are defined at a control location in the free field." However, even at the time of publication of that Standard, this definition applied only to NPPs that were licensed under 10 CFR 50 where the site-specific design was based directly on a site-specific ground motion. NPPs designed and licensed using the certified design process detailed in 10 CFR 52 were designed primarily to site-independent CSDRS. To address this disconnect, the term GMRS was created when the NRC developed RG 1.208, which was intended to be used with ASCE/SEI 43-05. The GMRS for a site is developed using the same methods used to develop the DBE ground motion spectra in ASCE/SEI 43-05 (described above). In a plant licensed under 10 CFR 50, the GMRS would typically be used as the basis for defining the safe shutdown earthquake (SSE) ground motion spectra to which the plant is designed. In a plant licensed under 10 CFR 52, the GMRS is used to define the FIRS that is compared to the CSDRS. It is also used for design of the elements of the plant not included in the certified portion of the NPP and therefore, becomes the "Site-Specific SSE" ground motion. The CSDRS, which is used for the certified portion of the NPP, becomes the SSE ground motion. In a certified design, it is inconsistent to use the term DBE for both the site-specific SSE (or GMRS) and the SSE from the CSDRS.

The incorporation of SI technology and the application of risk-informed design factors also create circumstances in which the assessed ground motions in the native materials are not directly applied to the design of the reactor. Because the term DBE is most closely tied in the U.S. regulatory framework with the design and licensing basis of some currently operating NPPs (regardless of how the ground motions were developed), it is suggested here that, moving forward, the term DBE should be used only when referring to the seismic vibratory motions to which the plant has actually been designed. This change should be suggested to code committees working on relevant codes and standards. This means that the term DBE should be associated with the SSE ground motion used for design moving forward, rather than with the GMRS or site-specific SSE.

3.4.2 Foundation

In the past, the definition of the word "foundation" as applied to an NPP facility was relatively straightforward because the foundation was more or less a monolithic surface-founded or near-surface-founded structure that moved as a single structural element when subjected to seismic loading. This structural element was in contact with both the geologic foundation materials (or some type of engineered fill/construction material) and the superstructure. When considering an isolated facility, the meaning or interpretation of the word "foundation" as written in the CFR or regulatory guidance documents may not be straightforward.

Both ASCE/SEI 4-16 and the draft NRC NUREG report provide definitions that describe and clarify the terminology used in those guidance documents, as shown in Figure 9. However, the definition of "foundation" as defined in ASCE/SEI 4-16 and the draft SI NUREG report cannot be blindly applied retroactively for interpreting the meaning of "foundation" as used in in the CFR and other existing guidance documents. For example, a particular discussion in a guidance document could be focused on a function that would be performed by the base mat in an isolated system. In the draft SI NUREG report, the upper base mat is defined as separate from the elements that compose the "foundation" (see Figure 9). The term "foundation" in existing guidance applies to the foundation (which is comprises those elements in contact with the ground), the upper base mat (which directly supports the superstructure), or the entire "isolation system" (which includes the foundation, the pedestals, the isolation layer, and the upper base mat). In some deeply-embedded advanced reactors, other conflicts may arise.

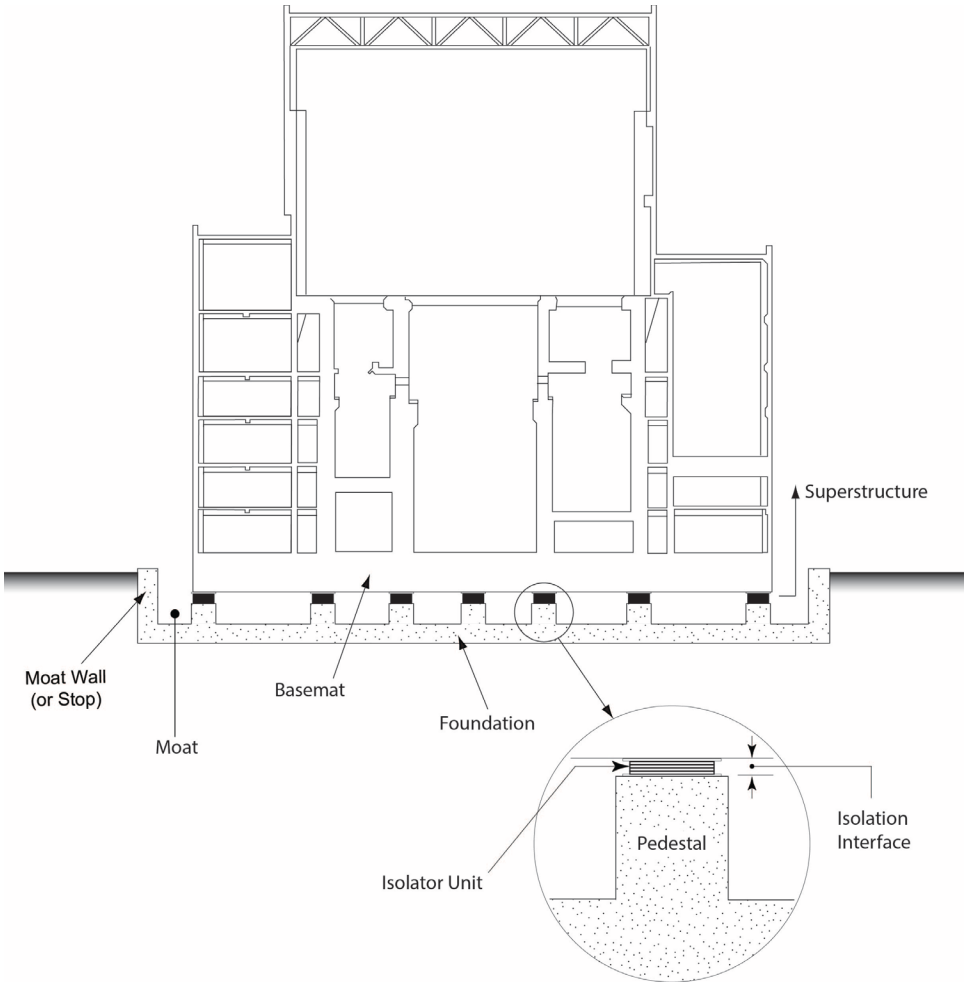


Figure 9. Definitions used in ASCE/SEI 4-16 and draft seismic isolation NUREG report.

