U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES

REVIEW RESPONSIBILITIES

Primary - Organization responsible for structural analysis reviews

Secondary - Organization responsible for reviews of material and coolability of the fuel assembly (Appendix D to this DSRS section)

I. AREAS OF REVIEW

This section describes the review of areas relating to all seismic Category I structures, other than the containment, and other structures important to safety that may not be classified as seismic Category I. The review of the foundations for seismic Category I structures is performed under Design Specific Review Standard (DSRS) Section 3.8.5.

The NuScale light-water small modular reactor (SMR) is a small-size reactor inside a tightly conforming containment (this combination is called a NuScale power module (NPM)) in a below-grade water-filled reactor pool in the reactor building with up to 12 total modules. A refueling bay and the spent fuel pool are also part of the reactor building pool within the reactor building structure. The NPMs are moved to the refueling bay for inspection, refueling and transfer of spent fuel to the spent fuel pool for storage. The reactor building encloses the reactor module pool, the refueling bay, and the spent fuel pool. The pools are below grade. The turbine generator buildings sit above grade on either side of the reactor building. The control building, with the control room below grade, sits next to one end of the reactor building. The radioactive waste building sits adjacent to the reactor building at the end opposite from the control building.

The specific areas of review are as follows:

1. Description of the Structures. The staff reviews the descriptive information, including plans and sections of each structure, to establish that there is sufficient information to define the primary structural aspects and elements relied upon for the structure to perform the intended safety function. The staff also reviews the relationship between adjacent structures, including the separation provided or structural ties, if any. The following describes the major plant structures that are reviewed and the descriptive information reviewed for each:

   A. Reactor Building

   The reactor building (RB) houses the systems and components required for plant operation and shutdown, and it provides protection and access for service to the

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NPMs for activities such as fuel loading and unloading. A substantial portion of the Reactor Building is below grade with a portion of the structure including the roof, side walls and crane above grade. The Reactor Building houses all of the NPMs, spent fuel pool, fuel handling areas, remote shutdown station, and safety-related, important to safety systems and components, and other non-safety plant components.

The RB is structure constructed of reinforced concrete. The staff reviews the general arrangement of the structural walls, columns, floors, roof, and any removable sections such as the biological shield which provides cover over each NPM.

The staff's review identifies the types of concrete and steel structures associated with the RB and examines their structural and functional characteristics. Any special features of the RB, including the reactor pool and modular construction, should be included in the description of the structures.

A unique feature of the NuScale plant is the capability to house up to 12 NPMs in the reactor pool. Therefore, the structures within the RB that are shared among the multiple NPMs are reviewed to understand the arrangement of the NPMs over the life of the plant. This includes the number and locations of the NPMs that will initially be placed, the number and locations of NPMs that will be incrementally added over the life of the plant, and the possible rearrangement of the NPMs. This information is important in order to determine if the design of the reactor pool and other shared structures in the RB is acceptable.

The geometry of these structures is reviewed, including sketches showing plan views at various elevations and sections in at least two orthogonal directions. The reviewer examines the arrangement of the RB and the relationship and interaction of the RB with its surrounding structures and with its interior compartment walls and floors to determine the effect that these structures could have on the design boundary conditions and expected structural behavior of the RB when subjected to design loads.

The review encompasses general information related to the RB, including special features such as sump and drain areas, seismic gaps between the RB and adjacent building/structural elements (internal and exterior), subfoundation drainage system (if applicable), use of waterproofing membrane, and RB settlement monitoring systems.

B. Reactor Pool

The reactor pool is a part of a large reinforced concrete pool inside the RB which is located below the ground level. The reactor pool houses and supports up to 12 NPMs. The reactor pool contains water to provide cooling for the NPMs and is, therefore, lined with stainless steel. The surface of the reactor pool water is a few feet below the ground elevation. The NPMs are installed in two rows of up to six NPMs in each row. Concrete walls separate each NPM in an individual reactor bay with one side open to permit moving the NPM into and out of each reactor bay. A concrete cover is located in each bay to provide biological shielding. Between the two rows of NPMs, a central channel is provided to allow movement of the NPMs between the reactor pool and the adjacent refueling pool. The staff reviews the general arrangement of the structural walls, floors, supports, and any removable sections. As discussed above for the RB, the configuration of the reactor pool is also reviewed to understand the arrangement of the NPMs over the life of the plant.

C. Fuel Handling and Reactor Maintenance Areas

Areas in the RB for fuel handling and reactor maintenance include several pools and space for new fuel storage. The pool inside the RB includes the following areas: the spent fuel pool, refueling pool, reactor pool, and the dry dock. The refueling pool is contiguous with the reactor pool, which permits the transfer of the NPM while under the pool water. Between the refueling pool and the spent fuel pool is an open channel that provides access for fuel assembly transfer under water during the refueling process. The dry dock area contains the module inspection rack and has a gate which separates it from the refueling pool. When the gate is in the closed position, the water level in the dry dock can be lowered to perform maintenance activities on the NPM. The new fuel storage area includes the fuel receiving area, new fuel storage racks, and a jib crane for handling fuel assemblies. The structures described above are also constructed of reinforced concrete and may contain structural steel and/or utilize modular construction technology. In addition to the information reviewed for the RB, the staff will evaluate the general arrangement of the structures associated with fuel handling and reactor maintenance areas, including their walls, floors, and steel liners.

D. Control Building

The main control room (MCR), central alarm statin (CAS), technical support center (TSC), tunnel to reactor building (underground connection by airlock adjacent to MCR) and control room habitability systems (at floor below the MCR) are located in the control building. The staff reviews this building as a separate structure. This building is made of reinforced concrete, with a steel weather enclosure over the roof. The staff reviews the general arrangement of the structural walls, columns, floors, roof, and any removable sections.

E. Other CSDRS/SSE Seismically Designed Structures and Systems

Other structures that do not contain safety-related systems or components, but may be important to safety because of other design provisions, should be described. These structures are usually made either of reinforced concrete or structural steel, or a combination of the two. The descriptive information reviewed for such structures is similar to that reviewed for the RB.

Distribution systems, including their supports (e.g., cable trays, conduit, HVAC, and piping), and equipment supports are reviewed in accordance with NUREG-0800 Standard Review Plan (SRP) Sections 3.9.2 and 3.9.3, and DSRS Section 3.7.3. Intervening structural elements between these distribution systems and equipment supports and building structural steel/concrete (e.g., steel platforms, building frame members, embedment plates, and building steel members beyond the jurisdictional boundary of supports to mechanical components) are reviewed in this DSRS section.

