March 8, 2024

TP-LIC-LET-0122
Project Number 99902100

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTN: Document Control Desk

Subject: Transmittal of TerraPower, LLC Topical Report, "Reactor Seismic Isolation System Qualification Topical Report," Revision 0

This letter transmits the TerraPower, LLC (TerraPower) Topical Report "Reactor Seismic Isolation System Qualification Topical Report," Revision 0 (enclosed). The report contains an overview and description of the seismic isolation system qualification methodology for the Natrium™ Plant[1].

TerraPower requests the NRC's review and approval of the qualification methodology presented in this report for use by future applications utilizing the Natrium design.

TerraPower requests that a nominal review duration of 12 months be considered.

The report contains proprietary information and as such, it is requested that Enclosure 3 be withheld from public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." An affidavit certifying the basis for the request to withhold Enclosure 3 from public disclosure is included as Enclosure 1. Proprietary materials have been redacted from the report provided in Enclosure 2; redacted information is identified using [[ ]]$^{(a)(4)}$.

This letter and enclosures make no new or revised regulatory commitments.

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1 Natrium is a TerraPower and GE-Hitachi technology.

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If you have any questions regarding this submittal, please contact Ryan Sprengel at rsprengel@terrapower.com or (425) 324-2888.

Sincerely,

Ryan Sprengel
Director of Licensing, Natrium
TerraPower, LLC

Enclosures:
1. TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))
2. TerraPower, LLC Topical Report, "Reactor Seismic Isolation System Qualification Topical Report," Revision 0 – Non-Proprietary (Public)
3. TerraPower, LLC Topical Report, "Reactor Seismic Isolation System Qualification Topical Report," Revision 0 – Proprietary (Non-Public)

cc: Mallecia Sutton, NRC
William Jessup, NRC
Nathan Howard, DOE
Jeff Ciocco, DOE

ENCLOSURE 1

TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure (10 CFR 2.390(a)(4))

Enclosure 1
TerraPower, LLC Affidavit and Request for Withholding from Public Disclosure
(10 CFR 2.390(a)(4))

I, George Wilson, hereby state:

1. I am the Vice President, Regulatory Affairs and I have been authorized by TerraPower, LLC (TerraPower) to review information sought to be withheld from public disclosure in connection with the development, testing, licensing, and deployment of the Natrium™ reactor and its associated fuel, structures, systems, and components, and to apply for its withholding from public disclosure on behalf of TerraPower.

2. The information sought to be withheld, in its entirety, is contained in Enclosure 3, which accompanies this Affidavit.

3. I am making this request for withholding, and executing this Affidavit as required by 10 CFR 2.390(b)(1).

4. I have personal knowledge of the criteria and procedures utilized by TerraPower in designating information as a trade secret, privileged, or as confidential commercial or financial information that would be protected from public disclosure under 10 CFR 2.390(a)(4).

5. The information contained in Enclosure 3 accompanying this Affidavit contains non-public details of the TerraPower regulatory and developmental strategies intended to support NRC staff review.

6. Pursuant to 10 CFR 2.390(b)(4), the following is furnished for consideration by the Commission in determining whether the information in Enclosure 3 should be withheld:

    a. The information has been held in confidence by TerraPower.

    b. The information is of a type customarily held in confidence by TerraPower and not customarily disclosed to the public. TerraPower has a rational basis for determining the types of information that it customarily holds in confidence and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application and substance of that system constitute TerraPower policy and provide the rational basis required.

    c. The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR 2.390, it is received in confidence by the Commission.

    d. This information is not available in public sources.

    e. TerraPower asserts that public disclosure of this non-public information is likely to cause substantial harm to the competitive position of TerraPower, because it would enhance the ability of competitors to provide similar products and services by reducing their expenditure of resources using similar project methods, equipment, testing approach, contractors, or licensing approaches.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on: March 7, 2024

George Wilson

George Wilson
Vice President, Regulatory Affairs
TerraPower, LLC

ENCLOSURE 2

TerraPower, LLC Topical Report
"Reactor Seismic Isolation System Qualification Topical Report" Revision 0

Non-Proprietary (Public)

Verify Current Revision

Document Title: Reactor Seismic Isolation System Qualification Topical Report				
Natrium Document No.: NAT-8922	**Rev. No.:** 0	**Page:** 1 of 90	**Doc Type:** RPRT	**Target Quality Level:** N/A
Alternate Document No.: N/A	**Alt. Rev.:** N/A	**Originating Organization:** TerraPower, LLC. (TP)		**Quality Level:** N/A
Natrium MSL ID: RES	**Status (per NAT-1974):** Released			**Open Items?** N/A
Approval				
Approval signatures are captured and maintained electronically; see Electronic Approval Records in EDMS. Signatures or Facsimile of Electronic Approval Record attached to document.				

REVISION HISTORY

Revision No.	Effective Date	Affected Section(s)	Description of Change(s)
0		All	Initial issue.

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1 PURPOSE

The purpose of this report is to provide a description of the methodology and requirements to establish the design criteria and qualification of the Natrium^{TM} reactor seismic isolation system (SIS) for review and approval by the U.S. Nuclear Regulatory Commission (NRC).

Specifically, approval is sought for the use of the reactor SIS design and qualification methodology described in the following sections:

- Section 7: Reactor Seismic Isolation System Design and Qualification Methodology
    - Sub-section 7.1: Risk-Informed Performance Based Seismic Design and Classification Process
    - Sub-section 7.2: Reactor Seismic Isolation System Industry Standards
    - Sub-section 7.3: Commentary on Seismic Isolation NRC Reports
    - Sub-section 7.4: Reactor Seismic Isolation System Requirement Allocation
    - Sub-section 7.5: Reactor Seismic Isolation System Analysis
    - Sub-section 7.6: Reactor Seismic Isolation System Design and Construction
    - Sub-section 7.7: Reactor Seismic Isolation System Qualification
    - Sub-section 7.8: Reactor Seismic Isolation System Lifetime Management

2 ASSUMPTIONS REQUIRING VERIFICATION AND OPEN ITEMS

There are no assumptions used in the development of this report requiring verification. There are no open items that require future actions to verify and close.

3 INPUTS

The inputs to this report to develop the methodology for design and qualification are comprised of industry technical reports, industry codes and standards, and applicable technical background information, and are referenced throughout this report.

4 TERMINOLOGY

Table 4-1 defines terms, acronyms and abbreviations used in this document.

Table 4-1. Terminology and Abbreviations

Term	Acronym / Abbreviation	Description Definition
Active Mechanical Equipment		Mechanical equipment containing moving parts, which, in order to accomplish its required function as defined in the Qualification Specification, must undergo or prevent mechanical movement. This includes any internal components or appurtenances whose failure degrades the required function of the equipment [1].
Aging		The cumulative effects of operational, environmental, and system conditions on equipment during a period of time up to, but not including, design-basis events or the process of simulating these effects [1].
American Concrete Institute	ACI	American Concrete Institute is a technical and educational society dedicated to improving the design, construction, maintenance, and repair of concrete structures and to advancing concrete knowledge by conducting seminars, managing certification programs, and publishing technical documents.
American Institute of Steel Construction	AISC	The American Institute of Steel Construction (AISC), headquartered in Chicago, is a not-for-profit technical institute and trade association established in 1921 to serve the structural steel design community and construction industry in the United States.
American Society of Civil Engineers	ASCE	The American Society of Civil Engineers (ASCE) is a not-for-profit membership organization whose mission is to facilitate the advancement of technology; encourage and provide the tools for lifelong learning; promote professionalism and the profession; develop and support civil engineers.
American Society of Mechanical Engineers	ASME	ASME is an American professional association that promotes the art, science, and practice of multidisciplinary engineering and allied sciences around the globe via continuing education, training and professional development, codes and standards, research, conferences and publications, government relations, and other forms of outreach.
Anticipated Operational Occurrence	AOO	Anticipated event sequences expected to occur one or more times during the life of a nuclear power plant, which may include one or more reactor modules. Event sequences with mean frequencies of $1x10^{-2}$/plant-year and greater are

Term	Acronym / Abbreviation	Description Definition
		classified as AOOs. AOOs take into account the expected response of all SSCs within the plant, regardless of safety classification [2].
Application Report		Documentation for a specific application showing that the required pressure ratings, qualification loading levels, and operating condition capabilities are equaled or exceeded by the corresponding pressure ratings, qualification loadings, and operating condition capabilities shown in the Qualification Report [1].
Authorized Inspection Agency	AIA	An organization that is empowered by an enforcement authority to provide inspection personnel and services [3].
Beyond-Design Basis Event	BDBE	Rare event sequences that are not expected to occur in the life of a nuclear power plant, which may include one or more reactor modules, but are less likely than a DBE. Event sequences with frequencies of $5x10^{-7}$/plant-year to $1x10^{-4}$/plant-year are classified as BDBEs. BDBEs take into account the expected response of all SSCs within the plant regardless of safety classification [2].
Boiler and Pressure Vessel Code	BPVC	
Candidate Equipment		Active mechanical equipment to be qualified in accordance with the rules of ASME QME-1 [1].
Candidate Restraint		Those components qualified through extension of parent qualification [1].
Component Supports		Structural elements that transmit loads between the components and building structure; intervening elements, such as electric motors and valve operators, are not included in the component support load path [1].
Construction Permit Application	CPA	
Core Barrel Structures	CBS	
Damping resistance		A linear approximation of the relationship of the load velocity characteristics of the viscoelastic damper piston [1].
Defense-in-Depth	DID	An approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures. [2].

Term	Acronym / Abbreviation	Description Definition
Degradation mechanism		A phenomenon or process that attacks (e.g., wears, erodes, corrodes, cracks) the material under consideration [4].
Degradation Mechanism Assessment	DMA	Potential active degradation mechanisms for the SSCs within the RIM Program scope [4].
Degree-of-freedom	DOF	
Design Basis Accident	DBA	Postulated event sequences that are used to set design criteria and performance objectives for the design of Safety- Related SSCs. DBAs are derived from DBEs based on the capabilities and reliabilities of Safety-Related SSCs needed to mitigate and prevent event sequences, respectively. DBAs are derived from the DBEs by prescriptively assuming that only Safety-Related SSCs are available to mitigate postulated event sequence consequences to within the 10 CFR 50.34 dose limits [2].
Drag		The load required to maintain restraint movement at a specific velocity [1].
Dynamic Restraint		Any support that, by design, has a primary purpose of controlling dynamic movement of a pipe or component. Restraints may be single items or assemblies comprising multiple items [1].
Extreme Position		That limit on the piston position relative to the barrel of a viscoelastic damper where the specified damping or stiffness characteristics are no longer applicable [1].
Friction pendulum	FP	
Ground Motion Response Spectra	GMRS	Horizontal and Vertical site characterization response spectra developed from the UHRS in the free field on the ground surface or top of competent material (RG 1.208 [5]).
Guard Vessel	GV	
Inservice Inspection	ISI	Checks or inspections of safety performance functions and characteristics to ensure that any significant degradation is observed and timely corrective actions are taken.
International Atomic Energy Agency	IAEA	The IAEA is an independent intergovernmental, science and technology-based organization, in the United Nations family, that serves as the global focal point for nuclear cooperation.
Isolation damper unit	IDU	A viscoelastic damper unit used as a part of the seismic isolation assembly.
Isolation spring unit	ISU	Assembly of springs typically in parallel used as elastic foundation for seismic isolation.
Lead-rubber isolator	LR	
Light-water reactor	LWR	
Low-damping rubber isolator	LDR	
MANDE expert panel	MANDEEP	

Term	Acronym / Abbreviation	Description Definition
Modular isolated reactor support structure	MIRSS	
Monitoring and NDE	MANDE	A term used in ASME BPVC.XI.2 [4] that encompasses the activities of monitoring, NDE, and surveillance specimen use, as established by the Monitoring and NDE Expert Panel (MANDEEP).
N-certificate holder		An organization holding a Certificate of Authorization, Certificate of Authorization (Corporate), or Quality Assurance Program Certificate issued by ASME. This does not include the holder of a Quality System Certificate or Owner’s Certificate [3].
Non-destructive examination	NDE	An examination by the visual, surface, or volumetric method.
Nuclear power plant	NPP	
Nuclear regulatory commission	NRC	An independent agency of the United Sates government, the NRC regulates commercial nuclear power plants and other uses of nuclear materials, such as in nuclear medicine, through licensing, inspection, and enforcement of its requirements.
Owner	OWN	The organization legally responsible for the construction and/or operation of a nuclear facility including but not limited to the one who has applied for, or has been granted, a construction permit or operating license by the regulatory authority having lawful jurisdiction [3].
Parent Restraint		Components used to initially qualify a given design [1].
Pre-service inspection	PSI	
Pressurized water reactor	PWR	
Previously Qualified Restraint		An ASME BPVC restraint that was qualified to existing industry standards prior to Section QDR and that has an established performance history in similar safety-related applications [1].
Probabilistic risk assessment	PRA	A systematic method for assessing three questions used to define “risk.” These questions consider (1) what can go wrong, (2) how likely it is to go wrong, and (3) what are the consequences. These questions allow better understanding of likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty which can identify risk-significant scenarios. The PRA is used to establish a numeric estimate of risk to provide insights into the strengths and weaknesses of the design and operation of a nuclear power plant.

Term	Acronym / Abbreviation	Description Definition
Qualification Program		The overall cumulative process of specifying, conducting, and documenting the results of those activities required to qualify active mechanical equipment to perform its function in accordance with the Qualification Specification [1].
Qualification Report		Documentation of tests, analyses, operating experience, or any combination of these performed in accordance with this Standard or the Qualification Specification that demonstrates functionality of the active mechanical equipment [1].
Qualification Specification		The specification or portion of the Design Specification that describes the qualification requirements to be met in the qualification of the active mechanical equipment [1].
Rated load		The design load capacity for the restraint based on the use of Service Level A [1].
Reactor building	RXB	
Reactor enclosure system	RES	
Reactor head	RH	
Reactor support structures	RSS	
Reactor vessel	RV	
Regulatory guide	RG	
Reliability and integrity management	RIM	Those aspects of the plant design process that provide an appropriate level of SSC reliability and a continuing assurance that such reliability will be maintained over the life of the plant. RIM aspects include design features important to reliability performance, such as design margins; material selection; testing and monitoring; provisions for maintenance, repair, and replacement; leak testing; and NDE.
Reliability target		A performance goal established for the probability that an SSC will complete its specified function to achieve plant-level risk and reliability goals.
RIM expert panel	RIMEP	
RIPB Seismic Target Performance Goal		Seismic performance goal denoting a mean annual frequency of unacceptable seismic performance, commensurate with the risk objectives of the plant
Risk informed performance-based	RIPB	A licensing Strategy that infers implementation of NEI 18-04 [2]
Rotatable plug assembly	RPA	
Safe Shutdown Earthquake	SSE	Safe shutdown earthquake ground motion is the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional during and/or after a seismic event to assure safe shutdown of the plant.
Safety related	SR	

Term	Acronym / Abbreviation	Description Definition
Seismic isolation system	SIS	
Seismic PRA	SPRA	Probabilistic Risk Assessment that is specific to seismic hazards.
Service Conditions		Postulated conditions specified for environmental, dynamic/static/pressure loadings, material degradation, etc., for normal operation, abnormal operation, and design-basis events [1].
Service Level		Design, Service (A through D), Test Limits and expected performance for each Service Level are provided in ASME.BPVC.III Subsection NCA-2142.4 [3].
Spring Rate		The linear approximation of the relationship of the load displacement characteristics of the restraint [1].
Standard Review Plan	SRP	
Structures, systems and components	SSC	
Umbilicals or umbilical lines		Umbilical lines are nonstructural components and systems (mainly distribution systems) that cross the isolation interface and must sustain the large isolator displacements (or deformations) associated with design basis and extended design basis ground motions. Examples of umbilical lines could include system piping and electrical and I&C cables [6].
Uncertainty (as used in MANDE)		A quantification representing the variability associated with monitoring and non-destructive examination data and includes many technique and application specific parameters such as the minimum detection capability, sizing accuracy, resolution tolerance, repeatability, consistency, etc.
Uncertainty (as used in PRA)		A representation of the confidence in the state of knowledge about the parameter values and models used in constructing the PRA.
Uniform Hazard Response Spectra	UHRS	A set of site specific hazard response spectra developed through a Probabilistic Seismic Hazard Analysis (PSHA). UHRS are developed for multiple not-to-exceed frequencies.

5 BACKGROUND

5.1 Natrium Plant Description

The Natrium plant utilizes a pool-type, metal-fuel sodium fast reactor paired with a molten salt energy island. Using the pool-type molten sodium cooled reactor offers several distinct advantages when compared with traditional light water reactor designs. Molten sodium provides a simplified response to abnormal events due to its large thermal inertia inherent to the large volume of liquid sodium in the reactor. Additional benefits of molten sodium include high thermal efficiency due to its high thermal conductivity and low viscosity, and chemical compatibility with stainless steels, reducing the risk of corrosion or other adverse reactions. Furthermore, sodium remains liquid over the full operating temperature range at near atmospheric pressure, eliminating the need for high pressure primary coolant systems.