An approach to more robustly understanding and addressing this issue may be to undertake the following steps:

1. **Identify the use of the word "foundation" in the relevant regulatory documents and references.** This could start with the code, appropriate RGs, and NUREG-0800.
2. **Define precise SI-appropriate terms that describe the various elements that perform the functions of a foundation in a conventional LWR**. The definitions provided in ASCE/SEI 4-16 and the draft SI NUREG report would be an appropriate starting point for this, although additional definitions or terms may need to be developed.
3. **Identify the intention of the various foundation-related discussion/requirements in the regulatory documents and map the usage to the SI-appropriate terms defined in Step 2.**
4. **The results of the above activities could be summarized into a form that could be incorporated into a report for use in licensing submittals to the NRC.**

3.5 Application of 0.1g Minimum Spectrum

Section IV(a)(1)(i) of Appendix S to 10 CFR 50, states:

> *"The Safe Shutdown Earthquake Ground Motion must be characterized by free-field ground motion response spectra at the free ground surface. In view of the limited data available on vibratory ground motions of strong earthquakes, it usually will be appropriate that the design response spectra be smoothed spectra. The horizontal component of the Safe Shutdown Earthquake Ground Motion in the free-field at the foundation level of the structures must be an appropriate response spectrum with a peak ground acceleration of at least 0.1g."*

This requirement is addressed in the SRP, Section 3.7.1, and DC/COL-ISG-017. The acceptance criteria in the SRP, Section 3.7.1, specify the following:

> *"The response spectrum associated with this minimum PGA should be a smooth broad-band response spectrum (e.g., RG 1.60 or other appropriately shaped spectrum if justified) considered as an outcrop response spectrum at the free-field foundation level."*

ISG documents are intended to be temporary documents that provide guidance expected to be incorporated into the next revision of the appropriate RG. As such, they are generally considered to have the standing of a RG. Section 5.3 of the ISG provides guidance on a procedure intended to allow license applicants to perform the necessary checks to meet the minimum seismic input requirement at the foundation. The ISG DC/COL-ISG-017provides the following method:

> *"The minimum foundation input ground motion for design shall be at least an appropriate broad band response spectrum with value 0.1g peak ground acceleration (PGA). For all certified designs typically designed to a broad-band spectrum scaled to 0.30g, this check is not necessary. For the site-specific SSI analysis of Category I structures not covered by the certified design, the minimum requirement must be satisfied. For this evaluation, the FIRS used as the basis to develop the SSI input must be compared with the minimum motion. If the FIRS do not envelop the minimum spectra, the envelope of the FIRS and the minimum spectra should be used to develop the SSI input motion. Alternatively, the SSI analysis should be performed twice using the site-specific FIRS and the minimum spectra separately and enveloping the SSI responses. The three SSI soil profiles obtained from the generation of FIRS can be used for the SSI analysis. Consistent with Regulatory Position 5.2, "Vertical Spectrum," of the RG 1.208 and common industry practice, the vertical spectrum should be developed using the appropriate V/H ratios. For vertical analysis, site-specific FIRS can be used."*

The guidance provided in the SRP and ISG DC/COL-ISG-017 was developed for conventionally-founded NPPs for which the free field response can be tied directly to the motions going into the superstructure and the safety-related equipment. There is ambiguity as to where the minimum motions should be applied in a base-isolated NPP (i.e., at the foundation of the structure or at the upper basemat). On one hand, the language in 10 CFR 50 states that the minimum spectrum is developed using the SSE ground motions modified to bring the motions to the appropriate foundation horizon within the soil column. This would imply that the minimum motions would be applied at the foundation level. However, as stated in 10 CFR 50, the intention of the minimum spectrum is to account for uncertainty in the ground motion, which is important for ensuring adequate margin in the design of the safety-related SSCs. Therefore, it is reasonable to assume that the intent of the minimum spectrum was to ensure that all

safety-related equipment would be capable of remaining functional at the ISRS motions consistent with some minimum input motion.

For certified designs with an SI-CSDRS greater than the minimum spectrum, the issue is limited to the design of any safety-related equipment in the non-certified portion of the NPP. The draft SI NUREG report addresses the question of the minimum spectrum by incorporating it into the ground motions by specifying the use of a response spectrum called the GMRS+, which is the envelope of the GMRS and the minimum spectrum where the appropriate spectral shape is defined by RG 1.60, consistent with the guidance in the SRP. However, this may not meet the full intent of the CFR.

If the minimum spectrum is applied at the upper basemat, and even more so if the foundation is at multiple levels, it is unclear what the appropriate spectral shape would be as a result of the modification of input motions by the isolation system. The RG 1.60 spectra are not specified in the CFR, but are noted in current guidance as representing appropriate spectral shapes for meeting the code requirements in NRC regulatory guidance. RG 1.60 is a legacy document that is used for this purpose as a result of the absence of available alternatives. In 2008, the NRC initiated a project called Next Generation Attenuation Relationships for Central & Eastern North-America (commonly known as NGA-East), in part, for the purpose of developing appropriate minimum spectra for central and eastern North America. It is understood that the spectral shape in RG1.60, which is based on ground motions from western North America, may not be completely appropriate for the Central and Eastern U.S. As noted in the SRP, Section 3.7.1, a spectrum with the necessary qualities (e.g., a smooth broad-band spectral shape) can be used if appropriate justified. If further work on isolated NPPs places the minimum spectrum at the basemat, an associated “appropriate” spectral shape would need to be developed.

It is anticipated that the NRC will revisit the question of the appropriate spectral shape for meeting this minimum ground motion requirement for conventionally-founded structures, once the results of the NGA-East project are available.

4. CONFIGURATIONS AND APPLICATIONS NOT ADDRESSED IN FORTHCOMING GUIDANCE

4.1 Use of Isolators Exposed to Radiation and Other Uncommon Environmental Conditions

The draft SI NUREG report and the mandatory language for SI in ASCE/SEI 4-16 is written for reactor design in which the isolators are protected from exposure to both high-radiation dosage and extreme temperatures. This constrains the possible applications and benefits of utilizing SI technology.

The change in mechanical properties of elastomeric and sliding isolators installed where radiation significantly exceeds that occurring naturally in the environment has not been established by either first-principles chemistry calculations or numerical and physical experiments. Such calculations and experiments on a family of elastomers and composites would be required to support the application of SI in a high radiation environment. Cladding of contemporary isolators with perimeter radiation-hardened materials (i.e., similar to cover rubber used on elastomeric isolators to provide protection from environmental effects) may provide a viable alternative worthy of calculations and experiments.

The mechanical properties of elastomeric and sliding isolation bearings are generally selected and quantified by testing for ambient conditions of pressure and temperature at sea level. Isolators expected to operate at elevated (or significantly lower) temperatures for decades will require the development of specially compounded elastomers and composites, supported by first-principles calculations and numerical and physical experiments. Similar activities related to humidity may benefit some applications.

As discussed later in this report, an isolated NPP should be designed such that the isolation system is not exposed to standing water. This has led to questions related to the possible impact of flooding (e.g., in a design basis flood event) on the isolation system. Although contact with water has not been well explored using the types of techniques noted for elevated temperature, above, and the design should prevent standing water at the location of the isolators, it is not expected that a short-term flood event will require replacement of the isolators. However, a series of visual inspections after the event and thereafter on a frequent basis should be undertaken.

The operational environment of sodium fast reactors may present unique challenges for seismic isolators, depending on their proximity to metallic sodium and its reaction byproducts, and their exposure, if any, to chemical fires that may accompany the use of sodium. The compatibility of elastomeric or sliding isolators with sodium fast reactor technology, depending on the reactor vessel enclosure type and geometry, may require significant technology research and development.

4.2 Isolation of Components, Equipment, and Distributed Systems

Opportunities exist for the isolation of equipment and distributed systems, ranging in size from freestanding gloveboxes weighing tens of kilograms to combinations of reactor vessels and steam generators weighing hundreds of tons. These alternative isolation configurations may prove highly beneficial in addressing technical challenges that arise in various advance reactor designs. For the larger systems to be isolated, currently available 2D isolation systems may be sufficient. For equipment and systems with geometries and mass distributions that are fundamentally different from those of a traditional nuclear facility, alternate isolation system designs may be required.