Further, the reviewer may encounter site-specific special structures that are not located in the immediate vicinity of the site. When the failure of any such structure could affect the safety of the plant, it should be designed to withstand the effects of an SSE, and the surface faulting should be comparable to that of the nuclear plant itself. Examples of such structures include emergency cooling water tunnels, embankments, concrete dams, and water wells. These structures are reviewed on a case-by-case basis, and safety assessments should consider the material underlying the structure and its location with respect to the site. The staff will review the descriptive information provided to ascertain the structural behavior of such structures, particularly with respect to seismic events and plant process conditions during which they are required to remain functional.

F. Masonry Walls

If used, these are walls, partitions, or radiation shields which are components of the structures listed above. They are constructed of concrete masonry units bonded with mortar in single or multiple widths and may be reinforced horizontally as well as vertically. Masonry walls without reinforcement should not be used to support seismic Category I SSCs nor in areas that contain seismic Category I SSCs. The staff will review the arrangement and configuration of these walls.

G. Modular Construction

The NuScale design may use modular construction methods for major seismic Category I structures. Wall modules are typically constructed from large prefabricated sections of steel plates spaced apart with intermittent steel members, joined with other modules at the site, and then filled with concrete. The concrete fill used in wall modules could be structural concrete with reinforcement, fill concrete of low strength and no reinforcement, or heavy concrete for radiation shielding. Floor modules could consist of prefabricated steel members and plates which are combined with poured concrete to create a composite section. The structural design of modules, fabrication, configuration, layout, and connections are reviewed on a case-by-case basis.

2. <u>Applicable Codes, Standards, and Specifications</u>. The information pertaining to design codes, standards, specifications, regulatory guides (RGs), and other industry standards that are applied in the design, fabrication, construction, testing, and surveillance of seismic Category I structures is reviewed.

3. Loads and Loading Combinations. The review encompasses information pertaining to the applicable design loads and various load combinations thereof. The loads normally applicable to seismic Category I structures include the following:

A. Those loads encountered during construction of the seismic Category I structures which include dead loads, live loads, prestress loads (if applicable), temperature, wind, earth pressure, snow, rain, and ice, and construction loads that may be applicable such as material loads, personnel and equipment loads, horizontal and vertical construction loads, loads that are induced by the construction sequence and by the differential settlements of the soil under and to the sides of the structures, erection and fitting forces, equipment reactions, and form pressure.

B. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads resulting from operating temperature, and hydrostatic loads such as those in reactor and spent fuel pools.

C. Those loads to be sustained during severe environmental conditions, including those induced by the operating-basis earthquake (OBE) and the design wind specified for the plant. Subsection II.3.A defines the condition for which the OBE load is required for design of seismic Category I structures.

D. Those loads to be sustained during extreme environmental conditions, including those induced by the SSE and from the design basis tornadoes and hurricanes specified for the plant.

E. Those loads to be sustained during abnormal plant conditions. Such abnormal plant conditions include the postulated rupture of higher energy piping, if any. Loads induced by such an accident may include elevated temperatures and pressures within or across compartments and possibly jet impingement and impact forces associated with such ruptures.

F. Those loads induced by normal and abnormal dynamic loads, including hydrodynamic loads, if applicable (e.g., safety relief valve actuation and loss-of-coolant accidents (LOCAs)), which could generate building vibration inertial loads, including floor response spectra, direct loads on structures, and elevated temperatures.

The various combinations of the above loads that are normally postulated and reviewed include construction loads, normal operating loads, normal operating loads with severe environmental loads, normal operating loads with extreme environmental loads, normal operating loads with abnormal loads, normal operating loads with severe environmental and abnormal loads, and normal operating loads with extreme environmental and abnormal loads.

The loads and load combinations described above are generally applicable to all types of structures. However, other site-related loads might also be applicable. Such loads, which are not normally combined with abnormal loads, include those induced by floods, potential aircraft crashes (nonterrorism-related incidents), explosive hazards in proximity to the site, and projectiles and missiles generated from activities of nearby military installations.

Because the reactor building design provides locations for up to twelve NPMs operating simultaneously, the loads due to the individual and multiple NPMs considered in the design are reviewed. The loads reviewed for the COL Applicant include the loads developed based on the various combinations of NPMs which should consider the number of NPMs and possible locations of the NPMs over the entire life of the plant.

4. Design and Analysis Procedures. The review of the design and analysis procedures used for seismic Category I structures focuses on the extent of compliance with American Concrete Institute (ACI) 349, with supplemental guidance by RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants,” for concrete structures and American National Standards Institute (ANSI) / American Institute of Steel Construction (AISC) N690-1994 including Supplement 2 (2004) for steel structures. The use of more recent codes and standards is reviewed for adequacy on a case-by-case basis. The review includes the following areas:

   A. General assumptions on boundary conditions.

   B. The expected behavior under loads and the methods by which vertical and lateral loads and forces are transmitted from the various elements to their supports and eventually to the foundation of the structure. This includes the approach used to analyze and develop loads of the multiple NPMs and their interaction in the reactor pool water.

   C. The computer programs that are used.

   D. A design report on seismic Category I structures (Appendix C).

   E. Performance of a structural audit (Appendix B).

   F. Design of the spent fuel pool and racks (Appendix D).

   G. The design and analysis of anchors (steel embedments) used for component and structural supports on concrete structures are acceptable if found in accordance with “Anchoring to Concrete,” Appendix D to ACI 349-06 or 349-12, formerly Appendix B to ACI 349-01, as supplemented by RG 1.199.

   H. Dynamic soil pressures on earth retaining walls, if any exist, and embedded walls for nuclear power plant structures (Subsection II.4.H of this DSRS section).

   The review of the design and analysis procedures used for other structures that are important to safety are reviewed against applicable staff guidance (e.g., RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," for the radwaste building).

5. Structural Acceptance Criteria. The review includes the design limits imposed on the various parameters that serve to quantify the structural behavior of each structure and its components, with specific attention to stresses, strains, gross deformations, and factors of safety against structural failure. For each load combination specified, the allowable limits are compared with the acceptable limits delineated in Subsection II.5 of this DSRS section.

6. Materials, Quality Control, Special Construction Techniques, and Quality Assurance. The review covers information on the materials used in the construction of seismic Category I structures. Among the major materials of construction covered in the review are the concrete ingredients, the reinforcing bars and splices, and the structural steel and anchors.