The paramount mission of the Natrium reactor is to deliver safe, carbon-free power to society. Its simplified inherently safe design enables deployment across a wide range of sites. The Natrium plant comprises the nuclear island and the energy island. The energy island includes thermal energy storage and a power conversion unit, while the nuclear island includes the reactor and supporting safety-significant structure, systems and components (SSC).

The Reactor Building (RXB) is at the center of the nuclear island located between the Fuel Handling Building and the Reactor Auxiliary Building as shown in Figure 5-1. The RXB houses safety-significant systems including the Reactor Enclosure Systems (RES) which houses the reactor core. There are two main levels in the RXB: the refueling access area floor located at-grade level in the RXB steel-framed superstructure, and the operating deck, also known as the Head Access Area (HAA) located below grade in the reinforced concrete and steel RXB substructure. The HAA provides maintenance access to the reactor head and its associated piping and equipment. The reactor is located within, and supported by, the embedded RXB substructure that provides protection from external hazards. The reactor support design incorporates seismic isolation of the reactor from the RXB substructure to provide enhanced protection against seismic events. Heat generated by the reactor core is transferred through the intermediate heat exchanger to the intermediate sodium loops through piping umbilicals from the Reactor Auxiliary Building (depicted in Figure 5-1).

(a)(4)

Figure 5-1: Cross Section View of Nuclear Island Buildings. From left to right: Fuel Handling Building (FHB), the Reactor Building (RXB) and the Reactor Auxiliary Building (RAB).

5.2 Industry Technical Reports

Despite the lack of implementation of seismic isolation systems (SIS) for nuclear power plants (NPP) in the United States, advanced reactor vendors have an interest in exploring the use of SIS. One such advanced reactor design utilizing SIS appears in General Electric’s preapplication for a liquid metal reactor, PRISM, in the 1980s [7] which utilized composite steel-rubber seismic isolation devices. The US Nuclear Regulatory Commission (NRC) initiated several research programs in the 2000s to examine potential regulatory paths and performance of selected SIS through experimental studies. There are three technical reports prepared for the NRC that discuss potential regulatory guidance and technical considerations for seismic-isolated NPPs. The three reports are NUREG/CR-7253 [6], “Technical Considerations for Seismic Isolation of Nuclear Facilities,” issued February 2019; NUREG/CR-7254 [8], “Seismic Isolation of Nuclear Power Plants using Sliding Bearings,” issued May 2019; and NUREG/CR-7255 [9], “Seismic Isolation of Nuclear Power Plants using Elastomeric Bearings,” issued February 2019.

The objective of NUREG/CR-7253 is to develop and summarize a set of technical considerations, recommendations, and options that could serve as the basis for regulation and regulatory review of the design, construction, and operation of seismic-isolated NPPs. The report presents a risk-informed, performance-based design philosophy for SIS that is intended to be consistent with the NRC’s then-current objectives and criteria approaches. This report focuses on base-isolation of NPPs using two-dimensional (horizontal) bearing type isolation systems. NUREG/CR-7253 does not address the use of three-dimensional SIS for applications such as equipment isolation. NUREG/CR-7253 assumes the isolation of surface or near-surface-mounted, safety-related (SR) structures such as large light water reactor buildings. Although the report focuses on surface-mounted SIS consisting of isolation devices which are active only in the horizontal direction, the authors argue the principles discussed should apply more broadly to additional SIS technologies and mounting configurations.

The objective of NUREG/CR-7254 is to develop and codify a model to characterize the dynamic response of a particular type of horizontal SIS: Concave Friction Pendulum™ (FP) bearing. The report presents the mechanical design of FP bearings and describes the relationship between force and displacement, velocity, and friction coefficient, and the hysteretic dependency of the dynamic response to earthquake shaking. The mathematical model presented in Section 3 of the technical report is implemented in an open-source finite element computer code through use of a specific element library, and numerical results from the model are validated with published experiments. Additionally, the report performs a risk-based calculation to compute design factors for seismically isolated NPPs and understand the impact of modeling decisions and loading conditions.

The objective of NUREG/CR-7255 is to investigate existing applications of SIS in nuclear structures and focuses on elastomeric seismic isolation technology including Low Damping Rubber (LDR) and Lead-Rubber (LR) bearings. The report recommends an appropriate mathematical representation for elastomeric bearing for extreme earthquake shaking to account for non-linear effects such as softening due to cavitation. Models to simulate these characteristics are implemented in open-source and commercially available finite elements software. The models are then exercised using seismic motions scaled to a uniform hazard response spectrum with a return period of 10,000 years and conclusions are drawn regarding the efficacy of the technology for design basis earthquake and beyond design basis earthquake.

The current publicly available technical reports issued by the NRC primarily focus on SIS that are effective only in the horizontal direction and applied to building structures (with a similar configuration to the one shown in Figure 5-2). However, the considerations, principles, and recommendations provided by these documents can be extended to other technologies and applications, including three-dimensional SIS arrangements designed to isolate equipment rather than structures, and risk-informed licensing frameworks to achieve the desired seismic performance objectives. These additional seismic isolation technologies and applications merit additional considerations and requirements beyond the ones noted in the current NRC reports and are outlined in this report.

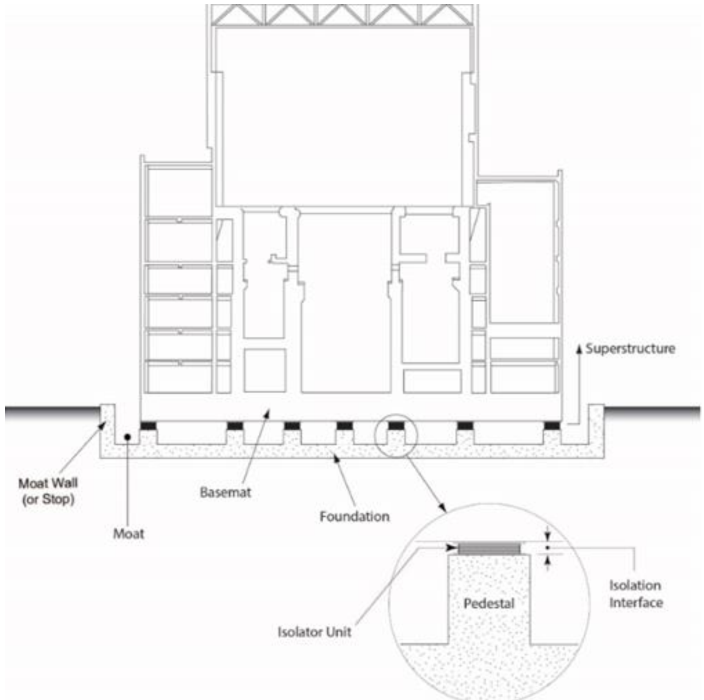

Figure 5-2: Elements of a seismically isolated nuclear plant structure [6]

5.3 Regulatory Precedence

The US NRC has issued its safety evaluation report (SER) of an advanced test reactor for a construction permit [10]. In its evaluation the staff noted in Section 3.5.3.2 of [10]:

"Based on its review of the PSAR, the staff finds that Kairos has provided an adequate level of detail on the seismic isolation system for the preliminary design and for issuance of a CP because, although details of the isolation system have not been specified, the design methodology aligns with a consensus code (ASCE 43-19) and Kairos has clearly identified information that will be provided in the OL application."

The construction permit recommended by the SER to be approved by the US NRC endorses the application of seismic isolation in nuclear facilities which can limit the seismic demand on safety related SSCs.

5.4 Basis for Performance Criteria Adaption

The underlying seismic performance criteria recommended in NUREG/CR-7253 rely on discrete design requirements for design basis and beyond design basis earthquakes, but without explicit evaluation of seismic risk quantified across the whole range of seismic hazard and the associated potential for cliff-edge effects. Consistent with the licensing modernization project approach described in NEI 18-04 [2], and endorsed by RG 1.233 [11], risk insights from functional performance across the complete range of seismic hazard can affect cumulative risk objectives such as quantitative health objectives (QHO). Therefore, although the recommended seismic performance objectives for seismic isolation presented in NUREG/CR-7253 are useful guiding principles, the corresponding design framework recommended is not fully compatible with the risk-informed performance based (RIPB) approach endorsed by RG 1.233 [11].

Furthermore, three-dimensional equipment isolation is distinctly different from two-dimensional full scale building isolation with the following considerations:

- Target isolation frequency for equipment is typically larger (in the range of [[ ]](a)(4)) than those for building isolation (in the range of 0.5-2 Hz). Displacement of the isolated equipment relative to non-isolated structures may be more limited than in building isolation applications.
- Equipment isolation footprint is considerably smaller than that of building isolation, providing smaller scale load distribution control and reduced uncertainties (weight, stiffness, load distribution, stiffness, etc.).
- The total mass of the isolated equipment is significantly reduced when compared with full building isolation, simplifying the analysis and qualification.
- Equipment isolation benefits from three-dimensional isolation by tuning the horizontal and vertical isolation frequencies to balance reduction in seismic demand in all three directions and to minimize rocking of the equipment.
- Equipment isolation is located inside the building providing protection from external environmental conditions to a much greater extent thereby reducing degradation mechanisms. On the other hand, proximity to the reactor may increase the significance of temperature and radiation exposure.
- Access to equipment isolation requires different considerations, including entering potential radiation zones and providing direct access to inspections and maintenance in confined spaces.

Given the limited available regulatory guidance and the considerations listed above, development of a design and qualification methodology for equipment isolation using three-dimensional seismic isolation technology is warranted. Such methodology can adapt seismic performance objectives similar (or equivalent) to existing published guidance as presented in NUREG/CR-7253 for alignment and consistency with the RIPB approach endorsed by RG 1.233 [11].

6 NATRIUM REACTOR SEISMIC ISOLATION SYSTEM

The primary system to be supported by the SIS is the RES, and the preliminary arrangement of the RES is shown in Figure 6-1 (equipment supported by the RES and umbilicals are not shown). The RES includes the reactor vessel and head, which encompasses the reactor core, reactor internal structures, primary sodium coolant, and essential equipment required for coolant circulation and reactor heat rejection. The RES incorporates a guard vessel surrounding the reactor vessel, offering defense-in-depth leak mitigation in the unlikely event of a primary sodium leak from the RV. The RES is supported by the reactor support structure (RSS), which includes the modular isolated reactor support structure (MIRSS), reactor support blocks, and SIS, and provides the load path from the reactor vessel head to the reactor building substructure basemat. The plant equipment that is isolated from the RXB by the SIS includes:

- RES and RES Internals:
    - Reactor Enclosure System, including the Reactor Vessel and Head, Rotatable Plug Assembly, Reactor Internals Structures, and the Reactor Support Structure.
    - Reactor Core System, including the Fuel, Control, Shield and Reflector Assemblies.
- Equipment attached or supported by the RES:
    - Control Rod Drive System, including the Control Rod Drive Mechanisms and drivelines.
    - Primary Heat Transport System, including the Intermediate Heat Exchangers and Primary Sodium Pumps.
    - Reactor Air Cooling System Collector Cylinder.
    - Sodium Cover Gas System components mounted on the Reactor Head.
    - Sodium Processing System, including the Main Heat Exchanger and Pump.
    - In-Vessel Fuel Handling System including the Fuel Transfer Lift and the In-Vessel Transfer Machine.
- RES Umbilicals:
    - Intermediate Heat Transport System piping connected to the intermediate heat exchangers.
    - Sodium Cover Gas System and Sodium Processing System piping.
    - Reactor Instrumentation System instruments and cabling.

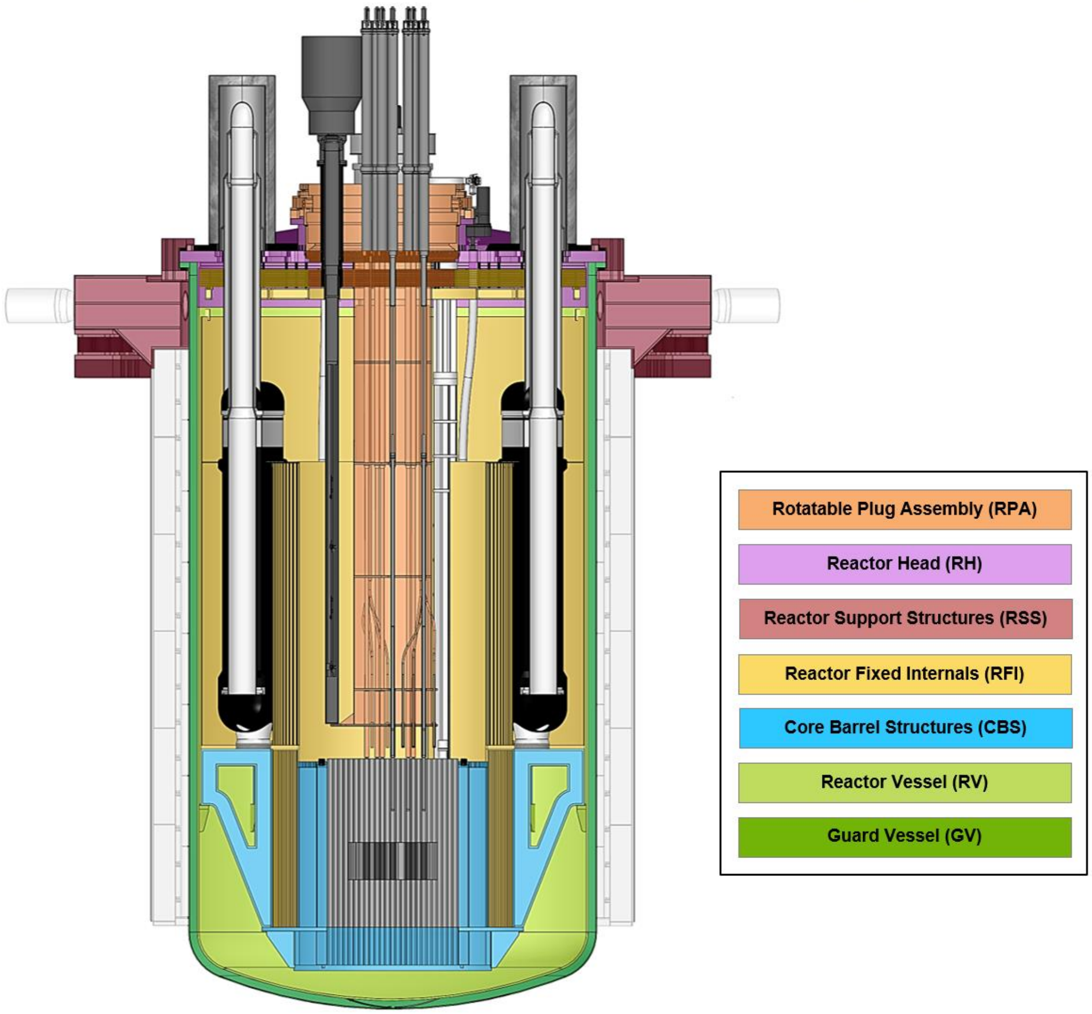

Figure 6-1: Reactor Enclosure System

The SIS is located at the interface between the RSS and the RXB and is attached to the HAA reinforced-concrete basemat and the MIRSS as shown in Figure 6-2. The MIRSS is a plate-girder steel structure which supports the RES on its inner diameter ledge through the reactor support blocks. The MIRSS, along with the reactor support blocks, constrain the reactor head such that a stiff load-path from the reactor head to the SIS is formed, while accommodating the relative thermal growth between the MIRSS and the basemat. The MIRSS also supports the collector cylinder assembly, which is suspended on the underside of the MIRSS around the reactor to support the reactor air cooling function.

The RSS incorporates the SIS, which isolates the reactor from the supporting RXB basemat using three-dimensional SIS technology. The SIS includes multiple isolation spring units (ISU) and isolation damper units (IDU) [[ ]](a)(4). The ISUs are constructed of coiled helical wire springs in parallel between a top and bottom mounting plate. The dampers consist of the damper housing, which is a non-pressurized fluid container filled with viscous damper fluid and a piston immersed in the fluid [12]. The damper housing and the piston are attached to opposite end plates of the damper. As a result of relative movement of the piston within the housing, forces resulting from the motion of the viscous fluid provide effective load transfer and motion damping between the supporting and supported SSC.

The ISUs and IDUs require to be sized and calibrated for adequate attenuation of seismic loads and support of the RES during normal and off-normal conditions. The ISUs serve to provide sufficient separation between the frequencies of motions transmitted from the RXB and the fundamental frequency of reactor internal equipment. The IDUs provide damping forces during seismic motions which limit displacement and the transmitted forces to critical equipment. The IDUs are unloaded when the reactor is at rest and provide negligible resistance to quasi-static motions such as thermal expansion. The ISUs and IDUs are coupled by the stiff MIRSS and HAA basemat to ensure even load distribution within the SIS and limit the seismic demands exerted on the safety related reactor from seismic motions.

(a)(4)

Figure 6-2: Conceptual RES SIS arrangement and interface with the RXB.