Gloveboxes are used for processing and handling radioactive products. These gloveboxes are both tall and light, which makes isolation somewhat challenging. An isolation system for either a single glovebox or a system of gloveboxes will not employ traditional 2D isolators, and a 3D isolation system capable of resisting relatively significant tensile forces due to overturning effects will be needed. Although trial solutions have been proposed, analytical, numerical and physical (earthquake simulator) studies will be required to characterize and qualify these isolation systems.

5. SEISMIC QUALIFICATION OF ISOLATOR UNITS

5.1 Overview of Proposed Approach to Seismic Qualification

One identified challenge related to the licensing of isolated NPPs involves identifying and justifying the most appropriate and achievable approach to seismic qualification of the isolators. The draft SI NUREG report, Section 9.2.1, provides the following discussion:

> *"Quality control and quality assurance procedures for the testing and construction of the isolator units and seismic isolation system should follow ANSI/ASME NQA-1-2015 Quality assurance requirements for nuclear facility applications (ASME, 2015) or approved equivalent. While the isolation system as a whole acts as a structural element, individual isolators can be considered to be mechanical components that are subject to the requirements of 10 CFR 50, Appendix B and NRC RG 1.28 (NRC 2010[11]).*"

Recent work in industry has determined that the commercial grade dedication (CGD) process, described in EPRI (2014), is an appropriate equivalent approach that is well suited to the qualification of isolator units. There are two general steps that comprise the CGD approach:

1. Identify important parameters and define the acceptable performance criteria as it relates to ensuring the safety function can be performed (i.e., identify the "critical criteria" defined in the approach).
2. Identify and define in detail the appropriate method for demonstrating that the equipment meets the critical criteria defined in Step 1.

Within the CGD approach, there are different methods that be used for demonstrating that the equipment meets the critical criteria as part of Step 2. EPRI (2014) describes a method (Method 1, which is noted as "Special Tests and Inspections") that is appropriate for the isolator units and ties directly to the required prototype and production testing that is described in the draft SI NUREG report and ASCE/SEI 4-16. This required testing is summarized in the section below.

5.2 Testing Requirements

Both the draft SI NUREG report and ASCE/SEI 4-16 require a testing program with three stages that play important roles in the design, construction and operation of an isolated facility. The testing series performed in the design and construction stages are central to the design and quality assurance of the isolators.[12]

The three elements of an isolator-testing program are:

1. **Prototype testing**. Prototype testing is performed in the design phase to ensure that the targeted mechanical properties of the isolator units are achieved. These tests are performed on a small number of bearings of each type and size incorporated into the design. Dynamic tests, mimicking the environment and loading the isolators will experience under design and BDB shaking levels, are used to characterize behavior. Axial forces and lateral displacements associated with the maximum possible movement of the isolators (as permitted by the stop) are imposed to provide the requisite confidence in the mechanical behavior of the bearings.

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[11] NRC (2010) is listed as NRC (2010b) in Section 8 of this report.

[12] Although the testing program is incorporated into this section of this report due to its fundamental role in qualification, the program is also associated with a number of regulatory elements and challenges discussed elsewhere in this report.

2. **Production testing**. Production testing is performed during the construction process to provide very high confidence in the mechanical properties of the isolators in design basis shaking. Each isolator delivered to the construction site is tested. Static or dynamic tests are permitted to characterize behavior.
3. **In-service testing**. In-service testing is performed periodically during the operation of the NPP to ensure the mechanical properties of the isolators remain within the design assumptions. See also Subsection 6.2.

According to the draft SI NUREG report:

> *"A minimum of three (3) models of each prototype isolator proposed for use in the plant must be tested using full-scale loads and seismic displacements in order to assess/verify the dynamic properties of the bearings."*
>
> *"Prototype tests should be performed individually on full size specimens of each predominant type and size of isolator unit in the isolation system. Test results of full size specimens should be included in applications. Prototype tests should be dynamic and to displacements equal to or greater than the clearance to the stop. Prototype test specimens may not be used in construction. A sufficient number of prototype isolators should be tested to provide 90% confidence in the performance of the isolators. The rate of testing should be appropriate for bearings that are sensitive to the rate of loading."*
>
> *"Production (construction quality control) tests should be performed on each isolator unit to be installed in the NPP as part of the isolation system. Each isolator should be tested to the mean GMRS+ displacements. Damaged isolators should be rejected and cannot be used for construction."*

Isolator testing programs have been written around the prototype and production testing (Testing types 1 and 2) above for building, bridge and infrastructure applications of SI. The requirement for in-service testing (Testing type 3) has been added to provide very high confidence in the mechanical properties of isolators installed in a nuclear facility. One topic requiring study is how to cost-effectively achieve the required 90% confidence in the results of the prototype tests.

5.3 Commercial Grade Dedication Implementation and Documentation Requirements

The production and prototype testing described above form an appropriate basis for the Special Tests and Inspections (Method 1) applied in the CGD process if appropriate processes, controls and documentation are developed. In order to use the testing program with the CGD process, appropriate quality control and quality assurance programs must be employed for all testing programs. The documentation for these programs should identify important parameters and define acceptable performance as it relates to ensuring the safety function (critical criteria) to meet Step 1 in Section 5.1, above. The documentation should also identify and define in detail the appropriate protocols for demonstrating that the equipment meets the critical criteria to meet Step 2, in Section 5.1above. The protocols should be aligned with Method 1 in EPRI (2014). Although not required for the CGD process, the peer review activities required in the draft SI NUREG report for the SI design could provide additional assurance for the CGD process.

Certain fabrication records should also be maintained by the isolator vendor to ensure that critical characteristics of the design specification have been met. The production and testing of each isolation unit should be tracked through the use of individual serial numbers. These records should allow for complete backward traceability of materials, including metals, (shim plates and end plates), raw rubber, rubber additives, lead cores and any other materials. Fabrication and testing must also be traceable. Applicable records identified to date for the seismic isolators include (1) Certified Material Test Reports for the base materials, and (2) documentation of witnessed fabrication and testing. Other applicable records may be identified as the CGD process is applied.

6. LICENSING COMMITMENTS FOR CONSTRUCTION AND OPERATIONS

6.1 Considerations for Inspection, Testing, Analysis, and Acceptance Criteria

In INL report INL/EXT-14-33234 (Coleman and Sabharwall, 2014), appropriate development of inspection, testing, analysis, and acceptance criteria (ITAAC) is noted as a regulatory consideration (and potential issue) for facilities using SI technology. The ITAAC are detailed provisions to a combined license issued by the NRC.[13] A provision to the license is a requirement that must be satisfied before the license is implemented in full. Successful completion of all ITAAC is intended to demonstrate that plant is constructed in accordance with the governing regulations and the plant's license.

The ITAAC are developed by the licensee and submitted to the NRC. Once the NRC approves the ITAAC, the ITAAC are incorporated into the license by the NRC and ITAAC-related items are inspected during construction of the NPP. As noted in Coleman and Sabharwall (2014), the SI designer will need to consider how they will demonstrate via the ITAAC that the SI system was constructed/installed as designed and will operate as designed. The designer and licensee will need to recognize early on what the basis of their ITAAC is to meet the acceptance criteria once installed.