   The staff reviews the quality control parameters that are proposed for the fabrication and construction of seismic Category I structures, including nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances.

   Special construction techniques, such as modular construction methods, if used, are reviewed on a case-by-case basis to determine their effects on the structural integrity of the completed structure.

   In addition, the applicant should provide the following information:

   A. The extent to which the materials and quality control programs comply with ACI 349, with additional criteria provided by RG 1.142 for concrete and ANSI/AISC N690-1994 including Supplement 2 (2004) for steel, as applicable.

   B. If welding of reinforcing bars is used, it should comply with American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code (Code), Section III, Division 2, as supplemented with additional guidance provided by RG 1.136, "Materials, Construction, and Testing of Concrete Containments." Any exception to compliance should be supported with adequate justification.

7. Testing and Inservice Surveillance Programs. For seismic Category I structures, the staff reviews information on structure monitoring and maintenance requirements.

   For seismic Category I structures, it is important to accommodate inservice inspection (ISI) of critical areas. The staff reviews any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, conducting remote visual monitoring of high-radiation areas) to accommodate ISI of other seismic Category I structures.

   Postconstruction testing and inservice surveillance programs for other seismic Category I structures, such as periodic examination of inaccessible areas, monitoring of ground water chemistry, and monitoring of settlements and differential displacements, are reviewed on a case-by-case basis.

8. Masonry Walls. Areas of review pertaining to masonry walls should include, at a minimum, those items identified in Appendix A to this DSRS section.

9. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this DSRS section in accordance with SRP Section 14.3.2, "Structural and Systems Engineering—Inspections, Tests, Analyses, and Acceptance Criteria," and SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this DSRS section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Sections 14.3 and 14.3.2.

10. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review also addresses COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

    For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other SRP and DSRS sections interface with this section as follows:

1. The determination of structures that are subject to QA programs in accordance with the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 is performed in accordance with SRP Sections 3.2.1 and 3.2.2. The review of safety-related structures is performed on that basis.

2. Determination of pressure loads from higher-energy lines located in safety-related structures other than containment, if any, is performed in accordance with SRP Section 3.6.1, "Plant Design for Protection against Postulated Piping Failures in Fluid Systems outside Containment." The loads thus generated are accepted for inclusion in the load combination equations of this DSRS section.

3. Determination of loads generated because of pressure under accident conditions, if any, is performed in accordance with DSRS Section 6.2.1. The loads thus generated are accepted for inclusion in the load combinations in this DSRS section.

4. The organization responsible for QA performs the reviews of design, construction, and operations phase QA programs under SRP Chapter 17. In addition, while conducting regulatory audits in accordance with Office Instruction NRR-LIC-111 or NRO-REG-108, “Regulatory Audits,” the technical staff may identify quality-related issues. If this occurs, the technical staff should contact the organization responsible for QA to determine if an inspection should be conducted.

5. Review of the Probabilistic Risk Assessment is performed under SRP Section 19.0, “Severe Accidents.”

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1, as they relate to structures, systems, and components being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed

2. GDC 2, as it relates to the design of the safety-related structures being able to withstand the most severe natural phenomena such as wind, tornadoes, hurricanes, floods, and earthquakes and the appropriate combination of all loads

3. GDC 4, as it relates to safety-related structures being appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit

4. GDC 5, as it relates to safety-related structures not being shared among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions

5. 10 CFR Part 50, Appendix B, as it relates to the QA criteria for nuclear power plants

6. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act, and the Commission’s rules and regulations

7. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will

operate in conformity with the COL, the provisions of the Atomic Energy Act, and the Commission's rules and regulations

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are set forth below. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. As an alternative, and as described in more detail below, an applicant may identify the differences between a DSRS section and the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and discuss how the proposed alternative provides an acceptable method of complying with the NRC regulations that underlie the DSRS acceptance criteria.

1. Description of the Structures. The descriptive information in the FSAR is considered acceptable if it meets the criteria set forth in Section 3.8.4 of RG 1.206 and includes information described in Subsection I.1 of this DSRS section. New or unique design features that are not specifically covered in RG 1.206, "Combined License Applications for Nuclear Power Plants," may require a more detailed review. The reviewer determines the additional information that may be needed to accomplish a meaningful review of the structural aspects of such new or unique features.

   RG 1.206 provides the basis for evaluating the description of structures to be included in a DC or a COL application.

2. Applicable Codes, Standards, and Specifications. The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of seismic Category I structures are covered by codes, standards, and guides that are either applicable in their entirety or in portions thereof. A list of such documents is given below. The use of more recent codes (e.g., applicable to ACI 349 and ANSI/AISC N690) is reviewed for adequacy on a case-by-case basis.

Codes/Specifications	Title
ACI 349	"Code Requirements for Nuclear Safety-Related Concrete Structures" (with additional criteria provided in RG 1.142)
ANSI/AISC N690-1994, including Supplement 2 (2004)	"Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities"
RGs	
1.69	"Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants"
1.91	"Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes near Nuclear Power Plants"

1.115	“Protection Against Turbine Missiles”
1.127	“Inspection of Water-Control Structures Associated with Nuclear Power Plants”
1.136	“Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments”
1.142	“Safety-Related Concrete Structures for Nuclear Power Plants (other than Reactor Vessels and Containments)”
1.143	“Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants”
1.160	“Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
1.199	“Anchoring Components and Structural Supports in Concrete”
1.221	“Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants”

3. Loads and Load Combinations. The specified loads and load combinations are acceptable if found to be in accordance with the guidance given below:

A. Concrete Structures

All loads and load combinations are to be in accordance with ACI 349 with additional guidance provided by RG 1.142. Supplemental criteria on the use of loads and load combinations are presented below.

Dead loads include hydrostatic loads and, for equipment supports, include static and dynamic head and fluid flow effects.

Live loads include any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure. The dynamic effects of lateral soil pressure should be accounted for in accordance with the provisions of Subsection II.4.H of this DSRS section. For equipment supports, live loads also include loads resulting from vibration and any support movement effects. Alternate load cases, in which the magnitudes and locations of the live loads are arranged so that the design includes worst-case conditions, should be investigated, as appropriate.

As noted in Appendix S to 10 CFR Part 50, the OBE is associated only with plant shutdown and inspection unless specifically selected by the applicant as a design input. If the OBE is set at one-third or less of the SSE ground motion, an explicit analysis or design is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the seismic Category I structures remain functional and are within applicable stress, strain, and deformation limits. DSRS Sections 3.7.1, 3.7.2, 3.7.3, and SRP Section 3.7.4 provide further guidance on the use of OBE.