7 REACTOR SEISMIC ISOLATION SYSTEM DESIGN AND QUALIFICATION METHODOLOGY

This section outlines the reactor SIS design and qualification methodology including the technical basis, and applicable regulatory guidance. The following topical areas were considered in the overall process for developing the reactor SIS design and qualification methodology for application to three-dimensional equipment isolation utilizing IDUs and ISUs:

- Risk-informed performance based seismic classification and design of the SIS.
- Evaluation of design-specific applicability of industry standards to the SIS including identification of the codes and standards based on operating experience and prior precedence in nuclear applications.
- Evaluation of design-specific applicability of industry technical information for the SIS consistent with the NEI 18-04 [2], RIPB approach. Adaptation of seismic performance objectives similar (or equivalent) to existing published technical information, such as is in NUREG/CR-7253 [6].
- Requirements allocation to establish performance criteria for the SIS. Requirements may be categorized as functional, performance, and interface requirements or design constraints. Requirements are elicited from expected SIS safety classification, technical considerations applicable to reactor seismic isolation based on review of industry technical reports, and applicable codes and standards.
- Reactor SIS analysis to derive the critical SIS parameters that form the basis for qualification and verification of the requirements and performance criteria.
- Design and construction requirements pertaining to ASME certificate holder responsibilities.
- Qualification program description with specific applicability to both ISUs and IDUs that includes augmenting the requirements of the construction code.
- Life-time management description for assuring the reliability and integrity of the SIS over the life of the plant.

The discussion provided for the topical areas in the following sections elaborate on the context, background, and rationale for the presented methodology and will be incorporated as part of the SIS design and qualification basis documents when approved.

7.1 Risk-Informed Performance Based Seismic Design and Classification Process

Natrium is using a RIPB approach to seismic design and qualification that is consistent with NEI 18-04 [2]. NEI 18-04 establishes a RIPB decision making process that incorporates principles of frequency of event occurrences versus consequences of failure and measurable performance objectives.

Seismic design requirements are identified through an iterative process that considers SSC design capability and seismic risk, informed by a seismic probabilistic risk assessment (SPRA). The iterative process establishes the required seismic performance criteria based on SSC seismic risk significance, and seismic special treatments inform SSC design and qualification requirements such that there is reasonable assurance that required seismic performance is achieved. Seismic performance criteria and special treatments are applied commensurate with the SSC safety-significance and contribution to seismic risk. The SIS is classified safety-related and has been determined to be a risk significant contributor to overall plant seismic risk, and a rigorous approach to the identification of seismic performance requirements and the application of seismic special treatment has been developed, as described herein.

The RIPB approach to seismic design and qualification supplements existing and applicable regulations to nuclear power plants.

7.1.1 Seismic Design Basis Hazard Level

The Design Basis Hazard Level (DBHL) for Natrium is established as the Safe Shutdown Earthquake (SSE) based on guidance provided in RG 1.208 [5]. A probabilistic seismic hazard analysis (PSHA) is performed as part of a Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 study. The PSHA, in combination with a probabilistic site response analysis, is used to develop site-specific Uniform Hazard Response Spectra (UHRS) within the site profiles needed for performing seismic analysis. The hazard consistent, site-specific ground motion response spectra (GMRS) is developed from the UHRS at or near the ground surface or top of competent material using guidance from RG 1.208 [5] and NUREG/CR-6728 [13]. The GMRS is used to develop the SSE and forms the basis for development of foundation input response spectra using soil-structure interaction analysis.

The seismic DBHL is initially used in the Natrium CPA to inform Licensing Basis Event (LBE) selection and the safety classification process under NEI 18-04 [2]. In addition, the seismic DBHL is used to satisfy the requirements for the Seismic Design Basis Accident (DBA) LBE, which evaluates the design for SR SSCs to withstand the effects of the seismic DBHL without loss of capability to perform their required safety functions. In addition to evaluating DBAs, full implementation of NEI 18-04 [2] requires the evaluation of event sequences for selection of LBEs that include Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), and Beyond Design Basis Events (BDBEs) to the full range of seismic events to confirm the adequacy of safety classifications and special treatments.

The SSE bounds terminology for the GMRS and DBHL and will be used for establishing the reference DBHL for SIS qualification requirements.

7.1.2 Seismic Classifications and Seismic Special Treatments

Graded seismic classifications are used to assign seismic special treatments and associated design requirements for SSCs and safety functions, consistent with the safety classifications developed under NEI 18-04 [2]. Seismic Classifications, or seismic special treatment categories, are assigned a set of seismic special treatments that inform the design, qualification, and quality requirements for project life cycle phases and define seismic design requirements via a graded application of codes and standards. The resulting seismic performance of SSCs can be verified through feedback between SSC design and the seismic PRA via the fragilities associated with the seismic special treatment categories to evaluate the seismic risk significance of SSCs against a RIPB seismic target performance goal.

The adequacy of the assigned seismic classifications and associated special treatments, are evaluated through an iterative process between design and SPRA. The overall process is outlined as follows:

- Initial safety classifications assigned to SSCs consistent with the NEI 18-04 [2] safety classification process: safety related (SR), non-safety related with special treatment (NSRST), non-safety related with no special treatment (NST).

- Preliminary seismic design for SSCs is performed based on assigned seismic criteria.
- Update SPRA using conservative SSC fragilities developed from assigned seismic criteria.
- Feedback from SPRA results used to evaluate SSC seismic risk significance and update SSC fragilities to meet NEI 18-04 [2] risk targets.
- Update SSC seismic design and fragilities, and iterate based on SPRA feedback as needed.

Feedback from SPRA is used to evaluate whether seismic classifications for SSCs and/or safety functions warrant a change based on seismic risk significance and will identify seismic event sequences that require SSC specific or refined fragilities beyond those established by the seismic classifications. These fragilities may be used to inform additional SSC specific seismic special treatments to meet Natrium's risk objectives and associated SSC seismic performance criteria to supplement the design and qualification requirements for the SIS. The overall RIPB SIS seismic design and seismic classification process is illustrated in Figure 7-1.

The reactor SIS ISUs and IDUs are assigned a SR designation under the NEI 18-04 [2] safety classification process and has a SR seismic risk significant special treatment category designation, consistent with an SCS1 seismic classification. The seismic special treatments associated with the SCS1 classification specify that, at minimum, SSCs are designed to withstand the effects of the SSE and remain functional and the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 [14] will be applied to all activities affecting the safety-related functions of the SIS, which is reflected in the selection of applicable Codes and Standards.

(a)(4)

Figure 7-1: Seismic Isolation System risk informed performance-based design process.

7.2 Reactor Seismic Isolation System Industry Standards

ASME Boiler and Pressure Vessel Code, Section III

The design and construction of high-temperature reactor coolant boundary components, including the reactor vessel and head, are governed by ASME BPVC Section III, Division 5 (ASME.BPVC.III.5) "High Temperature Reactors" [15], as endorsed by RG 1.87 [16]. Per RG 1.87 Table A-1, the RV and RH are ASME.BPVC.III.5 Class A components, and Subsection HB Class A Metallic Pressure Boundary Components and Subpart B Elevated Temperature Service (HBB) apply. In accordance with the requirements of ASME.BPVC.III.5 [15], supports of high temperature reactor systems are designed and constructed to the requirements of ASME.BPVC.III.5, Subsection HF Class A and Class B Metallic Supports. Preliminary analyses indicate that the RSS temperatures do not exceed the permitted values in ASME.BPVC.III.5 HAA-1130-1 and therefore, Subpart A Low Temperature Service applies. Furthermore, per ASME.BPVC.III.5 HFA-1110(b), the rules of Subsection HF, Subpart A are contained in ASME.BPVC.III.1, Subsection NF [17].

In accordance with ASME.BPVC.III.5 HFA-1110(g), the rules of ASME.BPVC.III.1 Subsection NF do not apply to spring elements and damper unit hydraulic fluid, except for the following requirements:

1. Material shall tolerate the environmental conditions.
2. The exempt item shall be designed to the same loading as other requirements for non-exempt parts.
3. The design specification and design report shall indicate the exempt items.
4. Materials, fabrication, and installation shall comply with the design output documents.
5. Spring coils shall be inspected to ASME.BPVC.III.1 NF-2520.
6. Compression spring (soft compression stops) end plates shall comply with ASME.BPVC.III.1 NF-3000, NF-4000, NF-5000, and NF-8000.
7. Compression dynamic stops shall comply with ASME.BPVC.III.1 NF-3000, NF-4000, NF-5000, and NF-8000.

NUREG-2245 [18] provides a technical review of ASME.BPVC.III.5. NUREG-2245 Section 3.17 [18] details the review of Subsection HF Class A and Class B Metallic Supports with the conclusion that the NRC staff finds the ASME BPVC acceptable for designing Class A component supports such as the supports for the RV.

Jurisdictional boundaries between component supports and the building structure are governed by ASME.BPVC.III.1 NF-1132. Figure NF-1132-1 includes typical examples of jurisdictional boundaries. Figure NF-1132-1(e) shows a typical arrangement in which a damper element is included with the component support (also shown in Figure 7-2). The damper and the supporting steel structure attached to the building structure is under the jurisdiction of ASME.BPVC.III.1 Subsection NF. There are three types of support categories in accordance with NF-1200. Standard supports are those identified in NF-1214 that include constant and variable type springs, and dampers. Typical examples of standard supports are provided in Figure NF-1214-1 (see Figure 7-2). [[

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[[ ]](a)(4) As such, the ASME BPVC Section III provides a complete set of rules for construction (including design, materials, fabrication, testing, examination, and installation) commensurate with ASME NQA-1 [19] and NCA [3] (N-type certification). The ASME BPVC is flexible to accommodate a range of user-specified seismic performance criteria.

The use of ASME.BPVC.III.1, Subsection NF is prevalent in NPPs. In accordance with NUREG-0800 Standard Review Plan (SRP) 3.9.1 [20] the operating fleet reactor vessel supports were designed and constructed to one of the Editions of ASME.BPVC.III.1, Subsection NF. Reactor vessel supports in recent license approvals for the Westinghouse AP1000 [21], GEH ESBWR [22], and NuScale US600 [23] also used one of the Editions of ASME.BPVC.III.1, Subsection NF for reactor supports. The precedence cited did not employ seismic isolation systems.

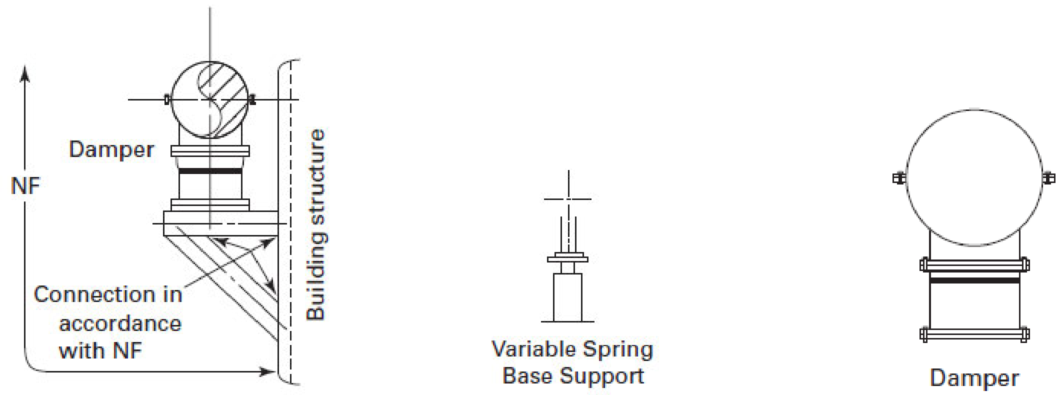

Figure 7-2: ASME jurisdictional boundary example (left); spring standard support (middle); damper standard support (right)

<u>ASME QME-1</u>

The purpose of ASME QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Facilities, [1], as endorsed by RG 1.100 [24], is to provide requirements to qualify active mechanical equipment based on critical characteristics to meet functional requirements for licensing basis events. The reactor SIS falls under the qualification program of ASME QME-1. Section QDR of ASME QME-1 provides rules for qualification of dynamic restraints and section QR discusses the associated general requirements. The boundaries of jurisdiction of ASME QME-1 align with those defined in ASME.BPVC.III.1, Subsection NF and governed by QDR-1100. Qualification principles based on functional requirements pertaining to the spring rate and damping resistance (defined in QDR-3000) of dynamic restraints are in QDR-4000 (QDR-4400 Viscoelastic Dampers). In accordance with QDR-5000, the Owner shall provide a Qualification Specification which is reconciled with the Design Specification per ASME.BPVC.III.1, Subsection NF. The qualification program, governed by QDR-6000 and QDR-7000 (for documentation), generally includes the following elements:

- Approach to qualification (QDR-6210).
- Testing plan (QDR-6220):
    - Installation and orientation
    - Test and monitoring equipment

  - Test sequence
  - Functional parameter testing for springs
  - Functional parameter testing for viscoelastic dampers
  - Aging and service condition simulation
  - Limits or failure definition
  - Post-test examination and analysis
- Parent or candidate qualification (similitude and analysis, QDR-6200 and QDR-6300).
- Documentation requirements (QDR-7000):
  - Qualification plan (QDR-7200)
  - Qualification and application reports (QDR-7300)
- Additional details on the content of the Qualification Specification are provided in Mandatory Appendix QDR-I.

Therefore, the use of ASME QME-1 augmenting ASME.BPVC.III provides the basis for complete qualification of the SIS prior to placing it in service. The qualification establishes the baseline for inservice activities including monitoring, inspection, testing, maintenance, and surveillance.

<u>ASME BPVC, Section XI</u>

For completeness, it is noted that the Reliability and Integrity Management (RIM) program, outlined in the ASME BPVC, Section XI, Division 2 (ASME.BPVC.XI.2) [4], applies to safety related supports. The 2019 Edition of this code was endorsed by the U.S. NRC in RG 1.246 [25]. The reactor and its support including the SIS is included under the RIM program which addresses inservice inspections, monitoring, and surveillance for the entire operating life of the plant.

The reactor support SIS design and qualification codes and standards applicability is summarized in Figure 7-3.

(a)(4)

Figure 7-3: Reactor Seismic Isolation System Design and Qualification Applicable Codes and Standards

7.3 Commentary on Seismic Isolation NRC Reports

The information provided in NUREG/CR-7254 [8] and -7255 [9] was reviewed and determined not to apply to the Natrium reactor SIS design and qualification methodology. Review of NUREG/CR-7253 [6] determined that the guidance warranted a detailed assessment of applicability of technical considerations.

The stated objective of NUREG/CR-7253 is to develop and summarize a set of technical considerations, recommendations and options that could serve as the basis for regulation and regulatory review of the design, construction, and operation of seismic-isolated NPPs. The report states that it presents a risk-informed, performance-based design philosophy for seismic isolation that is consistent with the NRC's then-current objectives and criteria approaches.

Based on a review of NUREG/CR-7253, the underlying performance assumptions rely on discrete design requirements for design basis and beyond design basis earthquakes without explicit evaluation of seismic risk resulting from the full range of external hazards, and the associated potential for cliff-edge effects. NUREG/CR-7253 is not consistent with the licensing modernization approach described in NEI 18-04 [2], which relies on evaluation of risk insights from the full range of external hazards to meet QHOs. Furthermore, NUREG/CR-7253 focuses on base-isolation of NPPs using horizontal bearing type isolation systems. As noted in Section 5.4, three-dimensional equipment isolation is distinctly different from two-dimensional full scale building isolation.

However, the technical considerations presented in NUREG/CR-7253 were reviewed to identify relevant inputs, and the results of the review are provided in the form of a commentary on the applicability to the Natrium three-dimensional reactor SIS design and qualification methodology in Table 7-1.

Table 7-1 identifies each technical area applicability in three categories:

- Applicable – the technical area or requirement is applicable and corresponding requirement(s) is adapted for which compliance will be demonstrated.
- Not Applicable – the technical area or requirement is not applicable to the three-dimensional reactor SIS.
- Meet Intent – the technical area or requirement is not directly applicable however the underlying intent or performance target is adapted in an alternate requirement which meets the intent.

Those requirements that are deemed Applicable, or Meet Intent, are mapped to SIS requirements presented in Section 7.4.