Each criterion in the ITAAC has three elements:

1. **Design commitments**. Key features from the design basis that must be addressed in construction.
2. **Inspections, tests, and analyses**. A description of the observations, tests, or examinations that will be performed to demonstrate that the commitment is met.
3. **Acceptance criteria**. A description of the acceptance criteria that is defined based on assumptions in the safety analysis report.

The draft SI NUREG report addresses a number of elements that should be considered for inclusion in the ITAAC. These SI-related considerations for the ITAAC, include the following items:

1. **Isolator testing program**. This program forms the basis for confirming design assumptions and for qualifying the isolator units.
2. **Additional seismic monitoring equipment**. Requirements for additional seismic monitoring instrumentation for isolated NPPs (beyond the requirements for conventionally-founded NPPs) are discussed in the draft SI NUREG report in Section 9.3.2. In an isolated facility, accelerometers are also needed on the lower foundation and the top of basemat, each in multiple locations. Additional locations in the structure for which ISRS are calculated are also recommended. In case of a seismic event, the seismic monitoring equipment would provide valuable information on the dynamic response of the isolation system and superstructure that can be compared with expected behavior and used to assess (and further validate) the SSI models. Costs for the additional static monitoring equipment is minimal if considered during the design phase.

[13] A combined license is the license the NRC issues that allows the licensee to construct and operate the plant.

3. **Tilt meters**. Tilt meters should be installed on the lower foundation, the upper basemat, and the moat walls. Tolerances for tilt of the lower foundation, the upper basemat, and the moat walls, as well as tolerances for the relative tilt between these elements, should be defined in the design process. The tilt of the moat walls must be also be monitored to ensure that the clearance to the physical stop is maintained and the walls are performing as expected. An alternative to the installation of tilt meters is the regular, periodic surveying of these civil engineering constructions, which may be preferable if the cost of installing, maintaining and monitoring the tilt meters is significant.
4. **Umbilical lines and other crossover SSCs**. Some safety-related SSCs will cross from the isolated to the non-isolated portions of the nuclear facility (i.e., the SSCs will cross the "isolation interface"). These SSCs have special performance-based design requirements as described in the draft SI NUREG report and ASCE/SEI 4-16. Special testing or inspections may be required, depending on the design of the facility.
5. **Additional performance-based criteria for deformation of the foundation and basemat**. The basemat and foundation must be designed to have adequate stiffness in both the horizontal and vertical directions to engage all of the seismic isolators in gravity and lateral-load resistance. The analysis and design of the individual isolators and the isolation system should address the short-term and long-term effects of differential settlement of the soil and of foundation flexibility. Determination of loads on the isolators and comparison with design assumptions should be considered if isolators need to be replaced. If the loads are not within the range assumed for design, additional analyses may be needed to demonstrate that the system will perform acceptably. Elements that are required to meet operational needs, such as physical access to the isolation layer, strong points for jacking, and ability of the basemat to span over one (or more) "lost" isolator units should be incorporated into the design.

Because an NPP utilizing SI has not been licensed in the U.S. to date, the ITAAC for such a facility has not been developed. This section provides information on elements that have been identified for inclusion into the ITAAC. However, the necessary licensing submittal would require a far more technical detail than is provided above. Although some of the information in the ITAAC will be facility-specific, a significant portion of the necessary documentation could be developed in the short-term using available information and focusing on the three elements of the ITAAC, as noted above (i.e., design commitments; inspections, tests and analyses; and acceptance criteria).

6.2 In-Service Inspections, Testing, and Operations

The in-service inspection and testing program must be defined as part of the license application. The program should provide complete information on the method of inspection and testing, the acceptance criteria, and the steps to be taken if the inspection or testing indicates that the isolators or isolation system does not continue to meet the requirements.

1. **In-service testing.** The program should include periodic verification of mechanical properties of isolator devices to assess the effect of ageing on their performance. To achieve this, a minimum of two spare isolators of each type used are required to be loaded in a manner similar to the isolators in service (say using flatjacks and a self-equilibrating fixture) in the same environment as the in-service isolators. One should be removed at each 10-year interval and subjected to the initial testing protocol used for design qualification. The use of spares will permit their removal for testing as per the inspection protocols and would not require the plant to shut down while the testing is being performed. Spare units located in an environment with radiation could be reinstalled during the next service period. Isolation of the plant is not compromised during this period. Section 9.1.1 in the draft SI NUREG report describes the approach used for assessing the acceptability of changes in dynamic properties observed during the testing. Generally, in U.S. practice, the mechanical properties of the isolation system (i.e., the force-displacement relationship) should not vary over the lifespan of the structure by more than ±20% from the best-estimate values (with 95% probability) from those assumed for analysis and design. The range of values used in design and expected over the life of the

facility should be clearly specified and the in-service inspection program framed accordingly. If the change in isolator properties exceeds the range used in design, the system should be reassessed and isolators may need to be replaced if sufficient safety margin cannot be demonstrated.

2. **In-service visual inspection.** An appropriate inspection plan that ensures that performance criteria consistent with the NPP design and operational assumptions are met must be developed for the isolator units. This plan must include both criteria and methods to be used for in-service inspection. The required periodic visual inspections for both rubber bearings and FP bearings are relatively straightforward. In the case of rubber bearings, the bearings are confined within a permanent, non-removable, protective rubber envelope intended to preclude environmental degradation of the bearing, so long as the recommended/design environmental conditions are maintained. The isolator units should be visually inspected for signs of delamination or debonding. FP bearings should be inspected for signs of wear and degradation. A camera may be used to access the interior of the FP bearing to ensure that no standing water is present. The anchorage to the pedestals (including the anchor bolts), basemat, and moat walls should be visually inspected for signs of deterioration and stress. The fire suppression system should be inspected. The area around the isolators should be inspected to ensure that the area is clear of debris and water. Visual inspections should be conducted at least annually.

3. **In-service monitoring of the foundation and moat.** Monitoring of foundation deformations during operation of the NPP is recommended in the draft SI NUREG report in Section 9.3.3 as a result of experience in the Cruas-Meysse NPP in France. The deformations to be monitored include movement of the moat walls, relative displacements between the basemat and foundation, and vertical displacements of the basemat and foundations.

4. **Post-earthquake inspection.** A post-earthquake inspection plan for the isolation system should be developed and included in the licensing documentation to prevent unnecessary delays in restarting the facility after an event. The plan should, as a minimum, address cases where seismic motions exceeding the Operating Basis Earthquake ground motion and/or SSE ground motion were recorded. The plan should include visual inspection of the isolator units, the pedestal attachments, the foundation and basemat, the fire suppression system, the umbilical systems, the moat cover, and the moat wall. Data recorded by seismic monitoring equipment could be used to analyze the loading levels imposed on the isolation system and the dynamic response of the SI system and superstructure, as well as ISRS associated with risk-sensitive equipment. Representative isolators may be removed from service (and temporarily replaced) and tested post SSE shaking. Having an agreed upon post-earthquake inspection protocol in plant documentation may save significant time (including staff time), reduce regulatory risk, and result in a significantly earlier restart in cases where seismic loading exceeds the OBE or SSE ground motions. Therefore, it may be beneficial to develop this inspection plan as part of a broader post-earthquake inspection plan that could be informed by an SPRA conducted for the facility.