Earthquake loads include the inertial effects of the NPM units considering submergence in pool water and any potential interaction effects. These loads should also consider the various configurations of the NPMs in the reactor pool over the life of the plant.

For structures or structural components subjected to hydrodynamic loads, fluid-structure interaction associated with these hydrodynamic loads and those from earthquakes should be taken into account.

The design of concrete structures needs to consider the loads and load combinations that may occur during their construction. These loads consist of dead loads, live loads, temperature, wind, snow, rain, ice, and construction loads that may be applicable such as material loads, personnel and equipment loads, horizontal and vertical construction loads, erection and fitting forces, equipment reactions, and form pressure. Structural Engineering Institute / American Society of Civil Engineers Standard 37 gives additional guidance on construction loads. This standard provides supplemental guidance, and in cases where the criteria in the standard and in the Code/DSRS conflict, then the Code/DSRS shall govern.

The analysis should consider other site-related or plant-related loads applicable to seismic Category I structures such as floods, explosive hazards in proximity to the site, potential aircraft crashes (nonterrorism-related incidents), and missiles generated from activities of nearby military installations or turbine failures. The inclusion of these loads and the related load combinations are reviewed on a case-by-case basis.

B. Steel Structures

All loads and load combinations are to be in accordance with ANSI/AISC N690-1994, including Supplement 2 (2004). This specification uses the allowable stress design (ASD) method. The supplemental criteria on the use of loads and load combinations presented above for concrete structures also apply to steel structures.

4. Design and Analysis Procedures. The design and analysis procedures used for seismic Category I structures, including assumptions about boundary conditions and expected behavior under loads, are acceptable if found to be in accordance with the following:

A. For concrete structures, the procedures are in accordance with ACI 349, as supplemented by RG 1.142. The design and analysis of anchors (steel embedments) used for component and structural supports on concrete structures are acceptable if found in accordance with “Anchoring to Concrete,” Appendix D to ACI 349-06 or 349-12, formerly Appendix B to ACI 349-01, as supplemented by RG 1.199.

B. For concrete structures, the effects of concrete shrinkage and cracking should also be considered. These effects may be established by tests performed on the concrete to be used, from data obtained on other structures, or industry codes and standards. In establishing these effects, the analysis should consider the differences in the environment between those at the facility and the data obtained from other sources.

Depending on the magnitude of the loads and potentially harmful environment experienced by the structure/foundation, concrete cracking may occur. Concrete cracking can result in redistribution of member forces. It can also affect the stiffness of the structure/foundation and cause shifting of the natural frequency, thereby affecting the response/loads used to design the structure and its foundation. Accordingly, the analysis used to calculate the dynamic response of the structure and its foundation resulting from dynamic loads such as earthquake and hydrodynamic loads (if applicable) needs to consider the potential effects of concrete cracking, if significant. The approach used should include the effect of redistribution of the various loads caused by concrete cracking. With improvements in the development of computer programs for analysis of concrete structures, the evaluation of concrete cracking can be analyzed directly within the finite element model (FEM). Alternatively, additional analyses can treat the effect of concrete cracking by determining the response of the structure and foundation to variation in the stiffness characteristics of the structure (e.g., shear stiffness and flexural stiffness reduction). Thus, concrete cracking needs to be considered depending on the stress levels caused by the most severe load combination. Technical justification should be provided, if cracking is not considered or is determined to be insignificant. Additional guidance on the modeling and treatment of concrete cracking is provided in DSRS Section 3.7.2, and Sections 3.1.3 and C 3.1.3 of ASCE 4-98.

The staff reviews the methods used for considering shrinkage and concrete cracking, or the justification for not considering these effects, on a case-by-case basis.

C. For steel structures, the procedures are in accordance with ANSI/AISC N690-1994, including Supplement 2 (2004).

D. Computer programs are acceptable if they are validated by any of the following procedures or criteria:

i. The computer program is recognized in the public domain and has had sufficient history of use to justify its applicability and validity without further demonstration.

ii. The computer program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained by a similar and independently written and recognized program in the public domain. The test problems should be demonstrated to be similar to or within the range of applicability of the problems analyzed by the public domain computer program.

iii. The computer program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained from classical solutions or from accepted experimental tests or to analytical results published in technical literature. The test problems should be demonstrated to be similar to or within the range of applicability of the classical problems analyzed to justify acceptance of the program.

A summary comparison should be provided for the results obtained in the validation of each computer program.

E. The design report is considered acceptable if it contains the information specified in Appendix C to this DSRS section.

F. The structural audit is conducted in accordance with the provisions of Appendix B to this DSRS section.

G. The design of the spent fuel pool and racks is considered acceptable when it meets the criteria of Appendix D to this DSRS section.

H. Consideration of dynamic lateral soil pressures on embedded walls is acceptable if the lateral earth pressure loads are evaluated for the governing of the following three cases. These are (1) lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure calculated in accordance with ASCE 4-98, Section 3.5.3.2(2), (2) lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure calculated using an embedded SSI/FEM analysis model, and (3) lateral earth pressure equal to the fraction of the passive earth pressure that is effectively mobilized, which is dependent on the relative magnitude of the wall displacements against the soil that may occur for a given wall configuration. For case (3), the analysis should include, as a minimum, the fraction of the passive earth pressure assumed in the stability calculations performed in accordance with DSRS Section 3.8.5.

ASCE 4-98 Section 3.5.3.2(2) describes a method based on the well-known elastic solution by Wood (1973). This method assumes linear elastic strains in a homogeneous soil mass, a rigid wall with fixed base supported on stiff soil, and no displacement or sliding of the wall base relative to the underlying soil. Soil dynamics and wave propagation effects in the soil-wall system are not considered. These assumptions may not be satisfied, for example, in the case of massive structures in soil sites where rocking could be important. Nevertheless, for cases where the assumptions of Wood's solution are realistic, the method yields conservative estimates of the dynamic pressures.

To account for a broad range of kinematic conditions, heterogeneity of the soil, as well as soil dynamics and wave propagation effects, a second method should be included based on soil-structure interaction analysis of an embedded

SSI/FEM model, as described in DSRS Section 3.7.2. A limitation of such analysis is that it also assumes linear (or equivalent-linear) elastic strains in the soil. Therefore, a third method based on passive pressure should also be included to account for potential inelastic strains.