Table 7-1. Commentary on NUREG/CR-7253

Technical Area (NUREG/CR-7253)	Technical area applicability / reference requirement	Commentary
Section 1, 2, 3.1, and 3.2: Introduction, A brief history of seismic isolation, and Basics of Seismic Isolation	[[ ]](a)(4)	[[ ]](a)(4)
Section 3.3: The following qualification tasks should be accomplished before a new type of "three-dimensional SIS" is used. a) Dynamic testing full-scale (prototype) for beyond design ground motions (BDGM). b) Development of verified and validated numerical models. c) Demonstration that mechanical properties do not change by more than 20% over the design lifetime. d) System-level testing of the isolation system using three translational components of earthquake ground motion. e) Verification and validation of numerical tools used to predict response.	a) [[ ]](a)(4)	a) [[ ]](a)(4)

Technical Area (NUREG/CR-7253)	Technical area applicability / reference requirement	Commentary
	[[ ]](a)(4)	[[ ]](a)(4)
Sections 5 through 6: domestic and international codes and standards.	[[ ]](a)(4)	[[ ]](a)(4)
Section 7-1: Analysis of seismically isolated structures.	[[ ]](a)(4)	[[ ]](a)(4)

Technical Area (NUREG/CR-7253)	Technical area applicability / reference requirement	Commentary
Section 7-2: Modeling of seismic isolator units, equivalent linear models, and non-linear models of isolators.	[[ ]](a)(4)	[[ ]](a)(4)
Section 8.1: Philosophy in developing performance criteria.	[[ ]](a)(4)	[[ ]](a)(4)

Technical Area (NUREG/CR-7253)	Technical area applicability / reference requirement	Commentary
		[[ ]](a)(4)
Section 8.2: Performance matrix, isolators and isolation system, foundation, umbilical lines, stop.	[[ ]](a)(4)	[[ ]](a)(4)
Section 8.3: Hazard definitions for analysis of SIS.	[[ ]](a)(4)	[[ ]](a)(4)
Section 8.4: Performance expectations for ground motion response spectrum+ shaking.	[[ ]](a)(4)	[[ ]](a)(4)
Section 8.5: Performance expectations for beyond design	[[ ]](a)(4)	a) [[ ]](a)(4)

Technical Area (NUREG/CR-7253)	Technical area applicability / reference requirement	Commentary
basis ground motion response spectrum shaking. a) Clearance to the stop b) Isolators c) Umbilical lines d) Stop	a) [[ ]](a)(4)	[[ ]](a)(4)
Section 8.6: Seismic probabilistic risk assessment (SPRA).	[[ ]](a)(4)	e) [[ ]](a)(4)
Section 9.1: Additional considerations: a) long-term changes in isolator mechanical properties b) basemat and foundation design c) anchorage design d) other external events	a) [[ ]](a)(4)	a) [[ ]](a)(4)

Technical Area (NUREG/CR-7253)	Technical area applicability / reference requirement	Commentary
e) accident conditions and emergency response f) moat cap design g) near-fault ground shaking h) peer review	e) [[ ]](a)(4)	[[ ]](a)(4)

Technical Area (NUREG/CR-7253)	Technical area applicability / reference requirement	Commentary
		[[ ]](a)(4)
Section 9.2: Additional manufacturing and construction considerations (quality control and quality assurance, testing of prototype and production isolators, construction assurance).	[[ ]](a)(4)	[[ ]](a)(4)
Section 9.3: Operation considerations: a) Inservice inspection, replacement and maintenance	a) [[ ]](a)(4)	a) [[ ]](a)(4)

Technical Area (NUREG/CR-7253)	Technical area applicability / reference requirement	Commentary
b) additional seismic monitoring equipment c) monitoring of foundation deformations d) requirements for safety-related equipment e) operating temperature	c) [[ ]](a)(4)	[[ ]](a)(4)

7.4 Reactor Seismic Isolation System Requirement Allocation

Reactor three-dimensional seismic isolation performance criteria represent the set of requirements that the three-dimensional SIS must meet in order to provide assurance of acceptable performance. Requirements may be categorized as design constraints, functional, performance, and interface requirements. The requirements presented are elicited from the expected SIS safety classification, technical considerations applicable to reactor seismic isolation based on review of industry technical reports, and applicable codes and standards associated with the design and qualification of the SIS comprised of ISUs and IDUs. Compliance with requirements may be verified by one or more of the following methods:

- **Analysis**: The verification of a product/system using models, calculations, and/or testing equipment. Analysis allows someone to make a predictive statement about the typical performance of the product/system based on the confirmed results of a sample set or by combining the outcome of individual tests to conclude something new about the product/system. This can include analyses via analogy or similarity.
- **Inspection**: The nondestructive examination (NDE) of a product/system using one or more methods.
- **Demonstration**: The manipulation of the product/system as it is intended to be used to verify that the results are as planned or expected.
- **Acceptance Testing**: To establish that the unit is performing correctly and within predetermined tolerances.
    - The acceptance process may include inspections, testing, as well as other activities and shall be performed on SSCs produced. The acceptance test procedure shall be performed on qualification SSCs and other production SSCs, as well. The tests and other activities are intended to establish the SSC is performing correctly and within predetermined tolerances. The procedure shall include acceptance criteria.
- **Qualification Testing**: To establish that the SSC will perform its intended function under any foreseeable operating condition and shall be performed in accordance with an approved qualification test procedure.
    - The qualification process may include inspections, tests, analysis, other activities and shall establish, as far as practical, under laboratory conditions, that the SSC will perform its intended function under any foreseeable operating condition and shall be performed in accordance with an approved qualification test procedure.
    - Each qualification test is to be accomplished on an SSC(s), which is representative of future production SSCs. Qualification tests may be performed at the place of manufacture or by an approved testing laboratory.
    - The procedure shall include acceptance criteria.
    - Qualification test SSCs will generally not be acceptable for delivery as production SSCs unless approved; [[ ]]$^{(a)(4)}$

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]]$^{(a)(4)}$

7.4.1 Seismic Isolation System risk-informed performance-based requirements allocation strategy

In accordance with the RIPB seismic classification process outlined in Section 7.1 the SIS is classified as safety related (SR) and seismic risk significant (SRS) with the seismic special treatment categories and seismic special treatments summarized in Table 7-2. The special treatments planned to be applied to the SIS reflect the underlying intent presented in NUREG/CR-7253 [6] by addressing quality, construction, inspection, and operational standards, as well as cliff-edge effects.

Table 7-2. Seismic Isolation System Anticipated Seismic Special Treatment Category and Seismic Special Treatments

Seismic Special Treatment Category	Seismic Demand	Design Criteria
SR SRS: Safety Related, Seismic Risk Significant	SSE Local demands developed from seismic response analyses (site or building).	Structural design: • AISC N690 (steel) [31] • and ACI 349 (concrete) [32] Mechanical: • Construction ASME.BPVC.III [33] • and qualification ASME QME-1 [1]
SIS Seismic Special Treatment: • Seismic isolation system shall exhibit no damage for SSE shaking. • The isolated system shall be spaced at least at a distance from adjacent construction at the elevation of the isolator units equal to the maximum displacement necessary to achieve seismic target performance goal. ○ If utilized any displacement stop shall be spaced at least at a distance equal to the maximum SSE displacement plus margin, such that it is free to displace without impedance up to this distance. • Seismic isolation system shall retain gravity-load capacity when subjected to deformations consistent with the minimum distance to adjacent construction or the full compression of a displacement stop.		

The performance-based design philosophy of the SIS enables the multi-objective search of design solutions that leverage risk insights with other design constraints and goals such as isolation frequency tuning, spatial arrangement, and inspectability. [[

]]$^{(a)(4)}$ These are competing priorities and require careful balancing of the objectives. The SIS seismic special treatments align with the NEI 18-04 [2] event classification derived from the NEI 18-04 [2] F-C target which at a high level consists of [[

]]$^{(a)(4)}$

(a)(4)

Figure 7-4: Licensing Modernization Project event types by frequency of event

Table 7-3. Seismic Isolation System performance expectations

(a)(4)

The third element in establishing the SIS requirements is to consider the performance envelope of the SIS which is mapped to the event classes of Figure 7-4 and performance expectations of Table 7-3. The SIS is primarily characterized by its stiffness (force displacement relation), displacement capacity and the viscous damper coefficient. The requirements and qualification program focuses on ensuring:

- [[

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The SIS performance envelope is graphically represented in Figure 7-5. The figure provides a simple interpretation to guide the qualification and testing requirements that serve as the basis for demonstrating acceptable performance for the full range of NEI 18-04 [2] events [[

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The springs are tested to the [[

]](a)(4) for each performance state. Using this testing sequence allows for verifying the theoretical stiffness in each direction, any coupling between the directional stiffness, and assessing vertical gravity load carrying stability.

The IDU is tested to [[

]](a)(4)

Testing is performed using [[

]](a)(4) The following sections outline the requirements that constitute the design basis and lifetime performance characteristics of the Natrium three-dimensional reactor SIS.

(a)(4)

Figure 7-5: Seismic Isolation System performance envelope

7.4.2 Functional Requirements

7.4.2.1 Seismic Isolation Direction

The SIS shall be effective in the three orthogonal directions (vertical and horizontal, i.e. three-dimensional seismic isolation).

Rationale: Advanced reactors operating at high temperatures and at near atmospheric pressure utilize structures that require reduced thickness to manage loads, resulting in relatively flexible structures. Head mounted advanced reactors benefit from seismic attenuation in all spatial directions.

7.4.2.2 Seismic Load Attenuation for SSE

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7.4.2.3 Seismic Load Attenuation for BDBE

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7.4.2.4 Seismic Isolation System Operation

The SIS shall require no external power or control for licensing basis and other quantified events.

Rationale: Consistent with passive advanced reactor design objective.

7.4.2.5 Seismic Isolation System Vertical Load Path

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7.4.2.6 Seismic Isolation System Centering

The SIS shall provide sufficient restoring force to re-center the supported SSCs within acceptable tolerance after an SSE.

Rationale: Recentering safety significant SSCs after an SSE is desirable to maintain configuration of SSCs relative to each other.

7.4.2.7 Seismic Isolation System Service Life

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7.4.3 Design Constraint and Quality Requirements

7.4.3.1 Seismic Isolator Construction Code

The SIS shall conform with the requirements of ASME Boiler and Pressure Vessel Code, Section III, Division 5, Subsection HF, 2017 Edition.

Rationale: RG 1.87 [16] endorses ASME.BPVC.III.5 “High Temperature Reactors” [15] as an acceptable means to meet regulatory expectation for SSCs. The Seismic Isolators fall within the jurisdictional boundary of ASME.BPVC.III.5, Subsection HF, Class A and Class B Metallic Supports, 2017 Edition.

7.4.3.2 Seismic Isolator Qualification

The SIS shall be qualified in accordance with ASME QME-1, 2017 Edition.

Rationale: The purpose of ASME QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Facilities, [1] as endorsed by RG 1.100 [24] is to provide requirements to qualify mechanical equipment based on functional and critical characteristics requirements for licensing basis events. The Reactor SIS falls under the qualification program of ASME QME-1. Section QDR of ASME QME-1 provides rules for qualification of dynamic restraints and section QR discusses the associated general requirements. The boundaries of jurisdiction of ASME

QME-1 align with those defined in ASME.BPVC.III.1, Subsection NF and governed by QDR-1100.

7.4.3.3 Seismic Isolator Reliability and Integrity Management

The SIS shall conform with ASME BPVC, Section XI, Division 2, 2019 Edition for monitoring, inservice inspections, and surveillance for the entire operating life of the plant.

Rationale: The SIS is required to maintain reliability over the life of the plant which include operational considerations for inservice inspection, replacement and maintenance. The Reliability and Integrity Management (RIM) Program provides direction for assuring the reliability and integrity of passive components whose failure could adversely affect plant safety and reliability. The RIM Program is outlined in the ASME BPVC, Section XI, Division 2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants”. The 2019 Edition of this code is endorsed by the U.S. NRC in RG 1.246 [25].

7.4.3.4 Seismic Isolator Quality Assurance

The SIS shall comply with ASME NQA-1 2015 Edition, Quality Assurance Requirements for Nuclear Facility Applications [19] and 10 CFR 50, Appendix B [14].

Rationale: Commensurate with the SIS safety significance and seismic risk significance imposing the most stringent quality provision provides assurances for the highest standards in manufacturing, construction, installation, operation over the life-cycle of the plant.

7.4.3.5 Seismic Isolator Fabrication

The SIS fabricator shall have an issued and active ASME NS-Certificate for construction of supports.

Rationale: Certificate holder responsibilities are included in ASME.BPVC.III, Subsection NCA-3200 [3]. Commensurate with the SIS safety significance and seismic risk significance imposing the most stringent quality provision provides assurances for the highest standards in manufacturing, construction, installation, operation over the life-cycle of the plant.

7.4.3.6 Seismic Isolator Installation

The SIS shall be installed in the plant by a Supplier that has an issued and active ASME NA and/or NS-Certificate.

Rationale: ASME.BPVC.III. NCA-1282 [3] provides requirements for support installation certificates. Commensurate with the SIS safety significance and seismic risk significance imposing the most stringent quality provision provides assurances for the highest standards in manufacturing, construction, installation, operation over the life-cycle of the plant.

7.4.3.7 Seismic Isolator Stop Design

Compression spring end plates and compression dynamic stops shall conform with ASME.BPVC.III.1 Subsection NF, Articles NF-3000, NF-4000, NF-5000, and NF-8000.

Rationale: If used, compression dynamic stops used as stops do not need to meet the requirements of ASME.BPVC.III.1, Subsection NF with the exception of the code articles as described in ASME.BPVC.III.5 HFA-1110(g)(8 and 9).

7.4.3.8 Seismic Isolator Parameters

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7.4.3.9 SIS Material of Construction Flammability

The SIS shall use non-combustible materials for construction.

Rationale: To reduce risk of fire in proximity to safety significant components construction materials should be non-combustible.

7.4.3.10 SIS Inservice Position Indication

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7.4.3.11 Seismic Isolator Analysis Methods

The SIS analysis shall conform to the analysis methods outlined in ASCE 4-16 [34].

Rationale: The criteria for seismic analysis outlined in ASCE 4-16 industry consensus standard is applicable to the Natrium seismic analyses.

7.4.4 Performance Requirements

7.4.4.1 Seismic Isolator Reliability for SSE

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7.4.4.2 Seismic Isolation Redundancy

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7.4.4.3 Seismic Isolator Debris Exclusion

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7.4.4.4 Seismic Isolator Displacement Capacity

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7.4.4.5 Seismic Isolator Extended Displacement Capacity

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7.4.4.6 Seismic Isolator Uplift

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7.4.4.7 Seismic Isolator Displacement Clearance

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7.4.4.8 SSE SIS Testing

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7.4.4.9 Licensing Basis Events SIS Testing

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7.4.4.10 SIS Failure Mode Testing

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7.4.4.11 Impact Assessment of Dynamic Stop

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7.4.4.12 Seismic Isolators Long-Term Performance

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7.4.4.13 Seismic Isolators Differential Settlement

The SIS analysis shall address short-, and long-term effects of differential settlement of the soil and foundation flexibility.

Rationale: Differential settlement of the foundation could result in uneven loading of the isolators and redistribution of loads that could lead to failure of the SIS.

7.4.5 Environmental Requirements

7.4.5.1 Seismic Isolator Protection against External Events

The SIS shall be protected against or designed for, fire, high winds, flood and other hazards consistent with the licensing basis of the plant.

Rationale: Safety-related equipment shall meet performance expectations for external events.

7.4.5.2 Seismic Isolator Environmental Conditions

The SIS design shall accommodate environmental degradation due to aging effects, creep, fatigue, operating temperature, radiation and exposure to moisture or damaging substances.

Rationale: Appropriate environmental conditions shall be accounted for in accordance with ASME.BPVC.III.5 [15], ASME QME-1 [1], and ASME.BPVC.XI.2 [4], for the construction, qualification and life-time operation of the SIS.

7.4.6 Interface Requirements

7.4.6.1 Seismic Isolator Interfacing Structures

Structures directly interfacing with the SIS shall be designed with adequate rigidity to ensure all seismic isolators are engaged.

Rationale: The interfacing support structures should be stiff enough to ensure even load distribution between the SIS units. Stiff interfaces ensure that failure of a single isolator does not result in significant load redistribution.

7.4.6.2 Isolated SSCs Clearance to Non-isolated SSCs

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7.4.6.3 Umbilical Lines Crossing the Isolation Interface

Safety-significant umbilical lines and their connections across the isolation interface shall be shown to accommodate the maximum displacement of the SIS.

Rationale: Ensuring adequate displacement capacity of umbilical lines at the isolation interface to mitigate adverse interaction.

7.4.6.4 Interfacing SSCs Redundancy

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7.4.6.5 Interfacing SSCs Loads

SSCs directly interfacing with the SIS shall be designed for loads developed in the SIS corresponding to its maximum displacement or impact loads due to engaging a dynamic stop.

Rationale: Ensuring that the anchorage and support structure design at the interface is robust to transmit the loads through the interface by not presenting a weak link is necessary.

7.4.6.6 Attachment to Interfacing Structures

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7.5 Reactor Seismic Isolation System Analysis

The SIS key dynamic characteristics include its stiffness and damping. These characteristics are tuned by analysis and validated by testing to meet the functional and performance requirements of the SIS. The SIS must also maintain practical dimensions to accommodate space allocation in the RXB and reduce design constraints. The main consideration in tuning the spring stiffness is a compromise between limiting deflections between the isolated and non-isolated structures, while maximizing the attenuation of accelerations over the frequency range of interest of the seismically isolated systems. The SIS displacement capacity must also be sufficiently large to accommodate the required range of deflection and additional safety margin, while maintaining structural integrity and stability.

Analyzing SIS performance and the development of attenuated seismic loads on RES SSC is achieved through using a [[

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]][(a)(4)] While a detailed review of the modeling methods is beyond the scope of this report, the models, methods and outputs conform to relevant requirements in Section 7.4.

7.6 Reactor Seismic Isolation System Design and Construction

As outlined in Section 7.2, the SIS components are ASME.BPVC.III.5 HFA standard supports, designed and constructed under the rules of ASME.BPVC.III.1, Subsection NF. Per NF-3411.1, standard supports can be used as component supports, such as the supports for the RV. The ASME Code is a construction code, which means that it covers the entire lifecycle from Design through Installation by providing a complete set of rules for construction (design, materials, fabrication, testing, examination, and installation). The specific ASME activities applicable during the construction of the SIS are shown in Figure 7-6. In order to perform any ASME activities, an organization shall obtain and maintain an appropriate ASME certificate of authorization. The various N-type Certification of authorizations (N-Certificates) necessary for the construction of the SIS are shown in Table 7-4.