5. **Isolator unit removal and replacement requirements.** Over the life of the facility, it may be necessary to remove and reinstall or replace individual isolator units. The performance criteria for the isolator bearings and foundation system must include the requirement that a bearing be replaceable if necessary, although the bearings are designed to remain in service throughout the life of the plant. Removal and replacement of the isolator units is typically achieved through the use of a flat jack system. Elements that allow for removal and replacement of the isolators (e.g., room to transport the necessary equipment) must be incorporated into the facility design. Protocols for removing and replacing isolator units must be provided in the license application.

Although many items related to the and in-service inspections, in-service testing, and operational considerations have been identified and discussed, the complete treatment of these elements in the regulatory process (and a proposed best practice approach) have not been fully investigated and

documented. An effort focused in this area could be conducted in the short-term and would not be expected to involve a significant level of additional research.

6.3 Technical Specifications and Descriptions of Seismic Isolation Equipment for Licensing Documentation

The mechanical and dynamic properties of the isolation system are a function of both the mechanical properties of the individual isolators that comprise the system and the plan distribution of the isolators in the system. The isolator properties are site-specific and evolve over the course of the design as gravity loadings and resultant earthquake loadings are refined. The mechanical properties of the isolators will be developed from analysis, calculations, specifications, relevant engineering tests, and standards (e.g., ASCE/SEI 4-16) and guidelines (e.g., draft SI NUREG report). Because appropriate design of both the individual isolators and the complete isolation system and are dependent on the site dynamic properties and its seismic hazard, the mechanical properties are defined on a site-specific basis.

Mechanical and dynamic properties (and geometric characteristics) for the seismic isolators provided in the licensing documentation should include the following information for low damping rubber and lead rubber isolators:

- Dimensions and baseplate attachment details (to basemat above and pedestal below), including end plates, flange plates and bolts
- Mechanical properties and relevant American Society for Testing and Materials International standards for all materials, including end plates, flange plates, shim plates and bolts
- Surface treatment and adhesives used for shim plates
- Cover rubber layer thickness
- Isolator diameter and height
- Lead plug diameter, if any
- Dynamic properties of the elastomer (e.g., shear stiffness as a function of shear strain)
- Dynamic and static properties of the lead (in the plug, if any)
- Vertical stiffness in tension and compression
- Axial compressive capacity at zero lateral displacement
- Compressive capacity as a function of horizontal displacement
- Tensile capacity as a function of horizontal displacement
- Lateral force-lateral displacement hysteresis loops from dynamic and static loading.

Mechanical and dynamic properties (and geometric characteristics) for the seismic isolators provided in the licensing documentation should include the following information for FP isolators:

- Dimensions and baseplate attachment details (to basemat above and pedestal below), including bolts
- Mechanical properties and relevant American Society for Testing and Materials International standards for all materials, including housing plates, articulated sliders, and inlays for the sliding surfaces
- Surface treatments and adhesives used for composites
- Protective coating type and thickness (to prevent water and debris ingress to the interior of the isolator)
- Radius of the sliding surfaces and geometries of all articulated sliders

- Dynamic properties of the composites used (e.g., friction coefficient as a function of pressure and velocity)
- Vertical stiffness in compression
- Axial compressive capacity at zero lateral displacement
- Compressive capacity as a function of horizontal displacement
- Lateral force-lateral displacement hysteresis loops from dynamic and static loading.

7. IDENTIFIED ISSUES AND PROPOSED RESOLUTIONS

This report identifies and discusses regulatory gaps and challenges for the licensing of NPPs using SI technologies, including isolated advanced reactor designs. Table 4 summarizes the gaps and challenges that were identified in prior sections of this report, and the identified approaches for addressing each topic. The "Importance" indicator in the table relates to the degree to which the issue may impact ability to license advanced reactors with seismic isolation in a timely manner; the designators are self-explanatory. Table 4 includes information on whether the timeframe to complete the activities described in the path forward is a short-, medium-, or long-term effort, where short term is 1–3 years, medium term is 3–5 years, and long term is 5–10 years. The issues that have been identified as the most high-impact/high-value topics in terms of their degree of support for licensing activities are highlighted orange in Table 4.

Table 4. Summary of identified issues, importance, and proposed resolutions.

Issue	Section	Importance	Time-frame	Applies to	Proposed Actions and Path Forward
Orange highlighted activities are those identified as addressing the most high-impact/high-value topics.					
Lack of verified and validated tools for nonlinear SSI	2.1.3	High	Medium	LWRs and Advanced Reactors	Develop 1D, 2D, and 3D nonlinear constitutive models consistent with typical soil profiles at the sites of nuclear facilities in the U.S. Identify, catalog, and document existing soil models.
		High	Medium	LWRs and Advanced Reactors	Verify 1D, 2D, and 3D nonlinear constitutive models. Develop, solve, and document sample problems.
		High	Short	LWRs and Advanced Reactors	Plan and execute 1D, 2D, and 3D static and dynamic element-level tests on soil samples.
		High	Medium	LWRs and Advanced Reactors	Validate 1D, 2D, and 3D nonlinear constitutive models using experimental data.
		High	Medium	LWRs and Advanced Reactors	Verify accuracy of computer codes for SRA and SSI analysis.
		High	Short	LWRs and Advanced Reactors	Plan and execute tests in 1D geotechnical laminar box.
		High	Medium	LWRs and Advanced Reactors	Validate accuracy of computer codes for SRA and SSI analysis.
		High	Short	LWRs and Advanced Reactors	Plan and execute gapping and sliding tests on an earthquake simulator.
		High	Medium	LWRs and Advanced Reactors	Validate accuracy of computer codes for gapping and sliding.
Lack of technical bases or guidance	2.1.4	High	Medium	Advanced Reactors	Using new capabilities recently added to MOOSE, run case studies to understand impact of inclined waves to a variety of

Issue	Section	Importance	Time-frame	Applies to	Proposed Actions and Path Forward
Orange highlighted activities are those identified as addressing the most high-impact/high-value topics.					
related to cases where assumption of 1D vertically propagating shear waves does or does not apply					reactor structural configurations. Once it is determined what wave forms are significant, the wave forms can be tied back to the sources that cause them (e.g., steeply dipping bedrock or soil layers, significant surface waves). This effort is directly tied to GMSM.
Lack of technical basis or guidance for GMSM	2.1.5	High	Short	LWRs and Advanced Reactors	Identify optimal spectral representation for SSI analysis and how to scale earthquake ground motions to that representation.
		High	Short	LWRs and Advanced Reactors	Determine minimum number of sets of ground motions for design and SPRA.
		Medium	Medium	LWRs and Advanced Reactors	Prepare guidance for an upcoming revision of ASCE/SEI 4.
Lack of guidance for design and analysis considerations for the stop	2.2	Medium	Medium	LWRs and Advanced Reactors	Develop trial designs for stops. Analyze and design stops for a sample surface-founded LWR, isolated reactor vessel in deeply-embedded advanced reactor, and surface-founded diesel generator.
	2.2	Low	Long	LWRs and Advanced Reactors	Identify construction materials suitable for use as a soft stop. Identify implications of implementing a soft stop. Extend trial designs for stops (above) to include soft stops.