The staff reviews the validity of the assumptions that are the basis of each of these three methods and the extent to which they correspond to the actual site conditions. In particular, the staff reviews the SSI/FEM model used in Method (2) to ensure it is appropriate to this type of application.

If other effects such as SSSI are important, these should be included in addition to the pressures computed using the methods described above.

If these methods are shown to be overly conservative for the cases considered, then the staff reviews alternative methods on a case-by-case basis. For earth retaining walls (if any exist) that are not restrained by a building, the guidance in ASCE 4-98 Sections 3.5.3.1 through 3.5.3.3 is acceptable.

I. The design of masonry walls is considered acceptable when it meets the requirements of Appendix A of this DSRS.

J. The design of structures that use modular construction methods are reviewed and evaluated on a case-by-case basis. NUREG/CR-6486, “Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants,” and/or other applicable industry documents provide guidance related to the use of modular construction methods. Appendix B to NUREG/CR-6486 includes modular construction review criteria.

K. The design and analysis procedures for the structures that may not be classified as seismic Category I but are important to safety are considered acceptable if they are in accordance with relevant guidance including RGs (e.g., RG 1.143 for radwaste structures).

L. The design and analysis procedures for the reactor pool and RB are acceptable if they consider the multiple NPMs and their interaction in the reactor pool water. This includes consideration of the number of NPMs and their locations which may vary over the life of the plant. The seismic analysis which is performed in accordance with DSRS Section 3.7.2, as well as the design approach covered under this DSRS Section 3.8.4, should consider these various combinations of NPMs. The envelope of the individual analyses of the various NPM arrangements is an acceptable approach unless it is shown that certain combinations clearly govern the design.

The effects of submergence of the NPMs in water need to be addressed in the seismic analysis and structural design. Details of the mathematical model(s) used in these analyses, including a description of how the important parameters are obtained, should be provided. The details should include the methods used to model the water in the reactor pool using finite elements or the methods used to represent the effects of submergence of the NPMs, and their interaction with each other and the pool walls. In addition, details should be provided regarding the methods used to account for the effect of sloshing water on the pool walls and biological shield cover (if applicable) and the effective damping used in the analysis. Due to the unique nature of these multiple NPMs submerged in water on a common support structure, the design and analysis approach is evaluated by the staff on a case-by-case basis.

M. To be acceptable, the design of the RB and pool structure should also include design details to prevent and monitor potential leakage from the pool.

N. The design should address potential leakage into the seismic Category I structures due to groundwater.

5. Structural Acceptance Criteria. For each of the loading combinations delineated in Subsection II.3 of this DSRS section, the structural acceptance criteria appear in ACI 349 with additional guidance provided by RG 1.142 for concrete structures, and ANSI/AISC N690-1994, including Supplement 2 (2004), for steel structures.

The structural acceptance criteria for structures that use modular construction methods are evaluated on a case-by-case basis. See Subsection II.4.J of this DSRS for information.

6. Materials, Quality Control, and Special Construction Techniques. For seismic Category I structures, the materials and quality control programs are acceptable if found in accordance with the codes and standards indicated in Subsection I.6 of this DSRS section.

Special construction techniques, if any, are evaluated on a case-by-case basis. For modular construction, reviewers evaluate the materials, quality control, and special construction techniques on a case-by-case basis. See Subsection II.4.J of this DSRS section for more information.

7. Testing and Inservice Surveillance Requirements. For seismic Category I structures, structures monitoring and maintenance requirements are acceptable if the program is in accordance with 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” and RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

For water control structures (if applicable), ISI programs are acceptable if in accordance with RG 1.127, “Inspection of Water-Control Structures Associated with Nuclear Power Plants.” Water control structures covered by this program include concrete structures, embankment structures, spillway structures and outlet works, reservoirs, cooling water channels and canals and intake and discharge structures, and safety and performance instrumentation.

For seismic Category I structures, it is important to accommodate ISI of critical areas. The staff considers that monitoring and maintaining the condition of other seismic Category I structures are essential for plant safety. The staff reviews any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high-radiation areas) to accommodate ISI of other seismic Category I structures on a case-by-case basis.

For plants with nonaggressive ground water/soil (i.e., pH>5.5, chlorides<500 parts per million (ppm), sulfates<1,500 ppm), an acceptable program for normally inaccessible, below-grade concrete walls and foundations is to (1) examine the exposed portions of below-grade concrete, when excavated for any reason, for signs of degradation, and (2) conduct periodic site monitoring of ground water chemistry, to confirm that the ground water remains nonaggressive.

For plants with aggressive ground water/soil (i.e., it exceeds any of the limits noted above), an acceptable approach is to carry out a periodic surveillance program to monitor the condition of normally inaccessible, below-grade concrete for signs of degradation.

8. Masonry Walls. Appendix A to this DSRS section contains the acceptance criteria for masonry walls.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this DSRS section is discussed in the following paragraphs:

1. Compliance with 10 CFR 50.55a, “Codes and Standards,” requires that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

   This section of the DSRS cites RGs 1.69, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants,” 1.91, “Evaluations of Explosions Postulated To Occur at Nearby Facilities and on Transportation Routes near Nuclear Power Plants,” 1.115, “Protection against Turbine Missiles,” 1.127, 1.136, 1.142, 1.143, 1.160, 1.199, and 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” to provide guidance regarding construction, quality control, tests, and inspections that are acceptable to the staff. ACI 349, as supplemented by RG 1.142, and ANSI/AISC N690-1994, including Supplement 2 (2004), contain basic specifications for concrete and steel structures, respectively. These guides and specifications impose specific restrictions to ensure that SSCs will perform their intended safety function.

   Meeting these requirements and criteria provides added assurance that the SSCs described here will perform their safety function and limit the release of radioactive materials.

2. Compliance with GDC 1 requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of their safety function; that a QA program be established and implemented; and that sufficient and appropriate records be maintained. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented as necessary to assure a quality product in keeping with the required safety function.

   This DSRS section describes staff positions related to static and dynamic loadings and evaluation programs for structures other than containment. It also describes acceptable materials, design methodology, quality control procedures, construction methods, and ISIs, as well as documentation criteria for design and construction controls.

   This section cites ACI 349, ANSI/AISC N690-1994, including Supplement 2 (2004), and RGs to provide guidance describing design methodology, materials testing, and construction techniques that are commensurate with the importance of the safety function to be performed. Conformance with these requirements imposes specific restrictions to ensure that structures other than the containment will perform acceptably, commensurate with their intended safety function, when designed in accordance with the above standards.