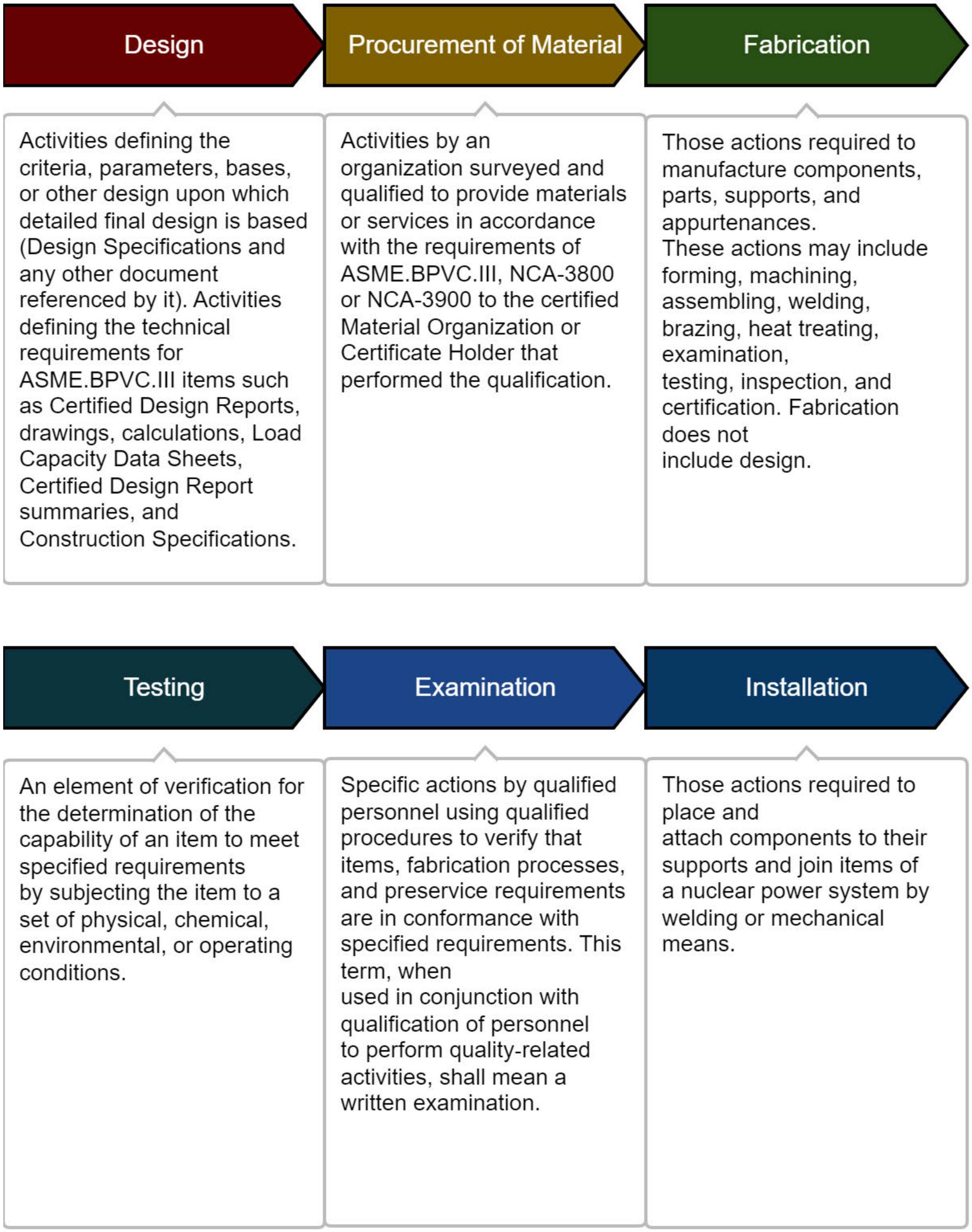

Figure 7-6. ASME BPVC Section III Nuclear Facility Construction Activities [3].

Table 7-4. ASME Issued N-Type Certificates and Scopes Necessary for SIS

N-type Certificate	Description
N	Construction of vessels, pumps, valves, piping systems, storage tanks, core support structures and concrete containments – maintains overall design responsibility
NA	Field installation and shop assembly of all items
NS	Fabrication of supports with or without design responsibility
OWN	Nuclear power plant Owner
QSC	Manufacture and supply of material

In accordance with ASME.BPVC.III NCA-1233, the Design Conditions of the SIS shall be included in the Design Specification. In addition, a Design Report, Load Capacity Data Sheet, or Design Report Summary shall be furnished. Certification of the documents shall be as required by ASME.BPVC.III NCA-8000. The Owner (or if designated, the N-Certificate Holder) has the overall design responsibility. The N-Certificate holder is responsible for compiling all lifetime and non-lifetime quality records and turning them over to the Owner. Where certification of documents is required, it shall be provided by certifying engineers (CE) meeting the qualification requirements of ASME.BPVC.III Mandatory Appendix XXIII [3].

As the SIS is classified as standard supports, an NS-Certificate is required for the fabrication of the ISU and IDU. Installation of the SIS shall be in accordance with ASME.BPVC.III NCA-1282 which consists of those activities required to attach the SIS to the building structures and other reactor support structures. The installation shall be performed by an NA-Certificate holder at the location authorized by its certificates. Specific responsibilities of certificate holders are in ASME.BPVC.III NCA-3211. Items constructed in accordance with ASME.BPVC.III shall be inspected by Authorized Inspection Agency (AIA) accredited by ASME. Certificate holders shall enter into a written agreement with an AIA as stipulated in NCA-3200.

7.7 Reactor Seismic Isolation System Qualification

The reactor SIS is an active mechanical equipment in accordance with ASME QME-1 [1] because it must undergo mechanical movement in order to accomplish its required function. The SIS consists of separate ISUs and IDUs as described in Section 6. The qualification program applies to both with consideration of the specific characteristics of the seismic isolation units. The qualification augments the requirements of ASME.BPVC.III.1 Subsection NF.

Qualification of the SIS follows the process described in ASME QME-1 QR General Requirements. The qualification process for the SIS is illustrated in Figure 7-7 conforming to the qualification principles outlined in ASME QME-1.

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Figure 7-7. Seismic isolation system qualification program development

7.7.1 Qualification principles and specification

The ISUs are most similar to the gap restraints described in ASME QME-1 QDR-4300 with the following differences:

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[[ ]](a)(4) the functional parameters applicable per ASME QME-1 QDR-4310 are:

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The functional parameters identified above are the minimum set and the Qualification Specification may identify additional ones as necessary.

The IDU is a viscoelastic damper and the functional parameters of QDR-4400 are applicable. Characteristics of the dampers is that they develop force-displacement/velocity relation during dynamic events, restraining the SSCs during seismic, operational vibration, or any other impact or impulse loads. [[

]](a)(4) Based on these characteristics the essential functional parameters of the dampers are:

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A qualification specification shall be furnished for the SIS in accordance with the requirements of ASME QME-1 Mandatory Appendix QDR-I. The qualification specification shall be provided by the Owner (or designee) or the restraint manufacturer with Owner's approval. The qualification specification provides the details of functional requirements and its minimum content shall conform to QDR-I as shown in Figure 7-8.

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Figure 7-8. Qualification specification minimum content

7.7.2 Qualification Program

Qualification of the SIS may utilize two basic methods in accordance with ASME QME-1 QDR-6100.

- The SIS may be qualified by a program of testing and analysis to become a qualified parent SIS using sub-article QDR-6200.
- The SIS may be qualified by an extension of a qualification program that has been previously performed on a similar parent restraint using sub-article QDR-6300.

If adequate previously performed qualification program is not available, the Natrium demonstration plant may utilize the parent qualification process while subsequent installations at other sites may utilize the candidate qualification process based on the qualified Natrum parent SIS. An important element of qualification approaches is the qualification by similarity (QR-7340 and QDR-A). In order to demonstrate the SIS for the full range of licensing basis events, including beyond design basis events, qualification by similarity and analysis maybe necessary due to the limitation and impracticality to test the SIS at full scale. The similarity qualification process is based on a high degree of similarity regarding the design, configuration, materials, dimensions, tolerances, surface finish, fabrication and assembly method, coating and plating, and production testing. Similarity shall be established considering the functional and other parameters in the qualification specification of the candidate unit. In all cases the similitude is established in a conservative manner to account for scaling distortions and uncertainties.

7.7.2.1 Parent Qualification

Program elements for parent qualification of the SIS is in accordance with ASME QME-1 QDR-6200. Each element applicability is briefly described in the following paragraphs.

<u>Approach to qualification:</u>

Parent qualification provides the generic qualification of the reactor SIS documented in the Application Report for its specific application. The number of units or sample set selected for qualification shall be established by the Owner accounting for uncertainties and for added conservatism. Root cause analysis of any failure shall be provided that serve as the basis for design changes.

<u>Testing:</u>

The SIS qualification plan specifies the functional parameters and environmental variables subject to testing as established in the qualification specification. [[

]](a)(4) The following elements shall be considered for testing:

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7.7.2.2 Candidate Qualification

Candidate SIS for future plant applications that are identical to the parent SIS (same manufacturer, type, size, rating, etc.) are qualified by providing an Application Report in accordance with ASME QME-1 QDR-7320 based on the parent Qualification Report.

Candidate SIS not identical in construction to the parent SIS may be qualified by extension through appropriate analysis and/or testing. The procedure for candidate SIS qualification requires a high degree of similarity to ensure that the mechanical strength, stiffness, and critical design tolerances of the candidate SIS favorably compare with the qualified parent SIS. The basis of addressing differences relies on test-verified analysis in accordance with ASME QME-1 QDR-6300. Similarity requirements, allowances for differences and the procedure requirements for test-verified analysis are provided in QDR-6320 and 6330 and will be adhered to for all candidate SIS qualification that are not identical to the parent SIS. Extension of the qualification requires that all requirements of the construction code ASME.BPVC.III.1 Subsection NF are met. In addition, the following are considered in establishing similarity of the design.

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7.7.3 Qualification Documentation Requirements

Qualification documentation is required to verify that the SIS is qualified to perform its intended functions within the environmental constraints specified. Qualification demonstrates that the service requirements are met by testing and/or analysis performed under the qualification program. Qualification documentation consists of the following:

a) Qualification Plan (QDR-7200) – translates the qualification specification requirements into a step-by-step qualification process.

b) Qualification Report (QDR-7310) – documents the qualification of the parent SIS in compliance with ASME QME-1 QDR. The qualification report shall be certified by a Registered Professional Engineer in accordance with QR-8620.

c) Application Report (QDR-7320) – document the qualification of a candidate SIS for a specific application in a nuclear facility. The application report shall be certified by a Registered Professional Engineer in accordance with QR-8630.

Additional requirements with respect to the content of qualification documentation are in QDR-7000 and shall be applicable to the SIS.

7.8 Reactor Seismic Isolation System Lifetime Management

The Reliability and Integrity Management (RIM) Program provides direction for assuring the reliability and integrity of passive components whose failure could adversely affect plant safety and reliability. The RIM Program is outlined in the ASME.BPVC.XI.2, “Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants” [4] as endorsed by RG 1.246 [25]. The RIM Program involves design interaction, performance monitoring, inspections, tests, maintenance, replacements, etc., as strategies to ensure the SSCs achieve an acceptable level of reliability to support PRA/SPRA of the plant.

The RIM program addresses the lifecycle of each component within the scope of the program. The RIM program ensures that each component performs as designed and have a reliability consistent with the assumptions used to develop the PRA for the plant.

The RIM program includes two expert panels: the RIM expert panel (RIMEP) and the monitoring and non-destructive examination expert panel (MANDEEP). The RIMEP provides the technical oversight and direction of the risk-informed aspects of the RIM Program development which consists of the following elements:

- Establishing the scope of the program
- Conducting a degradation mechanism assessment (DMA) for each component in the scope of the RIM Program

- Allocating Reliability Targets from the PRA/SPRA for each component within the scope of the program
- Establishing RIM strategies for each component within the scope of the program
- Implementing the RIM Program
- Uncertainty evaluation
- Monitoring program performance
- Updating the program

The MANDEEP develops procedures for monitoring and non-destructive examination (MANDE), including procedure, personnel, and equipment qualification requirements; developing new technologies for examination; and establishing acceptance criteria for MANDE indications identified. The overall RIM process over the life of the SSCs is illustrated in

Figure 7-9.

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Figure 7-9. Reliability and Integrity Management Program Implementation

7.8.1 Reliability and Integrity Management Program Elements

RIM Scope:

The scope of the RIM Program is determined by the RIMEP. ASME.BPVC.XI.2, RIM-2.2, “RIM Program Scope and Definition”, states that the scope shall include SSCs whose failure could adversely affect plant safety and reliability. The reactor SIS provides passive support and attenuation of seismic loads to the reactor and its classification is safety-related and seismic risk significant. In accordance with the scope definition in ASME.BPVC.XI.2 the SIS is included within the scope of the RIM program.

Degradation Mechanism Assessment:

A degradation mechanism assessment (DMA) for SSCs within the scope of the RIM Program shall be prepared in accordance with ASME.BPVC.XI.2, RIM-2.3. Mandatory Appendix VII, “Supplements for Types of Nuclear Plants,” of ASME.BPVC.XI.2 is typically used to complete this assessment. The Section of Appendix VII that addresses liquid metal-type reactors is in the course of preparation and it will focus on the degradations for components subjected to the sodium environment.

The RIMEP will develop the DMA for the reactor SIS. The DMA considers the following conditions:

- Design characteristics, including material, component type, and other attributes related to the system configuration.
- Fabrication practices, including welding and heat treatment.
- Operating and transient conditions, including temperatures, pressures, dynamic loads and service environment (humidity, radiation, etc.)
- Plant-specific, industry-wide service experience and research experience
- Results of preservice, inservice, and augmented examinations and the presence and impact of prior repairs in the system (may be provided by vendor operating experience)
- Recommendations by SSC vendors for examination, maintenance, repair, and replacement.

Once the DMA is completed for the SIS, the MANDEEP will determine what MANDE methods are applied to ensure the SSC will function with an acceptable level of reliability.

Reliability Target Allocation:

To perform the reliability target allocation for the SIS, the PRA/SPRA is reviewed in terms of scope, level of detail, and technical adequacy for use with RIM and the development of Reliability Targets. ASME.BPVC.XI.2, RIM 2.4.3, “Scope, Level of Detail, and Technical Adequacy of the PRA”, outlines the scope of the PRA/SPRA that is used to allocate Reliability Targets.

- The plant operating states relevant to the plant level risk and reliability goals and SSC-level Reliability Targets.
- A full set of initiating events including internal events and events associated with external plant hazards.

- Event sequence development that is sufficient to support the quantification of mechanistic source terms and offsite radiological consequences consistent with applicable regulatory limits on the frequencies and consequences of accident scenarios.

All plant operating modes are to be addressed; however, it is not required to have a full-scope PRA as outlined above (qualitative treatment of other risk information related to missing modes and hazard groups may be sufficient if it can be demonstrated that those risk contributions would not affect the Reliability Targets or other aspects of the RIM Program). The PRA should meet the requirements of ASME/ANS RA-S-1.4-2021, Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants [35], endorsed in Trial RG 1.247 [36], to the extent necessary to support RIM Program development.

RIM Strategy Determination:

Once the DMA for the SIS is complete and the reliability target is established, a RIM strategy is developed to address the degradation mechanisms applicable to the SIS such that the SIS will be able to function and achieve the reliability established for the SIS. The RIM strategies balance design margin and MANDE methods. Where design margin is low, increased MANDE would be expected. Where design margin is high, a lesser amount of MANDE methods would be needed. The RIM strategies shall account for all the factors that contribute to reliability, including but not necessarily limited to:

- Design strategies, including material selection
- Fabrication procedures
- Operating practices
- Preservice and inservice examinations
- Testing
- MANDE
- Maintenance, repair, and replacement practices

RIM strategies may include the use of monitoring, surveillance and/or inspections (NDE). [[

]](a)(4) The benefit of the RIM strategies is that they are developed for each component systematically as opposed to using generic prescriptive NDE.

Uncertainty Evaluation:

In accordance with ASME.BPVC.XI.2 RIM-2.6 uncertainties shall be accounted for in the development of RIM strategies. Specifically, the RIMEP shall identify additional RIM strategies over and above those determined in the normal development of strategies that are necessary to provide additional assurance that the reliability targets will be achieved and maintained during the SIS service lifetime in order to address uncertainties in predicting SIS reliability performance. These additional RIM strategies that are established to address uncertainty shall be documented in accordance with all the other RIM strategies. These strategies should clearly identify that they are intended to address uncertainty.

ASME.BPVC.XI.2, RIM-7, Glossary, has two definitions of uncertainty, one as used in PRA, and one as used in MANDE. Uncertainty as used in PRA is a representation of the confidence in the state of knowledge about the parameter values and models used in constructing the PRA. The uncertainties in PRA may be characterized by testing and/or operating experience and may utilize statistical inference analysis or similar. Uncertainty as used in MANDE is a quantification representing the variability associated with monitoring and non-destructive examination (MANDE) data and includes many technique and application specific parameters such as the minimum detection capability, sizing accuracy, resolution tolerance, repeatability, consistency, etc.