Issue	Section	Importance	Time-frame	Applies to	Proposed Actions and Path Forward
Orange highlighted activities are those identified as addressing the most high-impact/high-value topics.					
No robust SPRA calculation procedure for deeply-embedded advanced reactors	2.3.2	High/ Medium	Medium	Advanced Reactors	Develop trial designs of deeply-embedded advanced reactors to enable SPRA. Sample designs or test beds are needed for the civil/structural system and safety-related SSCs. Designs would be accompanied by fragility functions for chosen SSCs. Sites and soil profiles consistent with possible DOE construction of advanced reactors, including INL should be selected to maximize the benefit of test bed designs.
	2.3.2	Medium	Medium	Advanced Reactors	Develop accident sequences for sample reactor designs. Event trees and fault trees will be needed to develop systems models for the sample reactor and enable SPRA.
	2.3.2	Medium	Medium	Advanced Reactors	Perform SPRA calculations for sample reactors. The analysis, design, and risk assessment sample reactors (one reactor type, multiple sites, multiple soil profiles), coupled with careful documentation thereof, would lay bare the process to analysts, designers, and regulators.
No robust SPRA calculation procedure for deeply-embedded advanced reactors incorporating isolated components	2.3.3	Medium	Medium	Advanced Reactors	Develop trial designs of protective systems for components. Selected safety-related SSCs would be identified for possible isolation. Seismic design *spaces* would be developed for two cases for each SSC: capacity based on non-seismic load cases, and capacity based on the designs of Subsection 2.3.2. Protective systems (isolators and dampers) for each case, site and soil profile would be developed. All designs would be priced to establish the benefit, if any, of the use of protective systems.
	2.3.3	Medium	Medium	Advanced Reactors	Develop accident sequences for the protective systems and safety-related umbilical lines. Event trees and fault trees will be needed to augment the systems models of Subsection 2.3.2.
	2.3.3	Medium	Medium	Advanced Reactors	Perform SPRA calculations for the sample reactors. The analysis, design, and risk assessment sample reactors (one reactor type, multiple sites, multiple soil profiles), coupled with the careful documentation thereof, would clarify the process to analysts, designers, and regulators. Importantly, SPRA calculations would guide possible use of protective systems in deeply-embedded reactors.
	2.3.3	High	Medium	LWRs and	Develop isolation systems for large

Issue	Section	Importance	Time-frame	Applies to	Proposed Actions and Path Forward
Orange highlighted activities are those identified as addressing the most high-impact/high-value topics.					
				Advanced Reactors	components. Prepare guidance for inclusion in an upcoming revision of ASCE/SEI 4.
Implementation of advanced human factors approaches in SPRA	2.3.4	Medium	Medium to Long	LWRs and Advanced Reactors	Characterize the benefit of lesser shaking demands and building damage on the reliability of operator response and risk reduction.
Development of guidance for SPRA feedback to design	2.3.5	Medium	Medium	LWRs and Advanced Reactors	Develop guidance for design process that incorporates risk analysis as a fundamental element of the design process.
Assessment of applicability of screening tables in isolated facilities due to V/H changes	2.3.6	Low	Medium	LWRs and Advanced Reactors	Review information on development of screening tables and review potential failure mechanisms to determine sensitivity to vertical motions.
Development of updated approach and new terminology to clarify certified design approach applied to isolated NPPs	3.1, 3.2	High	Short	Advanced Reactors	Evaluate best approaches and new terminology to close the regulatory gap related to licensing of advanced reactors using the certified design process. Analyze case studies to understand challenges. Draft report to the NRC.
Identification of potential regulatory issues due to use of legacy terminology	3.4	Medium	Short	LWRs and Advanced Reactors	Identify where legacy words are used. Determine the function of each word and map to improved terminology.
Clarification of application of minimum spectrum	3.5	Low	Medium	LWRs and Advanced Reactors	Work with NRC to clarify expectations in terms of deeply-embedded advanced reactors, surface-founded base-isolated reactors, and isolated components in reactor buildings.
Use of isolators in new environmental and radiological conditions	4.1	Low	Medium to Long	LWRs and Advanced Reactors	Activity on this task will require trial design of advanced reactors or new applications of isolators to large LWRs. The importance of this activity is a function of the perceived need. Tests to characterize the effect of long-term radiation exposure on isolators may take years.
Developing new isolators to protect components,	4.2	Medium	Short, Medium and Long	LWRs and Advanced Reactors	Activity on this task will depend on need. Development of isolators and isolation systems similar in principal to those in service is a short-to-medium term activity.

Issue	Section	Importance	Time-frame	Applies to	Proposed Actions and Path Forward
Orange highlighted activities are those identified as addressing the most high-impact/high-value topics.					
equipment, and distributed systems					Development of a new class of isolators may take a decade.
Seismic qualification of isolator units	5	Low	Short	LWRs and Advanced Reactors	This approach should be documented more thoroughly and presented to the NRC for consideration and comment.
Documentation for ITAAC for isolation technology	6.1	Low	Short	LWRs and Advanced Reactors	Although some of the information in the ITAAC will be facility-specific, a significant portion of the necessary documentation could be developed using available information and focusing on the three elements of the ITAAC: design commitment; inspections, tests, and analyses; and acceptance criteria.
In-service testing and operations requirements and documentation	6.2	Low	Short	LWRs and Advanced Reactors	Although many items related to the in-service inspections and testing and operational considerations have been identified and discussed, the complete treatment of these elements in the regulatory process (and a proposed best practice approach) have not been fully investigated and documented. An effort focused in this area could be conducted in the short-term and would not be expected to involve a significant level of effort.

The issues and proposed actions identified in Table 4 are focused primarily on the needs of advanced (non-LWR) reactor technologies. However, some of the issues and recommended path forward actions are also applicable to LWRs. The proposed actions and path forward identified in Table 4 are identified at a high level and merit further development and refinement. A detailed "roadmap", developed in collaboration with a commercial advanced reactor partner, would be a logical "next step" in this development. This activity will be addressed in the next phase of this project. The roadmap generated by this forthcoming effort will offer an efficient vehicle for laying-out out detailed plans for closing technical and regulatory gaps for generic advanced reactor technologies.

8. REFERENCES

10 CFR 50, "Domestic Licensing of Production and Utilization Facilities."

10 CFR 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

ANSYS Inc., 2013. "Computer program ANSYS 13.0," Canonsburg, PA.

ASCE, 2005. ASCE/SEI 43-05, "Seismic design criteria for structures, systems and components in nuclear facilities," Structural Engineering Institute of the American Society of Civil Engineers, Reston, VA.

ASCE, forthcoming. ASCE/SEI 4-16, "Seismic analysis of safety-related nuclear structures," Structural Engineering Institute of the American Society of Civil Engineers, Reston, VA.

ASME, 2006. ASME VV-10, "Verification and validation in computational solid mechanics," American Society of Mechanical Engineers, New York, NY.

ASME, 2015. NQA-1-2015, "Quality assurance requirements for nuclear facility applications," American Society of Mechanical Engineers, New York, NY.

Baker, J. W., and C. A. Cornell, 2006. "Spectral shape, epsilon and record selection," *Earthquake Engineering and Structural Dynamics*, Vol. 35, No. 9, pp. 1077–1095.

Bielak, J., 1975. "Dynamic behavior of structures with embedded foundations," *Earthquake Engineering and Structural Dynamics*, Vol. 3, pp. 259–274.