   Meeting these requirements and criteria provides added assurance that the SSCs described here will perform their intended safety function.

3. Compliance with GDC 2 requires that SSCs important to safety be designed to withstand the effects of expected natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

   To ensure that structures other than containment of a nuclear power plant are designed to withstand natural phenomena, it is necessary to consider the most severe natural phenomena that have been historically reported with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. These data should be used to specify the design requirements of nuclear power plant components to be evaluated as part of COL and ESP reviews, or for site parameter envelopes in the case of DCs, thereby ensuring that components important to safety will function in a manner that will maintain the plant in a safe condition.

   This DSRS section provides detailed acceptance criteria and cites appropriate regulatory guidance for design methodology, materials testing, and construction techniques acceptable to the staff. GDC 2 requires that structures other than containment be designed to withstand the effects of natural phenomena combined with those of normal and accident conditions without loss of capability to perform their safety function. Load combinations and specifications cited in this DSRS section provide acceptable engineering criteria to accomplish that function.

Meeting these requirements and criteria provides added assurance that safety-related structures will be designed to withstand the effects of natural phenomena and will perform their intended safety function.

4. Compliance with GDC 4 requires that nuclear power plant SSCs important to safety be designed to accommodate the effects of and be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs, and be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

   This DSRS section provides methods acceptable to the staff to assure compliance with GDC 4. The methods include load combinations, acceptance criteria, standards, and codes. Meeting this criterion provides assurance that structures other than containment will withstand loads from internal events, such as those described above, and from external sources such as explosive hazards in proximity to the site, potential aircraft crashes (nonterrorism-related incidents), and missiles generated from activities of nearby military installations or turbine failures, thus decreasing the probability that these events would damage structures other than containment and cause release of radioactive material.

   Meeting these requirements and criteria provides added assurance that structures will not fail in function as designed, thus protecting against loss of their structural integrity.

5. Compliance with GDC 5 prohibits the sharing of structures important to safety among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

   The requirements of GDC 5 are imposed to ensure that the use of common structures in multiple-unit plants will not significantly affect the orderly and safe shutdown and cooldown in one plant in the event of an accident in another. Loads from normal operation and design-basis accidents are combined in the load combination equations so that the resulting structural designs provide for mutual independence of shared structures.

   Meeting this requirement provides added assurance that structures other than the containment and its associated components are capable of performing their required safety function even if they are shared by multiple nuclear power units.

6. Compliance with 10 CFR Part 50, Appendix B, requires that applicants establish and maintain a QA program for the design, construction, and operation of SSCs.

   This DSRS section provides guidance specifically related to the design, construction, testing, and inservice surveillance of structural concrete and steel in nuclear power plants. Subsection II.2 of this DSRS section cites ACI 349, with additional guidance provided by RG 1.142, ANSI/AISC N690-1994, including Supplement 2 (2004), and RGs 1.127, 1.136 and 1.160, to satisfy the requirements of 10 CFR Part 50, Appendix B.

Following this guidance provides added assurance that structures covered in this DSRS section will meet the requirements of 10 CFR Part 50, Appendix B, and thus perform their intended safety function.

III. REVIEW PROCEDURES

These review procedures are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. Selected Programs and Guidance—In accordance with the guidance in NUREG–0800, "Introduction – Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light-Water Small Modular Reactor Edition" (NUREG-0800, Intro Part 2), as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800, Intro Part 2, the NRC requirements that must be met by an SSC do not change under the small modular reactor (SMR) framework. Using the graded approach described in NUREG-0800, Intro Part 2, the NRC staff may determine that, for certain SSCs, the applicant's basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified, where applicable, as part of completing the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is shown in Figure 1 of NUREG-0800, Intro Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

   - 10 CFR Part 50, Appendix A, GDC, Overall Requirements, Criteria 1–5
   - 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
   - 10 CFR 50.49, Environmental Qualification of Electrical Equipment (EQ) Program
   - 10 CFR 50.55a, Code Design, Inservice Inspection, and Inservice Testing (ISI/IST) Programs
   - 10 CFR 50.65, Maintenance Rule requirements
   - Reliability Assurance Program (RAP)
   - 10 CFR 50.36, "Technical Specifications"
   - Availability Controls for SSCs Subject to Regulatory Treatment of Nonsafety Systems (RTNSS)

- Initial Test Program (ITP)

- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

This list of examples is not intended to be all inclusive. It is the responsibility of the technical reviewers to determine whether the information in the application, including the degree to which the applicant seeks to rely on such selected programs and guidance, demonstrates that all acceptance criteria have been met to support the safety finding for a particular SSC.

2. In accordance with 10 CFR 52.47(a)(8), (21), and (22), and 10 CFR 52.79(a)(17), (20), and (37), for DC or COL applications submitted under 10 CFR Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933, "Resolution of Generic Safety Issues," current on the date up to 6 months before the docket date of the application and which are technically relevant to the design, (2) demonstrate how the operating experience insights have been incorporated into the plant design, and (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v), for a DC application, and except paragraphs (f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v),for a COL application. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.

3. Description of the Structures. After the type of structure and its functional characteristics are identified, the staff obtains information on other similar seismic Category I structures of previously licensed plants for reference. Such information, which is available in FSARs and amendments to previous license applications, enables identification of differences for the case under review. These differences require additional scrutiny and evaluation. New and unique features that have not been used in the past are of particular interest and are thus examined in greater detail.

   The reviewer evaluates the information furnished in the SAR for completeness in accordance with RG 1.206 for a DC or a COL (for application submitted in accordance with 10 CFR Part 52).

4. Applicable Codes, Standards, and Specifications. The reviewer compares the list of codes, standards, guides, and specifications with the list in Subsection II.2 of this DSRS section. The reviewer verifies that the appropriate code or guide is used and that the applicable edition and stated effective addenda are acceptable.

5. Loads and Loading Combinations. The reviewer verifies that the loads and load combinations are as conservative as those specified in Subsection II.3 of this DSRS section. The reviewer identifies any deviations from the acceptance criteria for loads

and load combinations that have not been adequately justified as unacceptable and transmits these findings to the applicant.

6. Design and Analysis Procedures. The reviewer verifies that, for the design and analysis procedures, the applicant is utilizing the specifications for concrete and steel structures found in ACI 349, with additional guidance provided by RG 1.142, and ANSI/AISC N690-1994, including Supplement 2 (2004), respectively.