Program Implementation:

Once the RIM Program scope is established, the degradation mechanism assessment is completed, the Reliability Targets established and the RIM strategies are established, the program will shift towards the activities of the MANDEEP. [[

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After each outage during which RIM Program inspections are performed, an Owner's Activity Report (OAR) form will need to be filled out and sent to the NRC within 120 days (RG 1.246 [25]) of the outage completion date. The OAR is a record of the inspections performed in accordance with the RIM Program and results of the inspections, documentation of any repair/replacements that were made, etc. If there were analytical evaluations performed to accept any examination results that exceeded the initial acceptance criteria for a flaw, these are also to be submitted to the NRC within 120 days of the end of the outage completion date.

RIM Program Changes:

During the course of development and implementation of the RIM program changes may occur due to a variety of factors such as design maturity and changes, availability of improved MANDE methods, new operating experience, etc. In these situations, the RIM Strategy will need to be re-evaluated and alternative strategies developed to meet the reliability target for the SSC.

Certain changes are required to be reviewed by the NRC. Other changes do not require review and approval, but only notification of a review. These are outlined in RG 1.246 [25] position 4. Changes which require NRC review and approval are:

- Changes to methodologies for establishing Reliability Targets and for demonstrating RIM strategies will be of satisfying Reliability Targets.

- Alternatives to the ASME.BPVC.XI.2 as endorsed in RG 1.246.
- RIM Program changes involving alternate examination methods developed under ASME.BPVC.XI.2, Appendix A.
- Flaw evaluation criteria developed for temperatures that exceed the temperature ranges of ASME BPVC.III.1.
- Changes to the schedule for submitting OAR forms.

For any other RIM Program changes, the NRC should be notified of the changes, but NRC review and approval is not required. These notifications should be provided prior to the next scheduled refueling outage or within 3 years of making the change, whichever is less.

<u>Monitoring and Assessment:</u>

Once the bulk of the work of establishing the scope of the RIM Program, DMA, Reliability Target allocation, and RIM Strategies are identified the program transitions into the implementation phase. As inspections are completed, the data obtained from the inspections is evaluated for acceptability. [[

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Trending of results is also part of the monitoring and assessment phase of the program. The RIMEP is tasked with monitoring and assessing the RIM Program and will need to evaluate data over all the inspections and compare to the baseline established during the PSI. Evaluations are to be done to ensure the equipment will remain in an operable state until at least the next scheduled inspection.

<u>Program Updates:</u>

RIM Program updates are required periodically by ASME.BPVC.XI.2, however, the periodicity of update is generally on an as-needed basis, or it is to be updated no later than the end of each established inspection interval. ASME.BPVC.XI.2, RIM-2.8, discusses re-evaluation of the RIM program for when updates may be needed, such as new information becoming available. Changes to the SIS such as material changes, new configurations, stress changes resulting from design changes, or plant risk changes from PRA/SPRA updates could warrant a RIM Program update. Changes to plant procedures that result in different operating parameters, system line-ups, equipment and operating modes may result in different degradation mechanisms or impact the capability of MANDE. Changes in SIS performance, indicating a change in SIS reliability may warrant RIM Program updates. MANDE results that indicate service-related degradation may warrant a RIM Program update. Industry or research experience, including SIS failure or reliability data changes or new degradation mechanisms may warrant a RIM Program update. If no new information becomes available, the minimum frequency of RIM Program updates is to be at each inspection interval of the SIS. The inspection interval is to be established by the RIMEP and shall not exceed 12 years. The duration of each inspection interval should be documented in the RIM Program document.

8 CONCLUSIONS

The report herein outlines the background information, technical basis, and regulatory evaluation for developing the reactor seismic isolation system. Design specific applicability of the regulatory and industry guidance to the TerraPower Natrium Plant reactor seismic isolation is summarized based on research performed on publicly available documents for passive three-dimensional equipment SIS.

The construction and qualification methodology presents the flow-down of quality, standards, records, licensing, design, fabrication, inspection, monitoring, operations, and maintenance requirements for the reactor SIS. The framework provides a comprehensive and complete set of requirements through the entire life-cycle of the SIS that supports the fundamental safety of the plant stemming from decades of operating experience. The methodology provided presents a risk-informed, performance-based design approach for seismic isolation that is consistent with the NRC endorsed guidance of the licensing modernization project.

The methodology identifies the applicable codes and standards when the three-dimensional SIS is used for supporting the reactor for licensing basis events. Consistent with the risk-informed performance-based design approach, the reactor SIS is safety-related and seismic risk significant. Reactor support components have been licensed using the ASME.BPVC.III code for the operating fleet. Consistent with the ASME.BPVC jurisdictional boundaries and licensing precedence the Natrium reactor SIS is an ASME.BPVC.III standard support.

Furthermore, the reactor SIS is a mechanical component qualified in accordance with ASME QME-1. The SIS consists of separate ISUs and IDUs which are qualified with consideration of the specific characteristics of each. The RIM program provides direction for assuring the reliability and integrity of the passive seismic isolation system whose failure could adversely affect plant safety and reliability in accordance with ASME.BPVC.XI.2. The RIM Program involves design interaction, performance monitoring, inspection, test, maintenance, replacement, surveillance, as strategies to ensure the SIS achieves an acceptable level of reliability to support probabilistic risk assessment of the plant over its lifetime.

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10 APPENDICES

Appendix A. Seismic Isolation Technologies and Applications

10.1 Seismic Isolation Technology Overview

The general requirements of ASCE/SEI 4-16, “Seismic Analysis of Safety-Related Nuclear Structures,” ASCE 4-16, Chapter 12 [34] pertaining to SIS of safety related (SR) structures including requirements for analysis, construction as well as methods of analysis, and peer review/testing requirements are similar to those considerations and recommendations discussed in NUREG/CR-7253 [6]. Per ASCE 4-16, the following isolators are the only types assessed for use of SR nuclear structures. (1) low-damping (natural) rubber (LDR), (2) lead-rubber (natural) (LR), and (3) Friction Pendulum (FP) sliding isolators (shown in Figure 10-1). In accordance with ASCE 4-16, each has been tested extensively, can be modeled for nonlinear response-history analysis, and has been deployed in mission-critical structures. Some key characteristics of the three seismic isolation technologies are as follows:

- LDR bearings are composed of alternating layers of natural rubber and steel and can be modeled as viscoelastic components. The shear modulus of the rubber ranges between 60 psi and 120 psi. The equivalent viscous damping ratio is between 2% and 4% of critical damping.
- LR bearings are constructed similarly to LDR bearings but include a central lead core to dissipate earthquake-induced energy.
- FP bearings consists of an inner slide that slides along two (2) concave sliding surfaces with the restoring force provided by the gravity weight of the structure.

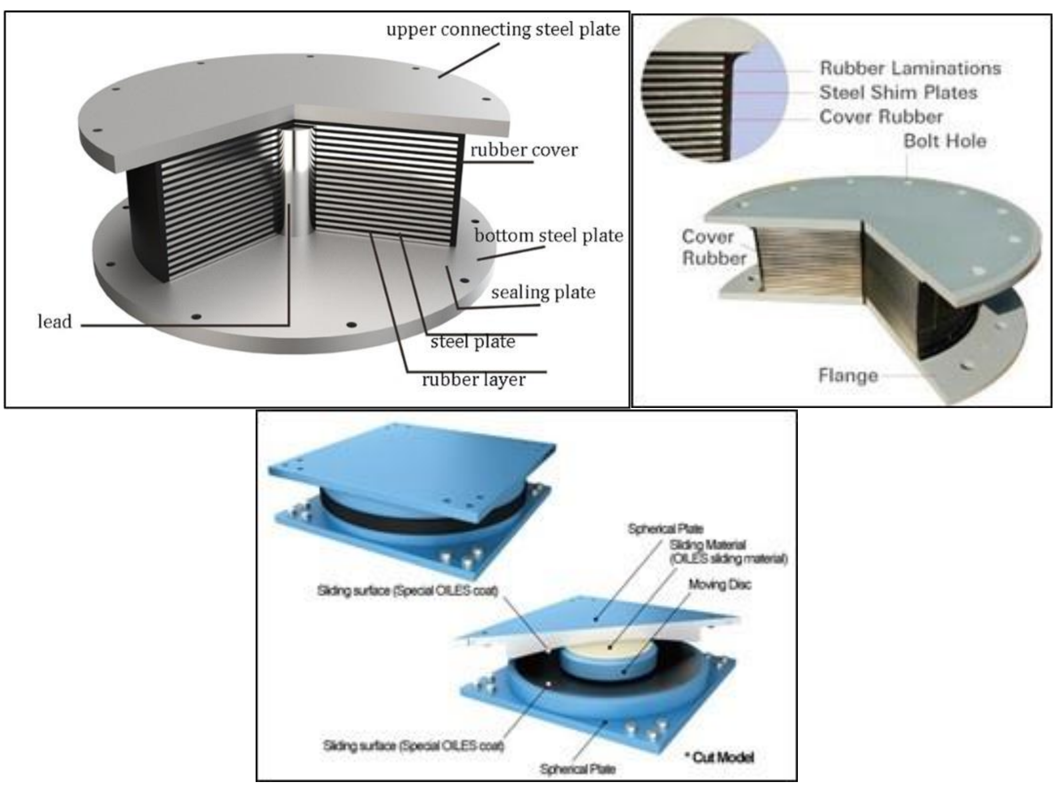

Figure 10-1: Seismic Isolation technologies addressed in regulations; a) low-damping rubber (LDR); b) lead rubber (LR); c) friction pendulum (FP) sliding.

The SIS technologies addressed by the currently available regulatory guides [6], [8], and [9] as well as ASCE 4-16, are only effective in the horizontal direction and generally require to be modeled with non-linear constitutive models as recommended by the reports. In contrast, many mission critical infrastructure in the United States as well as SR SSCs in NPPs around the world already benefit from three-dimensional SIS technology. One such technology consists of a plurality of helical spring and viscoelastic dampers (referred to as three-dimensional SIS). In Section A.1.3 of IAEA-TECDOC-1905 [37] published by the International Atomic Energy Agency (IAEA) on the topic of seismic isolation, helical spring elements are noted as the simplest rigidity element that can be used to construct SIS sub-assemblies. The report also notes that a relatively simple linear model can be used to assess the mechanical response of a structure mounted on three-dimensional SIS.

An example of three-dimensional SIS is shown in Figure 10-2 courtesy of GERB Vibration Control Systems of Germany, the manufacturer of these devices. The technology is based on helical springs which provide flexibility of similar order in all three directions and approximately velocity proportional viscoelastic dampers, also effective in all directions. It should be noted that GERB (established in 1908) is the most prominent vendor supplying three-dimensional SIS technology, and much of the publicly available information on the performance of three-dimensional SIS has been published by GERB. However, there are additional vendors that manufacture similar three-dimensional SIS devices based on helical springs and viscoelastic dampers, and the characteristics discussed herein and approach for seismic qualification for use in NPPs in the United States are expected to be valid regardless of the manufacturer.

Three-dimensional SIS technology such as shown in Figure 10-2 utilizes only passive components. The three-dimensional SIS is typically comprised of multiple assemblies of springs and dampers which are installed in parallel for redundancy. A spring unit and an integrated spring-damper unit is shown in Figure 10-2 (a) and (b), respectively. The internal design of the dampers is shown in Figure 10-3 [12] and consists of the damper housing, a non-pressurized fluid container, filled with viscous damper fluid and piston immersed in the fluid. The damper housing and the piston are attached to opposite end plates of the damper. As a results of relative movement of the piston to the housing, forces emanating from the motion of the viscous fluid provide effective load transfer and damping forces between the supporting and supported SSC.

The dampers are passive and do not require power or control signal to operate. Unlike other types of seismic restraint technologies, there are no seals separating pressurized chambers and/or valves that could fail. There are no adjustable orifices to set the operational range of the damper that needs to be calibrated and adjusted. In addition, unlike the base isolation SIS discussed in [6] [8], and [9], the dampers are only load bearing during seismic shaking. When at rest, they provide relatively easy access to the damper fluid which can be inspected, sampled, and serviced, and without a need for jacking the supported structure. Depending on the application, the dampers work with different viscous fluids. In applications where environmental conditions include radiation, the resistance of the damper fluid is an important factor in determining the appropriate chemical composition. Three types of damping fluids have been irradiated and the damper characteristic tested [28] up to 200 kGy gamma-radiation levels. Tests have demonstrated that the bituminous and polybutene based fluids remain functional to this level of radiation while the silicone oil-based fluid stiffens and its damping decreases. Radiation zones in nuclear facilities where the dampers are typically located usually remain well below these radiation levels.

The use of helical coil springs and viscoelastic dampers that provide approximately velocity proportional damping force means that the dynamic response to earthquake shaking can be modeled efficiently and with minimal uncertainty for design basis ground motions. Tests on seismic isolation units demonstrate good correlation between measurement and numerical models using ideal springs and dampers. One such model is the Double Maxwell-Model proposed in [12] and [38]. The relevant constitutive (or mathematical) model is linear, and available by default in most finite element software used in NPP design.

(a) isolation spring unit (b) integrated spring-damper unit

Figure 10-2: Three-dimensional Seismic isolation system examples. Image courtesy of GERB Vibration Control Systems of Germany.

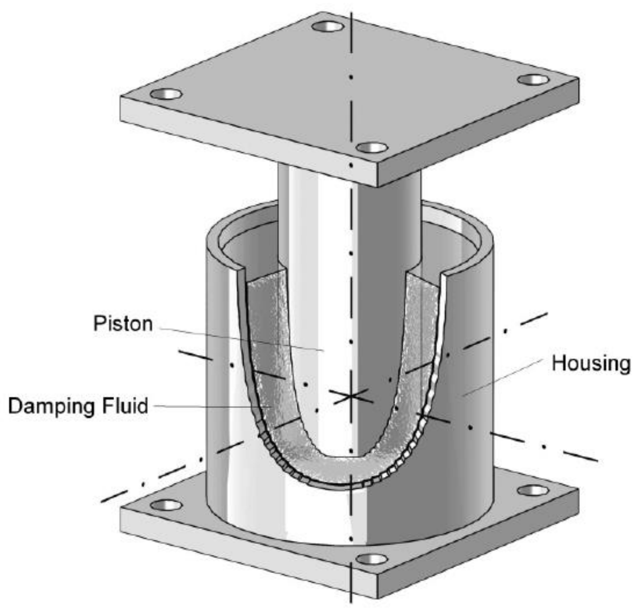

Figure 10-3: Elements of dampers. Image courtesy of GERB Vibration Control Systems of Germany.

10.2 Seismic Isolation Applications

There are six Pressurized Water Reactor (PWR) NPPs that use seismic isolation, all for superstructure horizontal isolation. All six plants were constructed in the 1980s. Four reactors are located at Cruas-Meysse NPP in France and two are located at Koeberg NPP in South Africa. Licensee application of the PRISM design planned to use a horizontal SIS system using high-damping, steel-laminated, elastomeric bearing. The NPP structure basemat is isolated from the foundation structure as shown in Figure 5-2 and Figure 10-4. The seismic isolation units are installed on concrete piers attached to the foundation structure which creates a seismic gap (moat) between the basemat and the foundation structure. The foundation structure also has a stop wall around the perimeter to prevent excessive movement of the isolated structure. The Isolators in these NPPs use neoprene elastomer bearings [6].

Due to the use of elastomeric materials and exposure to harsh environmental conditions (temperature fluctuations, and elements of nature) LDR and LR SIS require regular maintenance. The synthetic rubber (a neoprene) used in the French isolators, has stiffened significantly (37%) over time, changing the properties of the SIS. The isolator properties are monitored and changed out as necessary. The bimetallic interface used in the South African isolators is no longer considered viable for use in seismic bearings because the mechanical properties of such interfaces can change substantially with time.