Bielak, J., and P. Christiano, 1984. "On the effective seismic input for nonlinear soil-structure interaction systems," *Earthquake Engineering and Structural Dynamics*, Vol. 12, No. 1, pp. 107–119.

Bielak, J., K. Loukakis, Y. Hisada, and C. Yoshimura, 2003. "Domain reduction method for three-dimensional earthquake modeling in localized regions, Part-I: theory," *Bulletin of the Seismological Society of America*, Vol. 93, No. 2, pp. 817–824.

BNL, 2009. *Consistent site response—SSI calculations*, Report N6112-051208, Rev. 1, Brookhaven National Laboratory, Stonybrook, NY (ADAMS Accession No. ML091980384).

Bolisetti, C., and A. S. Whittaker, 2015. "Site response, soil-structure interaction, and structure-soil-structure interaction for performance assessment of buildings and nuclear structures," *Technical Report MCEER-15-0002*, University at Buffalo, State University of New York, Buffalo, NY.

Bolisetti, C., A. S. Whittaker, and J. Coleman, 2016. "Linear and nonlinear soil-structure-interaction analysis," submitted to *Soil Dynamics and Earthquake Engineering*.

Bolisetti, C., A. S. Whittaker, H. B. Mason, I. Almufti, and M. Willford, 2014. "Equivalent linear and nonlinear site-response analysis for design and risk assessment of safety-related nuclear structures," *Nuclear Engineering and Design*, Vol. 275, pp. 107–121, http://dx.doi.org/10.1016/j.nucengdes.2014.04.033.

Bolisetti, C., J. L. Coleman, M. Talaat, and P. Hashimoto, 2015. *Advanced seismic fragility modeling using nonlinear soil-structure interaction analysis*, INL/EXT-15-36735, Idaho National Laboratory, Idaho Falls, ID.

Coleman, J., and P. Sabharwall, 2014. *Seismic isolation working meeting gap analysis report development*, INL/EXT-14-33234, Idaho National Laboratory, Idaho Falls, ID.

Coleman, J., C. Bolisetti, and A.S. Whittaker, 2016. "Time-domain soil-structure interaction analysis of nuclear facilities," *Nuclear Engineering and* Design, Vol. 298, March 2016, Pages 264–270

Dassault Systèmes, 2015. "Computer program ABAQUS," Providence, RI.

DOE, 2002. DOE-STD-1020-2002, "Natural phenomena hazards design and evaluation criteria for Department of Energy facilities," United States Department of Energy, Washington, D.C.

EPRI, 1991. *A methodology for assessment of nuclear power plant seismic margin*, Report NP-6041-SLR1, Rev 1, Electric Power Research Institute, Palo Alto, CA.

EPRI, 2014. *Plant Engineering: Guideline for the acceptance of commercial-grade items in nuclear safety-related applications: Revision 1 to EPRI NP-5652 and TR-102260*, Report 3002002982, Electric Power Research Institute, Palo Alto, CA.

FEMA, 2012. *Seismic performance assessment of buildings, Volume 1—Methodology*, FEMA P-58-1, Federal Emergency Management Agency, Washington, D.C.

Huang, Y.-N., A. S. Whittaker, and N. Luco, 2008a. "Maximum spectral demands in the near-fault region," *Earthquake Spectra*, Vol. 24, No. 1, pp. 319–341.

Huang, Y.-N., A. S. Whittaker, and N. Luco, 2008b. "Performance assessment of conventional and base-isolated nuclear power plants for earthquake and blast loadings," *Technical Report MCEER-08-0019*, University at Buffalo, State University of New York, Buffalo, NY.

Huang, Y.-N., A. S. Whittaker, and N. Luco, 2009a. "Orientation of maximum spectral demand in the near-fault region," *Earthquake Spectra*, Vol. 25, No. 3, pp. 707–717.

Huang, Y.-N., A. S. Whittaker, and N. Luco. 2011a. "A seismic risk assessment procedure for nuclear power plants, (I) methodology," *Nuclear Engineering and Design*, Vol. 241, pp. 3996–4003.

Huang, Y.-N., A. S. Whittaker, and N. Luco, 2011b. "A seismic risk assessment procedure for nuclear power plants, (II) application," *Nuclear Engineering and Design*, Vol. 241, pp. 4004–4011.

Huang, Y.-N., A. S. Whittaker, R. P. Kennedy, and R. L. Mayes, 2009b. "Assessment of base-isolated nuclear structures for design and beyond-design-basis earthquake shaking," *Technical Report MCEER-09-0008*, University at Buffalo, State University of New York, Buffalo, NY.

Huang, Y.-N., A. S. Whittaker, R. P. Kennedy, and R. L. Mayes, 2013. "Response of base-isolated nuclear structures for design and beyond-design basis earthquake shaking," *Earthquake Engineering and Structural Dynamics*, Vol. 42, No. 3, pp. 339–356.

INL, 2011. *Modular HTGR safety basis and approach*, INL/EXT-11-22708, Idaho National Laboratory, Idaho Falls, ID.

Jayaram, N., T. Lin, and J. W. Baker, 2011. "A computationally efficient ground-motion selection algorithm for matching a target response spectrum mean and variance," *Earthquake Spectra*, Vol. 27, No. 3, pp. 797–815.

Jeremić, B., G. Jie, M. Preisig, and N. Tafazzoli, 2009. "Time domain simulation of soil-foundation-structure interaction in non-uniform soils," *Earthquake Engineering and Structural Dynamics*, Vol. 38, No. 5, pp. 699–718.

Jeremić, B., R. Roche-Rivera, A. M. Kammerer, N. Tafazzoli, J. Abell, M. B. Kamrani-moghaddam, F. Pisano, C. Jeong, and B. Aldridge, 2013. "The NRC ESSI simulator program," *Transactions*, 22nd International Conference in Structural Mechanics in Reactor Technology (SMiRT), San Francisco, CA.

Kammerer, A., A. S. Whittaker, and M. C. Constantinou, forthcoming. *Technical considerations for the seismic isolation of nuclear facilities, NUREG-* forthcoming, United States Nuclear Regulatory Commission, Washington, DC.

Kammerer, A., J. M. Pestana, and R. B. Seed, 2002. "Undrained response of Monterey 0/30 sand under multidirectional cyclic simple shear loading conditions," *Technical Report UCB/GT/02-01*, Department of Civil and Environmental Engineering, University of California, Berkeley, CA.

Kumar, M., and A. S. Whittaker, 2016. "Seismic probabilistic risk assessment of safety-related nuclear facilities," submitted to *Nuclear Engineering and Design*.

Kumar, M., A. S. Whittaker, and M. C. Constantinou, 2014. "An advanced numerical model of elastomeric seismic isolation bearings," *Earthquake Engineering and Structural Dynamics*, Vol. 43, No. 13, pp. 1955–1974.

Kumar, M., A. S. Whittaker, and M. C. Constantinou, 2015a. "Characterizing friction in sliding isolation bearings," *Earthquake Engineering and Structural Dynamics*, Vol. 44, No. 9, pp. 1409-1425, July 2015.

Kumar, M., A.S. Whittaker, and M.C. Constantinou, 2015b. "Seismic isolation of nuclear power plants with sliding bearings," *Technical Report MCEER-15-0006*, University at Buffalo, State University of New York, Buffalo, NY.