   The reviewer verifies the validity of any computer programs used in the design and analysis of the structure in accordance with the acceptance criteria delineated in Subsection II.4.D of this DSRS section.

   The reviewer ensures that the applicant has met the provisions specified in Subsection II.4 of this DSRS section regarding design report, structural audits, and design of spent fuel pool and racks.

   As discussed in Subsection II.4.J of this DSRS section, reviewers evaluate the use of modular construction methods on a case-by-case basis utilizing guidance provided in NUREG/CR-6486 and/or other applicable industry documents that have been endorsed by the NRC.

7. Structural Acceptance Criteria. The reviewer compares the limits on allowable stresses and strains in the concrete, reinforcement, structural steel, etc. with the corresponding allowable stresses specified in Subsection II.5 of this DSRS section. If the applicant proposes to exceed some of these limits for some of the load combinations and at some localized points on the structure, the reviewer evaluates the justification provided to show that structural integrity will not be affected. If the reviewer determines such justification to be inadequate, the proposed deviations are identified and transmitted to the applicant with a request for adequate justification and bases.

8. Materials, Quality Control, and Special Construction Techniques. The reviewer compares the materials, quality control procedures, and any special construction techniques with those referenced in Subsection II.6 of this DSRS section. If a new material not used in previously licensed cases is used, the reviewer asks the applicant to provide sufficient test and user data to establish the acceptability of such a material. Similarly, the reviewer evaluates any new quality control procedures or construction techniques to ensure that there will be no degradation of structural quality that might affect structural integrity.

9. Testing and Inservice Surveillance Requirements. For seismic Category I structures, the reviewer verifies that monitoring and maintenance requirements for structures are in accordance with 10 CFR 50.65 and RGs 1.127 and 1.160.

   Any special design provisions (e.g., providing sufficient physical access, supplying alternative means for identification of conditions in inaccessible areas that can lead to degradation, performing remote visual monitoring of high-radiation areas) to accommodate ISI of other seismic Category I structures are reviewed on a case-by-case basis.

The reviewer evaluates any other testing and inservice surveillance programs on a case-by-case basis.

10. Masonry Walls. The reviewer should ensure that the applicant meets the requirements identified in Appendix A to this DSRS section.

11. Design Certification / Combined License Application Reviews. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the FSAR meets the acceptance criteria. DCs have referred to the FSAR as the DCD. The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

    For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an ESP, or other NRC approvals (e.g., manufacturing license, site suitability report, or topical report).

    For review of both DC and COL applications, SRP Section 14.3.2 and SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's SER. The reviewer also states the bases for those conclusions.

The staff concludes that the design of safety-related structures other than containment or containment interior structures is acceptable and meets the relevant requirements of 10 CFR 50.55a and GDCs 1, 2, 4, and 5. This conclusion is based on the following:

1. The applicant has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the structures important to safety other than containment are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the safety function to be performed. The staff used the guidelines of RGs and industry standards indicated below in making this determination.

2. The applicant has met the requirements of GDC 2 by designing the structures important to safety described in this section to withstand the effects of natural phenomena, reflecting appropriate consideration of the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

3. The applicant has met the requirements of GDC 4 by ensuring that the design of the structures important to safety are appropriately protected against dynamic effects associated with missiles, pipe whipping, and discharging fluids.

4. The applicant has met the requirements of GDC 5 by demonstrating that SSCs are not shared between units or that, if shared, the applicant has demonstrated that sharing will not impair their ability to perform their intended safety function.

5. The applicant has met the requirements of 10 CFR Part 50, Appendix B, because the QA program provides adequate measures for implementing guidelines relating to structural design audits.

6. The criteria used in the analysis, design, and construction of all the plant seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed on each structure during its service lifetime conform with established criteria, codes, standards, and specifications acceptable to the NRC staff. These include the positions of RGs 1.69, 1.91, 1.115, 1.127, 1.136, 1.142, 1.143, 1.160, 1.199, and 1.221, and industry standards ACI 349 and ANSI/AISC N690-1994, including Supplement 2 (2004).

7. The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, hurricanes, earthquakes, and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

   For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

   In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The regulations in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), and 10 CFR 52.79(a)(41) establish requirements for applications for ESPs, DCs, and COLs, respectively. These regulations require the application to include an evaluation of the site (ESP), standard plant design (DC), or facility (COL) against the SRP revision in effect 6 months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed SMR designs, however, differ significantly from large light-water nuclear power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued Staff Requirements Memorandum (SRM)-COMGBJ-10-0004/COMGEA-10-0001, “Use of Risk Insights To Enhance Safety Focus of Small Modular Reactor Reviews,” dated August 31, 2010. In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated preapplication activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for evaluating a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section, as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether an application complies with NRC regulations for NuScale SMR applications, provided that the application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. If the design or siting assumptions in a NuScale application deviate significantly from the design and siting assumptions the staff used in preparing the DSRS, the staff will use the more general guidance in the SRP, as specified in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), or 10 CFR 52.79(a)(41), depending on the type of application. Alternatively, the staff may supplement the DSRS section by adding appropriate criteria to address new design or siting assumptions.

VI. REFERENCES

1. 10 CFR 50.55a, "Codes and Standards."

2. 10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

3. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

4. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

5. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."

6. 10 CFR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems, and Components."

7. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

8. 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

9. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

10. ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures," American Concrete Institute.

11. ANSI/AISC N690-1994, including Supplement 2 (2004), "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities."

12. ASCE 4-98, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," American Society of Civil Engineers (Section 3.5.3.2 for embedded walls and Sections 3.5.3.1 through 3.5.3.3 for earth retaining walls).

13. ASME Boiler and Pressure Vessel Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," American Society of Mechanical Engineers.

14. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

15. NUREG/CR-6486, "Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants," March 1997.

16. Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants."

17. Regulatory Guide 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.”

18. Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants."

19. Regulatory Guide 1.115, "Protection Against Turbine Missiles."

20. Regulatory Guide 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants."

21. Regulatory Guide 1.136, “Materials, Construction, and Testing of Concrete Containments.”

22. Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants."

23. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in LWR Plants."

24. Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

25. Regulatory Guide 1.199, "Anchoring Components and Structural Supports in Concrete."

26. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

27. Regulatory Guide 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants.”

28. SEI/ASCE 37, "Design Loads on Structures During Construction," American Society of Civil Engineers, 2002.

29. Wood, J. H., “Earthquake-induced Soil Pressures on Structures.” EERL Report 73-05, Earthquake Engineering Research Laboratory, California Institute of Technology, 1973.