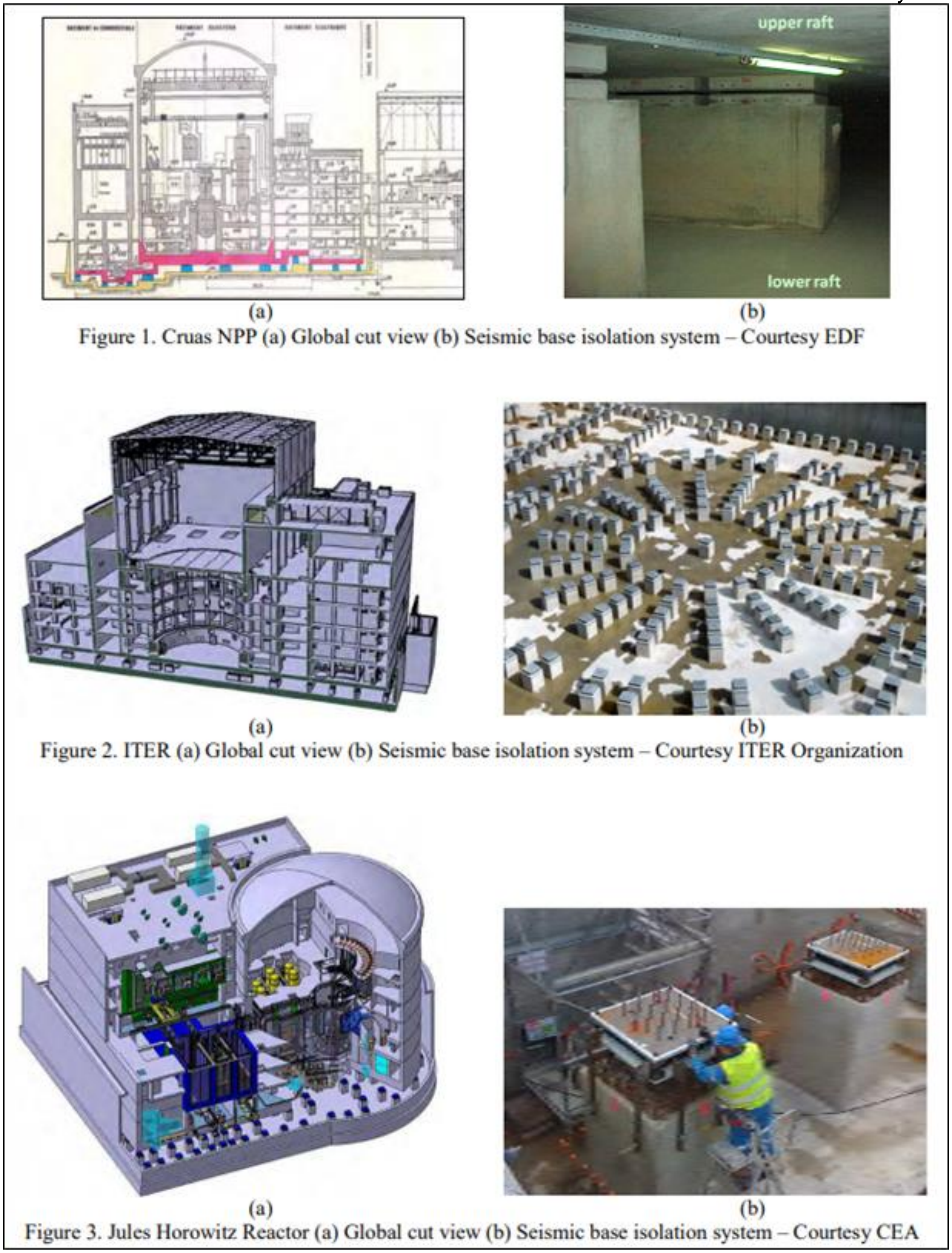

Figure 1. Cruas NPP (a) Global cut view (b) Seismic base isolation system – Courtesy EDF

Figure 2. ITER (a) Global cut view (b) Seismic base isolation system – Courtesy ITER Organization

Figure 3. Jules Horowitz Reactor (a) Global cut view (b) Seismic base isolation system – Courtesy CEA

Figure 10-4: Seismic isolation examples of nuclear facilities

Three-dimensional vibration control and seismic isolation of equipment in the power industry has numerous reference installations including nuclear facilities. GERB has provided solutions for various equipment seismic isolation as illustrated in Figure 10-5 [39]. A list of reference piping work SIS installed in NPPs is provided in Table 10-1. Another group of major power plant applications include SIS of turbine foundations as shown in the reference project list in Table 10-2. Other major reference installations in overseas and US nuclear facilities include:

- Vogtle AP1000 plants 3 and 4 turbine decks.
- Emergency diesel generators and spent fuel pool in Gösgen, Switzerland.
- Main Control Room in Olkiluoto, Finland.
- Waterford 3 piping system.
- Water-water energetic reactor (VVER) 440/213 primary loop in Mochovce NPP in Slovakia [40].
- Safety-related hot pipelines and other components including the steam generators and pressurizer in V1 in Jaslovske Bohunice VVER 440/230 type reactors [40].
- Main reactor cooling pump and steam generator seismic isolation of the VVER 1000 NPP in Temelin, Czech Republic [40].

In addition, the technology has been installed in thousands of mission-critical, commercial and infrastructure projects such as hospitals, bridges, opera houses and large commercial buildings around the world. Suppliers of the technology (such as GERB) report decades of operating experience and extremely low probability of failure. One of the earliest installations of such system in a NPP was in 1968 at the German Stade NPP [27]. After 35 years in operation, the plant was decommissioned and one of the turbine generator deck isolators was extracted and retested. Visual inspection of the springs only indicated minor paint spalling, but no corrosion of the spring elements was evident. The spring critical characteristics (spring rate, deflections and unloaded tolerances) were tested in a laboratory which indicated spring performance remained within the original specifications after 35 years of bearing load and exposed to NPP environmental conditions.

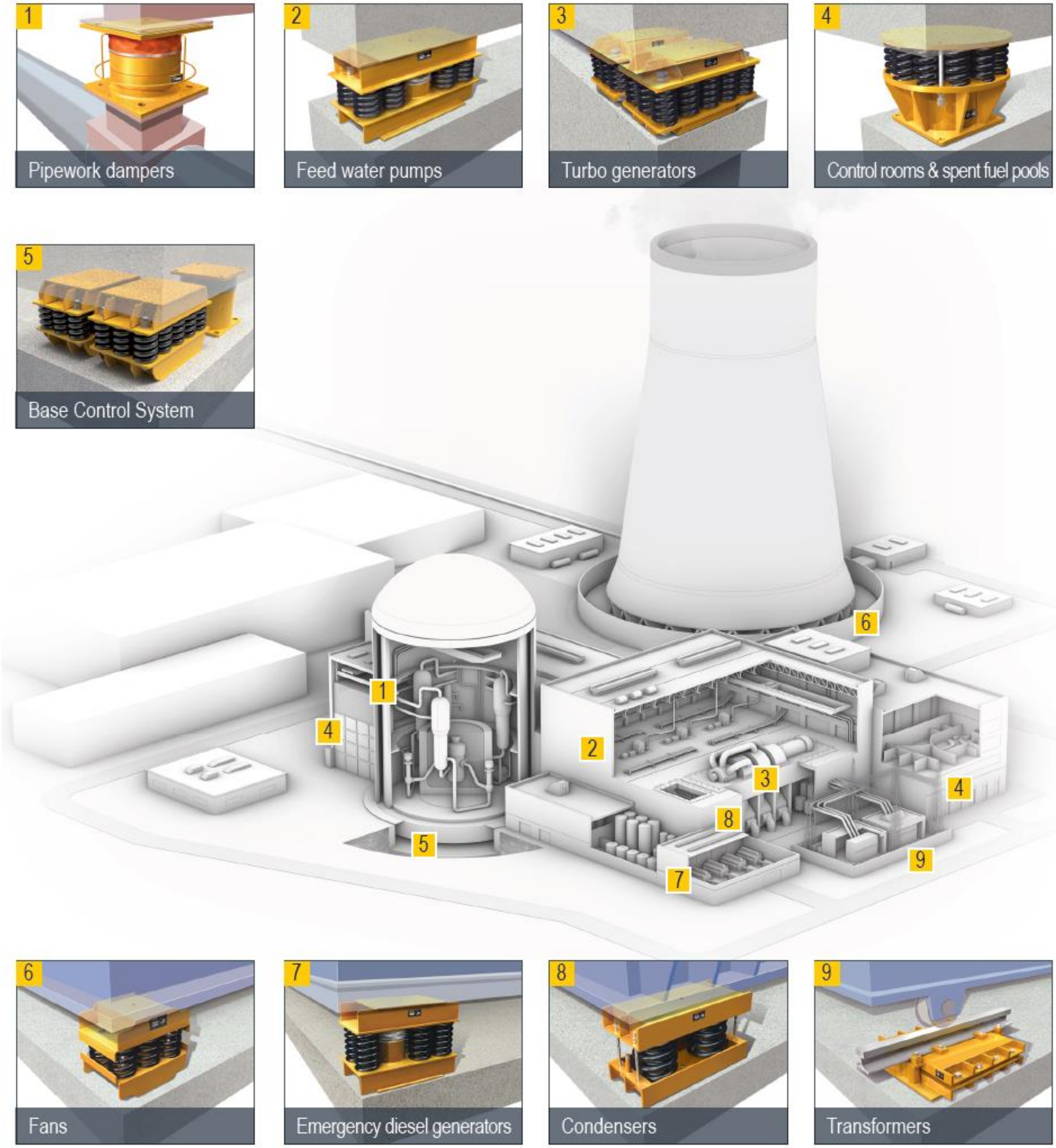

Figure 10-5. GERB seismic isolation systems in nuclear power plants and small modular reactors

Table 10-1. GERB Pipework Damping System for Nuclear Power Plants (1998-2022)

Nuclear Power Plant	Project Year	Country
Armynskaya NPP	2022	Armenia
Akkuyu NPP	2022	Turkey
Novovoronezhskaya NPP	2022	Russia
Arkansas NPP	2022	USA
Kudankulam NPP	2022	India
Armenian NPP	2021	Armenia
Akkuyu NPP	2021	Turkey
Kurskaya NPP	2021	Russia
Belorusskaya NPP	2021	Belarus
Flamanville 3 NPP	2021	France
Olkiluoto 3 NPP	2021	Germany
Ruppur NPP	2020	Russia
LAES II NPP	2020	Russia
Belorusskaya NPP	2020	Belarus
OKG NPP	2020	Sweden
Oskarshamn III NPP	2019	Sweden
Fortum NPP	2019	Finland
LAES II NPP	2019	Russia
KudanKulam NPP	2018	Russia
Olkiluoto III NPP	2018	France
Shimane II NPP	2018	Japan
Rivne NPP	2018	Czech Republic
Mochovce 1+2 NPP	2018	Slovakia
BELAES II NPP	2017	Ukraine
Olkiluoto III NPP	2017	Austria
Waterford 3 NPP	2017	USA
NVAES II NPP	2016	Russia
Oskarshamn I NPP	2016	Sweden
Tianwan III & IV NPP	2015	China
Oskarshamn 3 NPP	2015	Sweden
LAES II NPP	2015	Russia
KW Marl NPP	2015	Austria
AKRON Novgorod NPP	2015	Russia
Belorusskaja NPP	2015	Russia
Novoworoneshkaja II NPP	2015	Russia
LAES II NPP	2015	Russia
Leningradskja II NPP	2015	Russia
Chugoku NPP	2014	Japan
Novoworoneshkaja NPP	2014	Russia
Novoworoneshkaja II NPP	2014	Russia
Belojarskaja NPP	2014	Russia
Chugoku NPP	2014	Japan
Novoworoneshkaja II NPP	2014	Russia
Novoworoneshkaja I NPP	2014	Russia
Chugoku NPP	2014	Czech Republic
Belojarskaja NPP	2013	Russia
Belojarskaja NPP	2013	Russia

Nuclear Power Plant	Project Year	Country
Novoworoneshkaja NPP	2013	Ukraine
KKW Saporoshje Ukraine NPP	2013	Germany
Krasnodarskaja NPP	2013	Russia
Novoworoneshkaja NPP	2013	Ukraine
Mochovce NPP	2013	Czech Republic
PAKS NPP	2012	Germany
Mochovce NPP	2012	Czech Republic
Mezamor NPP	2012	Armenia
Cooper NPP	2012	USA
Temelin NPP	2012	Czech Republic
PAKS NPP	2012	Hungary
Temelin NPP	2011	Czech Republic
Mochovce NPP	2011	Slovakia
Kurskaja NPP	2010	Russia
Olkiluoto NPP	2009	Germany
Paks II NPP	2009	Hungary
Olkiluoto NPP	2009	Germany
Paks II NPP	2009	Hungary
Shearon’s Harris NPP	2009	USA
Paks IV NPP	2009	Hungary
Paks I NPP	2009	Hungary
Brunsbüttel NPP	2009	Germany
Mezamor NPP	2008	Czech Republic
Isar NPP	2008	Germany
Olkiluoto NPP	2008	Germany
Oskarshamn NPP	2008	Sweden
Paks NPP	2008	Hungary
Oskarshamn NPP	2007	Sweden
Tianwan NPP	2007	China
Krsko NPP	2007	Slovenia
Gösgen NPP	2007	Germany
Paks NPP	2006	Hungary
Temelin NPP	2006	Czech Republic
Bohunice NPP	2006	Slovakia
Paks NPP	2006	Hungary
Cernavoda II NPP	2005	Romania
Bohunice NPP	2005	Slovakia
Angra NPP	2005	Germany
Temelin NPP	2004	Czech Republic
Oskarshamn NPP	2004	Sweden
Grafenrheinfeld NPP	2004	Germany
Brunsbüttel NPP	2003	Germany
Tianvan NPP	2002	Rusia
Loviisa NPP	2002	Finland
Forsmark NPP	2002	Sweden
Temelin NPP	2002	Czech Republic
Temelin NPP	2001	Czech Republic
Bohunice NPP	2001	Slovenia

Nuclear Power Plant	Project Year	Country
Brunsbüttel NPP	2001	Germany
Loviisa NPP	2001	Finland
Brunsbüttel NPP	2000	Germany
Paks III - IV, NPP	2000	Hungary
Angra NPP	2000	Germany
Cernavoda NPP	2000	Romania
Angra NPP	1999	Germany
Loviisa NPP	1999	Finland
Mochovce NPP	1999	Slovakia
Paks I + II NPP	1998	Hungary

Table 10-2. GERB Pipework Damping System for Nuclear Power Plants (1998-2022)

Country	Power Plant	Manufacturer	Capacity (MW)	Delivery Date
Argentina	Atucha	KWU	745	1981
Australia	Northern Power Station	Mitsubishi	2 x 275	1979
	Loy Yang 'A"	KWU	3 x 500	1980
	Loy Yang 'A'	BBC	500	1980
	Loy Yang 'B'	Hitachi	500	1988
	Tarong	Hitachi	4 x 350	1983
	Callide	Hitachi	2 x 350	1983
	Collie	ABB Baden	300	1983
	Stanwell	Hitachi	4 x 350	1984
Austria	Tullnerfeld	KWU	730	1976
	Voitsberg	MAN	300	
	Dürnrohr	KWU	405	1982
	Riedersbach	Franco Tosi	160	1982
	Mellach Süd	BBC Baden	275	
	Liebenfels	MAN		2017
Bangladesh	Ashugong	BBC	2 x 150	
	Ruppur		1200	2018
Belgium	Drogenbos	Alsthom	172	1993
	Seraing	Alsthom	172	1994
	Brügge	GEC Alsthom	172	1996
Brazil	Angra	KWU	2 x 1300	1979
	TKS CSA Rio de Janeiro	Alstom Baden		2007
China	Beijing	ABB	2 x 300	1997/8
	Hefei	ABB	2 x 350	1993
	Tianwan 1 + 2	LMZ	2 x 1000	2001/2002
	Dabieshan	Alstom (Beizhong)	2 x 600	2007
	Pingliang	Alstom (Beizhong)	2 x 600	2008
	Ling'ao	Alstom (DongFang)	2 x 1000	2008
	Fangjiashan	Alstom (DongFang)	2 x 1000	2009/10
	Hongyanhe	Alstom (DongFang)	4 x 1000	2009-2011
	Shentou	Alstom (Beizhong)	2 x 660	2010
	Ningde	Alstom (DongFang)	4 x 1000	2010-2012
	FuQing	Alstom (DongFang)	4 x 1000	2010-2013
	Yangjiang	Siemens	6 x 1000	2010-2014
	Taishan 1 + 2	Alstom	2 x 1600	2011
	FangChengGang	Siemens	2 x 1000	2011/12
	Kashi	Shanghai Steam Turb.	2 x 350	2012
	Tangshan	Harbin Steam Turbine	2 x 350	2013
	Taizhou	Shanghai Steam Turbine	2 x 1000	2013
	Laiwu	Shanghai Steam Turbine	2 x 1000	2013

Country	Power Plant	Manufacturer	Capacity (MW)	Delivery Date
China	Hami	Alstom (Beizhong)	2 x 660	2013
	Tianwan 3 + 4	Harbin Turbine (MHI)	2 x 1250	2014
	Wucaiwan	Alstom	2 x 660	2015
	Dabieshan	Alstom	2 x 660	2015
	Laizhou	Shanghai Turbines	2 x 1000	2016
	Yuxian	Shanghai Turbines	2 x 1000	2017
	Yuncheng	Dongfang	2 x 1000	2018
	Changjiang	Harbin	2 x 1000	2019
	Shidongkou	Shanghai Turbines	2 x 700	2020
Côte d'Ivoire	Ciprel IV	GE	110	2014
Croatia	Zagreb	Doosan Skoda	40	2018
Czech Republic	Prunerov	Skoda		1989
	Olomouc	Foster Wheeler	40	1997
	Trebovice	Skoda	2 x 70	1998/2004
	Ledvice	Skoda	660	2010
Denmark	Asnaesvaerket	BBC Baden	600	1979
	Nefo 2	KWU	250	1976
	Sonderjyllands	KWU	630	1979
	Studstrup	BBC Baden	2 x 375	1981
	Fynsvaerket Bl. 7	Siemens	410	1989
	Oersted	Skoda	90	
	Vestkraft	ABB	400	
	Nordjyllandsvaerket	MAN Energie	414	1994
	Skaerbaekvaerket	MAN Energie	414	1994
	Avedore	Ansaldo	450	1999
	Lisbjerg Biomass	Doosan Skoda	39	2015
	Kalundborg	Doosan Skoda	25	2018
Estonia	Narva	Alstom	300	2012
Finland	Kainuun Voima	Siemens-KWU	84.6	1989
	Meri Pori	ABB Stal	580	1991
	Enso Gutzeit	MAN Energie	91.4	1991
	Oulu	LMZ	150	1992
	Oulu	Siemens-KWU	110	1995
	Alcholma	LMZ	250	1999
	Kymin Paper	Alstom	85	2001
	Olkiluoto	Siemens-KWU	1600	2005
	Tornio (Steel Plant)	Skoda	60	2006
	Juvaskyla	LMZ	200	2008
	Haapaniemi	Skoda	46	2010
	Raahe	LMZ	125	2014
	Hanhikivi	GE	1200	2018