Kumar, M., A.S. Whittaker, and M.C. Constantinou, 2015c. "Seismic isolation of nuclear power plants with elastomeric bearings," *Technical Report MCEER-15-0008*, University at Buffalo, State University of New York, Buffalo, NY.

LSTC, 2013. "Computer program LS-DYNA," Version 7.0, Livermore Software Technology Corporation, Livermore, CA.

Luco, J. E., and L. Contesse, 1973. "Dynamic structure-soil-structure interaction," *Bulletin of the Seismological Society of America*, Vol. 63, No. 4, pp. 1289–1303.

Luco, J. E., and R. A. Westman, 1971. "Dynamic response of circular footings," *Journal of Engineering Mechanics Division*, Vol. 97, EM5, pp. 1381–1395.

Lysmer, J., and R. L. Kuhlemeyer, 1969. "Finite dynamic model for infinite media," *Journal of Engineering Mechanics Division*, Vol. 95, EM4, pp. 859-877.

Lysmer, J., F. Ostadan, and C. Chin, 1999. "Computer program SASSI2000--a system for analysis of soil-structure interaction," University of California, Berkeley, CA.

Mason, H. B., 2011. "Seismic performance assessment in dense urban environments," *Ph.D. dissertation*, University of California, Berkeley, CA.

Mathworks, 2016. "Computer program MATLAB," Natick, MA.

Mazzoni, S., F. McKenna, M. H. Scott, and G. L. Fenves, 2009. "Computer program OpenSees: Open System for Earthquake Engineering Simulation," *Pacific Earthquake Engineering Research Center*, University of California, Berkeley, CA.

NEI, 2009. "Consistent site-response/soil-structure interaction analysis and evaluation," Nuclear Energy Institute, Washington, D.C. (ADAMS Accession No. ML091680715).

NIST, 2011. "Selection and scaling of earthquake ground motions for performing response-history analysis," *Technical Report NIST GCR 11-917-15*, National Institute of Standards and Technology, Gaithersburg, MD.

NRC, 1973. Regulatory Guide 1.60, "Design response spectra for seismic design of nuclear power plants," Rev. 1, Nuclear Regulatory Commission, Washington, D.C.

NRC, 2007a. NUREG-0800, "Standard review plan for the review of safety analysis reports for nuclear power plants," Nuclear Regulatory Commission, Washington, D.C. (ADAMS Accession Nos. ML070730593, ML070640306, ML070640311).

NRC, 2007b. Regulatory Guide 1.208, "A performance-based approach to define the site-specific earthquake ground motion," Nuclear Regulatory Commission, Washington, D.C.

NRC, 2009. DC/COL-ISG-017, "Interim staff guidance on ensuring hazard-consistent seismic input for site response and soil structure interaction analyses," Nuclear Regulatory Commission, Washington, D.C.

NRC, 2010a. DC/COL-ISG-020, "Interim staff guidance on implementation of a probabilistic risk assessment-based seismic margin analysis for new reactors," Nuclear Regulatory Commission, Washington, D.C.

NRC, 2010b. Regulatory Guide 1.28, "Quality assurance program criteria (Design and construction)," Rev. 4, Nuclear Regulatory Commission, Washington, D.C.

NRC, 2012. NUREG-2117, "Practical implementation guidelines for SSHAC level 3 and 4 hazard studies," Rev. 1, Nuclear Regulatory Commission, Washington, D.C.

NRC, 2013. "Design-specific review standard for mPOWER iPWR design," Nuclear Regulatory Commission, Washington, D.C. (ADAMS Accession No. ML13099A205).

Oberkampf, W. L., and C. J. Roy, 2010. *Verification and validation in scientific computing*, Cambridge University Press.

Ostadan, F., 2006a. "SASSI2000: a system for analysis of soil-structure interaction—user's manual," University of California, Berkeley, CA.

Ostadan, F., 2006b. "SASSI2000: a system for analysis of soil-structure interaction—theoretical manual," University of California, Berkeley, CA.

Ostadan, F., and N. Deng, 2011. "Computer program SASSI2010," Version 1.1, Geotechnical and Hydraulic Engineering Services, Bechtel National Inc., San Francisco, CA.

Presley, M. R., J. A. Julius, J. F. Grobbelaar, and K. D. Kohlhepp, 2013a. "A preliminary approach to human reliability analysis for external events with a focus on seismic HRA." *Proceedings*, ANS PSA 2013 International Topical Meeting on Probabilistic Safety Assessment and Analysis, Columbia, SC.

Presley, M. R., J. A. Julius, J. F. Grobbelaar, and K. D. Kohlhepp, 2013b. "A review of seismic operating experience with implications for human reliability," *Proceedings*, ANS PSA 2013 International Topical Meeting on Probabilistic Safety Assessment and Analysis, Columbia, SC.

Ryan, H., 1994. "Ricker, Ormsby, Klander, Butterworth—a choice of wavelets," *Canadian Society of Exploration Geophysicts Recorder*, Vol. 19, No. 7, pp. 8–9.

Schnabel, P. B., J. Lysmer, and H. B. Seed, 2012. "Computer program SHAKE: a computer program for earthquake response analysis of horizontally layered sites," University of California, Berkeley, CA.

Seed, H. B., and I. M. Idriss, 1970. "Soil moduli and damping factors for dynamic response analysis," *Report UCB/EERC 70/10*, Earthquake Engineering Research Center, University of California, Berkeley, CA.

Taborda, R., and J. Bielak, 2011. "Large-scale earthquake simulation: computational seismology and complex engineering systems," *Computing in Science and Engineering*, Vol. 13, No. 4, pp. 14–27.

Veletsos, A. S., and B. Verbic, 1973. "Vibration of viscoelastic foundations," *Earthquake Engineering and Structural Dynamics*, Vol. 2, No. 1, pp. 87–105.

Veletsos, A. S., and J. W. Meek, 1974. "Dynamic behavior of building foundation systems," *Earthquake Engineering and Structural Dynamics*, Vol. 3, No. 2, pp. 121–138.

Veletsos, A. S., and Y. T. Wei, 1971. "Lateral and rocking vibration of footings," *Journal of the ASCE Soil Mechanics and Foundations Division*, Vol. 97, SM9, pp. 1227–1248.

Willford, M., R. Sturt, Y. Huang, I. Almufti, and X. Duan, 2010. "Recent advances in nonlinear soil-structure interaction analysis using LS-DYNA," *Proceedings*, NEA-SSI Workshop, Ottawa, Canada, October 6–8, 2010.

Wolf, J.P., 1985. *Dynamic Soil-Structure Interaction*, Prentice Hall, Englewood Cliffs, NJ.

Xu, J., J. Bielak, O. Ghattas, and J. Wang, 2003. "Three-dimensional nonlinear seismic ground motion modeling in basins," *Physics of the Earth and Planetary Interiors*, Vol. 137, pp. 81–95.

Yang, T.Y., J.P. Moehle, B. Stojadinovic, and A. Der Kiureghian, 2009. "Performance evaluation of structural systems: theory and implementation," *Journal of Structural Engineering*, Vol. 135, No. 10, pp. 1146–1154.

Yoshimura, C., J. Bielak, and Y. Hisada, 2003. "Domain reduction method for three-dimensional earthquake modeling in localized regions, part-II: verification and examples," *Bulletin of the Seismological Society of America*, Vol. 93, No. 2, pp. 825–840.