Country	Power Plant	Manufacturer	Capacity (MW)	Delivery Date
Finland	Salo	MAN		2019
France	Saint-Laurent-des-Eaux	Alsthom	2 x 900	1979
	Paluel	Alsthom	2 x 1300	1979
	Flamanville	Alsthom	2 x 1300	1980
	Golfech	Alsthom	2 x 1300	1980
	Chinon	Alsthom	4 x 900	1981
	Belleville	Alsthom	2 x 1300	1981
	Chooz	Alsthom	2 x 1500	1984
	Civaux	Alsthom	2 x 1500	1988
	Flamanville III	Alstom	1600	2007
	Dunkerque	M+M		2018
Germany	Stade	KWU	630	1968
	Biblis A	KWU	1145	1972
	Biblis B	KWU	1240	1972
	Neckarwestheim	KWU	775	1972
	Lausward	BBC	300	1973
	Gersteinwerk	BBC	4 x 400	1974
	Kalkar	KWU	300	1974
	Schmehausen	BBC	300	1974
	GKN 2	KWU	1300	1974
	Krümmel	KWU	1260	1975
	Brunsbüttel	KWU	770	1976
	Philipsburg	KWU	864	1976
	Isar	KWU	870	1976
	Mülheim-Kärlich	BBC	1300	1976
	Emsland	BBC	2 x 400	1976
	Grohnde	KWU	1300	1977
	Isar 2	KWU	1300	1977
	Staudinger	BBC	600	1978
	Reuter	BBC	190	1978
	Heyden	KWU	740	
	Hamborn	BBC	108	1978
	Stadtwerke Duisburg	BBC	140	1978
	Mehrum	KWU	660	1978
	Voerde A + B	BBC	650	1979
	Philipsburg 2	KWU	1300	1979
	Unterweser	KWU	1230	1980
	Wilhelmshaven	KWU	720	1980
	Weiher III	BBC	650	1980
	Walsum	KWU	410	1980
	Heilbronn	BBC	720	

Country	Power Plant	Manufacturer	Capacity (MW)	Delivery Date
Germany	Bexbach	BBC	750	1981
	Bergkamen A	KWU	750	1982
	Gersteinwerk	KWU	750	
	Völklingen	KWU	195	1982
	GKM	BBC	2 x 200	
	GKM	BBC	84	
	Ibbenbüren	KWU	740	1982
	Völklingen	Siemens-KWU	210	1982
	Leiningerwerk	BBC	450	1983
	Elverlingsen	BBC	315	1983
	Grafenrheinfeld	KWU	1300	1983
	Gundremmingen	KWU	2 x 1300	1983
	Brokdorf	KWU	1360	1983
	Rheinhafen	BBC	550	1983
	Hastedt	BBC	130	
	VW/CHP West	KWU	2 x 130	
	Stadtwerke Duisburg	KWU	95.8	
	GKW Hannover	KWU	2 x 145	
	Herne IV	KWU	500	1983
	Moabit	Siemens KWU	110	1988
	Staudinger 5	ABB	600	1990
	Tiefstack	MAN Energie	160	1990
	Rostock	ABB	500	1992
	Dresden Nossener Brücke	Siemens-KWU	72,5	1993
	Schkopau	Siemens-KWU	110	1993
	Schkopau	ABB	2 x 450	1993
	Schwarze Pumpe	Siemens-KWU	2 x 800	1994
	Altbach	Siemens-KWU	345	1995
	Lippendorf A + B	ABB	2 x 900	1996
	Boxberg	Siemens-KWU	900	1996
	VW Nord	Siemens-KWU		1998
	Niederaußem	Siemens-KWU	1000	1999
	München Süd	Alstom	100	2003
	Neurath	Alstom	2 x 1100	2007
	Walsum	Hitachi	750	2007
	Datteln	Alstom	2 x 900	2007
	Boxberg	Alstom	900	2007
	Salzgitter	Siemens		2007
	Hoechst	Skoda	86	2008
	Moorburg	Alstom	2 x 900	2009
	Wilhelmshaven	Siemens-KWU	2 x 145	2009

Country	Power Plant	Manufacturer	Capacity (MW)	Delivery Date
Germany	Lünen	Siemens	900	2009
	RDK Karlsruhe	Alstom	900	2009
	Wilhelmshaven	Hitachi	750	2009
	GKM9	Alstom	900	2010
	Holzkirchen	Turboden		2017
	Kassel	M+M		2020
Great Britain	Templeborough	Doosan Skoda	45	2015
	Dunbar	Doosan Skoda	38	2015
	Margam	Doosan Skoda	45	2015
	Gloucester	MAN		2016
	Port Talbot	MAN		2018
Greece	Megalopolis	KWU	300	1970
	Komotini	Ansaldo	175	1999
Hungary	Csepel II	GE		1998
India	Trombay 6	Siemens	500	1987
	Anta	BBC	150	1988
	Dadri	BHEL	4 x 210	1989
	Talcher	ABB	2 x 500	1991
	Dahanu	BHEL	2 x 250	1991
	Gandhar	ABB	210	1992
	Suratgarh	BHEL	5 x 250	1994
	Chandrapur	BHEL	500	1994
	Khaparkheda	BHEL	2 x 210	1994
	Unchahar	BHEL	3 x 210	1996
	Vindhyachal	BHEL	4 x 500	1996
	Kayamkulam	BHEL	350	1997
	Soda Ash	Shin Nippon	2 x 16.34	1998
	Simhadri	BHEL	2 x 500	1999
	Neyveli	Alstom	250	2000
	Talcher	BHEL	4 x 500	2001
	Akrimota	Ansaldo	2 x 125	2001
	Rihand	BHEL	2 x 500	2002
	Ramagundam	BHEL	500	2002
	Kota	BHEL	195	2002
	Panipat	BHEL	2 x 250	2002
	Paricha	BHEL	2 x 210	2003
	Bellary	BHEL	500	2004
	Rayalaseema	BHEL	2 x 210	2004
	Birsingpur	BHEL	500	2004
	Kahalgaon	BHEL	3 x 500	2004
	New Parli	BHEL	250	2004

Country	Power Plant	Manufacturer	Capacity (MW)	Delivery Date
India	Korba East	BHEL	2 x 250	2004
	Sipat	BHEL	2 x 500	2004
	Sipat	LMZ	3 x 660	2004
	Kundankulam	LMZ	2 x 1000	2004
	Barh	LMZ	3 x 660	2004
	Giral Lignite	BHEL	125	2005
	Lehra Mohabbat	BHEL	2 x 250	2005
	Paras	BHEL	250	2005
	Santaldih	BHEL	250	2005
	Amarkantak	BHEL	210	2005
	Vijayawada	BHEL	500	2006
	Bhoopalapally	BHEL	500	2006
	Neyveli	BHEL	2 x 250	2006
	Tata Power	BHEL	120	2007
	New Parli	BHEL	250	2007
	Paras	BHEL	250	2007
	Dadri	BHEL	2 x 490	2007
	Paricha	BHEL	2 x 250	2007
	Farakka	BHEL	500	2007
	Korba	BHEL	500	2007
	Tau Devi Lal	BHEL	2 x 250	2007
	Mejia ‘B’	BHEL	2 x 500	2007
	Koderma	BHEL	2 x 500	2007
	Durgapur Steel	BHEL	2 x 500	2007
	Kothagudem	BHEL	1 x 500	2007
	Giral Lignite	BHEL	1 x 125	2007
	Harduaganj	BHEL	2 x 250	2008
	Simhadri-II	BHEL	2 x 500	2008
	Aravali	BHEL	3 x 500	2008
	Ennore	BHEL	2 x 500	2008
	Khaperkheda	BHEL	1 x 500	2008
	Bhusawal	BHEL	2 x 500	2008
	Rayalseema	BHEL	1 x 210	2008
	Santaldhi	BHEL	1 x 250	2008
	Bellary	BHEL	1 x 500	2008
	Raichur	BHEL	1 x 250	2008
	Mauda	BHEL	2 x 500	2009
	Anpara	BHEL	2 x 500	2009
	Ukai	BHEL	1 x 490	2009
	Bongaigaon	BHEL	3 x 250	2009
	Rihand	BHEL	2 x 500	2009

Country	Power Plant	Manufacturer	Capacity (MW)	Delivery Date
India	Vindhyachal	BHEL	2 x 500	2009
	Marwa	BHEL	2 x 500	2009
	Korba West Extn.	BHEL	1 x 500	2009
	Barh-II	BHEL	2 x 660	2009
	Avantha	BHEL	600	2010
	Tuticorin	BHEL	2 x 500	2010
	New Parli	BHEL	250	2010
	Chandrapur	BHEL	2 x 500	2010
	Bokaro	BHEL	500	2010
	Vallur-I	BHEL	500	2010
	Sagardighi	BHEL	2 x 500	2011
	Bhavnagar	BHEL	2 x 250	2011
	Nabinagar	BHEL	4 x 250	2011
	Jhabua	BHEL	600	2011
	Muzaffarpur	BHEL	2 x 195	2011
	Meja	Toshiba	2 x 660	2012
	Solapur	Alstom	2 x 660	2012
	Singareni	BHEL	2 x 600	2013
	Kudgi	Toshiba	3 x 800	2013
	Mauda	BHEL	2 x 660	2013
	Vindhyachal	BHEL	500	2013
	Raghunathpur	BHEL	2 x 660	2014
	Nabinagar	Alstom	2 x 660	2014
	Lara	Hitachi	2 x 800	2014
	Gadarwara	BHEL	2 x 800	2014
	Unchahar	BHEL	500	2014
	Neyveli	BHEL	2 x 500	2014
	Nabinagar	Alstom	3 x 660	2015
	Darlipali	Toshiba	2 x 800	2015
	Tanda	Alstom	2 x 660	2015
	Ennore	Dongfang	660	2015
	Raigarh	BHEL	600	2016
	Sagardighi	BHEL	500	2016
	North Chennai	BHEL	800	2017
	Uppur	BHEL	2 x 800	2018
	Godda	GE China	800	2018
	Kudankulam	Toshiba	3 x 1000	2018
Indonesia	Medan	Alsthom	2 x 65	1981
	Cilacap	Shanghai	1 x 1000	2017
Ireland	Moneypoint	BBC Baden	4 x 350	1983
	Lough Ree	Fuji	100	2003

Country	Power Plant	Manufacturer	Capacity (MW)	Delivery Date
Ireland	West Offaly	Fuji	150	2003
Israel	Rutenberg	ABB	2 x 550	1996
Italy	Sermide	Franco Tosi	350	1980
	Tavazzano	Franco Tosi	2 x 320	1980
	Fiume Santo	Ansaldo	2 x 320	1987
	Pietrafitta	Ansaldo	2 x 75	1992
	Sulcis	Ansaldo	320	2004
	Modugno	Alstom	250	2007
Kazakhstan	Pawlodar		120	2018
Korea	Ulsan	BBC Baden	3 x 400	1978
	Seoul City	Doosan	2 x 400	2014
Kuwait	Az Zour	Toshiba	8 x 300	1992
	Doha West	Mitsubishi	8 x 300	
	Sabiya	Mitsubishi	8 x 300	1989
	Shuaiba North	Toshiba	216	2008
Malaysia	Port Kelang	Mitsubishi	2 x 300	1986
	Port Kelang	GE	500	
	Saba Shipyard		100	1994
	Mentakab	ABB Turbinen		1997
	Manjung 4	Alstom	1000	2012
	Tanjung Bin 4	Alstom	1000	2013
	Manjung 5	Hitachi	1000	2014
Morocco	Jorf Lasfar	Mitsubishi	2 x 350	2011
Netherlands	Hemweg 7	BBC	500	1975
	Amercentrale	BBC	600	1980
	Centrale Velsen	KWU	360	1982
	Amer 9	ABB		1989
	Hemweg 8	ABB	500	1990
	Moerdijk	Siemens-KWU	150	1994
	Rotterdam (Maasvlakte)	Hitachi	750	2009
	Rotterdam (Maasvlakte)	Alstom	900	2010
Nigeria	Bouygues	Hitachi	6 x 250	1982
Northern Ireland	Coolkeeragh	Alstom	144	2003
Pakistan	Lucky Electric	GE China	660	2018
Poland	Zeran	Siemens Industrial Brno	100	2005
	EC Siekierki	Siemens	110	2008
	Czestochowa	Alstom	65	2008
	Konin		55	2011
	EC Zofiowka	Siemens	75	2015
	Rybnik	Alstom	900	2012

Country	Power Plant	Manufacturer	Capacity (MW)	Delivery Date
Poland	Opole	Alstom	2 x 900	2015
	Kozienice	Mitsubishi Hitachi	1000	2015
Portugal	Sines	BBC Baden	4 x 300	1985
	Pego	ABB Baden	2 x 300	1989/1992
Romania	Bucuresti Vest	Siemens		2006
Russia	TEZ St. Petersburg North-West	LMZ	2 x 150	1995
	TEZ-2 Saransk	LMZ	65	1998
	Juschno-Kusbasskaja GRES	LMZ	115	2001
	Tjumen	ABB	50	2003
	TEZ Süd-West St. Petersburg	LMZ	2 x150	2000/06
Russia	TEZ-27 Moskau	LMZ	2 x 150	2007
	TEZ-21 Moskau	LMZ	150	2007
	TEZ-22 St. Petersburg-Süd	LMZ	150	2008
	TEZ-2 Kalingradskaja	LMZ	2 x 150	2009
	TEZ-5 Prawobereschnaja	LMZ	150	2009
	TEZ-6 Perm	Siemens	30	2010
	Nowourengoiskaja GRES	LMZ	1 x 150	2010
	Nowourengoiskij GHK	SNM	40	2011
	Belojarskaja	LMZ	800	2012
	Leningradskaja	LMZ	2 x 1200	2012
	Bratsk	KTZ	2 x 30	2013
	Nowoworeneschskaja	LMZ	2 x 1200	2011/13
	TEZ-5 Ufimskaja	LMZ	2 x 80	2010/15
	Jaroslavskaja	LMZ	1 x 150	2014
	Belorusskaja	LMZ	2 x 1200	2015
	Zatonskaja TEZ	Doosan Skoda	60	2015
Saudi Arabia	Shoaiba 1 – 3	ABB	5 x 393	1999
	Shoaiba 4 – 5	Alstom	2 x 390	2001
	Shoaiba 9 - 11	Alstom	3 x 390	2004
	Shoaiba 12 - 14	Alstom	3 x 390	2008
	Shuqaiq	Alstom	4 x 720	2014
	Yanbu 3	Alstom	5 x 620	2014
Singapore	Tuas South	MHI	2 x 66	1998
Slovenia	Sostanj IV	KWU	335	1978
Spain	Trillo	KWU	1300	1982
	Pontevedra	M+M		2018
Sri Lanka	Kerawalapitiya	GE	100	
Sweden	Nyköping	ABB STAL	80	1993
	Malmö	Hitachi Power	161	2007
	Värtaverket	Doosan Skoda	154	2013

Country	Power Plant	Manufacturer	Capacity (MW)	Delivery Date
Sweden	Södra Cell VÄRÖ	Doosan Skoda	55	2015
	Sveg	M+M		2016
Switzerland	Gösgen	KWU	920	1974
	Leibstadt	BBC	900	1974
	Genf	Fincantieri	12	2018
Taiwan	Taitung	Alstom Brno		2001
Thailand	Khanom	Alsthom	75	1988
Turkey	Baymina	Alstom	320	2002
	Karabiga	Alstom Beijing	2 x 660	2015
	Hunutlu	Shanghai Turbines	2 x 660	2016
UAE	Deep	ABB		1992
	Jebel Ali	ABB		1994
	Jebel Ali	ALSTOM	2 x 120	2006
	Jebel Ali L	ALSTOM	221	2008
	Jebel Ali M	ALSTOM	3 x 221	2008
Ukraine	Smart Energy	M+M		2020
Uruguay	ENCE/Montes del Plata	Siemens	2 x 90	2011
USA	El Centro/CA	Siemens	60	2011
	Vogtle	Toshiba	2 x 1000	2011/12
	Virgil C. Summer	Toshiba	2 x 1000	2012/13
Venezuela	Cadafe	KWU	2 x 400	1976
	Cadafe	BBC	2 x 400	1982

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