APPENDIX A TO DSRS SECTION 3.8.4

CRITERIA FOR SAFETY-RELATED MASONRY WALL EVALUATION

This appendix provides minimum design considerations and criteria for the review of safety-related masonry walls that will meet the design standards specified in Subsection II of this design-specific review standard (DSRS) section.

1. <u>General Requirements</u>

The materials, testing, analysis, design, construction, and inspection related to the design and construction of safety-related concrete masonry walls should conform to the applicable requirements contained in Uniform Building Code 1979, unless the provisions to this criteria specify otherwise.

The use of other industrial codes, such as American Concrete Institute (ACI) 531, Applied Technology Council (ATC) 3-06, or National Concrete Masonry Association (NCMA), is also acceptable. However, when the provisions of these codes are less conservative than the corresponding provisions of these criteria, their use should be justified on a case-by-case basis.

The reviewer will evaluate the use of new or updated design standards such as ACI 530 and the International Building Code (IBC) on a case-by-case basis to assure that they achieve the same level of safety as the above-referenced standards.

No unreinforced masonry walls are permitted in new construction.

2. <u>Loads and Load Combinations</u>

The loads and load combinations should include consideration of normal loads, severe environmental loads, extreme environmental loads, and abnormal loads. The following load combinations should apply (for definition of load terms, see DSRS Section 3.8.4, Subsection II.3).

A. Service Load Conditions

(1) D + L

(2) D + L + E

(3) D + L + W

If thermal stresses from T_0 and R_0 exist, they should be included in the above combinations, as follows:

(1a) $D + L + T_0 + R_0$

(1b) $D + L + T_0 + R_0 + E$

(1c) $D + L + T_0 + R_0 + W$

Check load combination for controlling condition for maximum L and for no L.

B. Extreme Environmental, Abnormal, Abnormal/Severe Environmental, and Abnormal/Extreme Environmental Conditions

(4) $D + L + T_0 + R_0 + E'$

(5) $D + L + T_0 + R_0 + W_t$

(6) $D + L + T_a + R_a + 1.5\ P_a$

(7) $D + L + T_a + R_a + 1.25\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.25\ E$

(8) $D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_r + Y_j + Y_m) + 1.0\ E'$

In combinations (6), (7), and (8), the maximum values of P_a, T_a, R_a, Y_r, Y_j, and Y_m, including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7), and (8) and the corresponding structural acceptance criteria should be satisfied first without the tornado missile load in (5) and without Y_r, Y_j, and Y_m in (7) and (8). In the review of these loads, local section strength capacities may be exceeded under these concentrated loads, provided that there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent should be checked.

3. Allowable Stresses

Allowable stresses provided in ACI 531-79, shall be supplemented by the following modifications/exceptions.

A. When wind or seismic loads (operating-basis earthquake (OBE)) are considered in the loading combinations, no increase in the allowable stresses is permitted. See Subsection II.3 of this DSRS for further guidance regarding the OBE.

B. Use of allowable stresses corresponding to a special inspection category should be substantiated by demonstration of compliance with the NRC-recommended inspection criteria.

C. All the tensile stresses will be resisted by reinforcement.

D. For load conditions that represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions, the allowable working stress may be multiplied by the factors shown in the following table:

Type of Stress	Factor
Axial or flexural compression	2.5
Bearing	2.5
Reinforcement stress except shear	2.0 but not to exceed 0.9 fy
Shear reinforcement and/or bolts	1.5
Masonry tension parallel to bed joint	1.5
Shear carried by masonry	1.3
Masonry tension perpendicular to bed joint for reinforced masonry	0
Note: When anchor bolts are used, the design should prevent facial spalling of masonry unit.	

4. Design and Analysis Considerations

A. The analysis should follow established principles of engineering mechanics and take into account sound engineering practices.

B. The assumptions and modeling techniques used should give proper consideration to boundary conditions, cracking of sections, if any, and the dynamic behavior of masonry walls.

C. Damping values to be used for dynamic analysis should be those for reinforced concrete in accordance with guidance provided in Regulatory Guide (RG) 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

D. The seismic analysis should account for the variations and uncertainties in mass, materials, and other pertinent parameters used.

E. The analysis should consider both in-plane and out-of-plane loads.

F. The analysis should consider inter-story drift effects.

G. No unreinforced masonry wall is permitted; also, all grout in concrete masonry walls should be consolidated by vibration.

H. For masonry shear walls, the minimum reinforcement requirements shall be as provided in ACI 531.

I. The acceptance of special construction (e.g., multiwythe, composite) or other items not covered by the code are reviewed on a case-by-case basis.

J. Applicants should submit for review the quality assurance/quality control (QA/QC) information.

In the event QA/QC information is not available, a field survey and a test program reviewed and approved by the NRC staff should be implemented to ascertain the conformance of masonry construction to design drawings and specifications (e.g., rebar and grouting).

5. Revision of Criteria

The criteria will be revised, as appropriate, based on experience gained during review and additional information developed through testing and research.

6. References

1. American Concrete Institute, *Building Code Requirements for Masonry Structures*, ACI 530.

2. American Concrete Institute, “Building Code Requirements for Concrete Masonry Structures,” ACI 531-79, and “Commentary,” ACI 531R-79.

3. Applied Technology Council, “Tentative Provisions for the Development of Seismic Regulations for Buildings,” ATC 3-06, 1978.

4. International Code Council, *International Building Code*.

5. U.S. Nuclear Regulatory Commission, “Damping Values for Seismic Design of Nuclear Power Plants,” RG 1.61.

6. National Concrete Masonry Association (NCMA), “Specification for the Design and Construction of Load-Bearing Concrete Masonry,” August 1979.

7. “Trojan Nuclear Plant Concrete Masonry Design Criteria Safety Evaluation Report Supplement,” November 1980.

8. International Conference of Building Officials, *Uniform Building Code*, 1979 edition.

APPENDIX B TO DSRS SECTION 3.8.4

STRUCTURAL DESIGN AUDITS

1. Introduction

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 requires, in part, that the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods or by the performance of a suitable testing program. This appendix provides requirements and guidelines for implementation of structural design audits.

2. Objectives

The audit has the following objectives:

A. to investigate the manner in which the applicant has implemented the structural design criteria that it committed to use for the facility

B. to verify that the key structural design calculations have been conducted in an acceptable way

C. to identify and assess the safety significance of those areas where the plant structures were designed and analyzed using methods other than those recommended by the design-specific review standard (DSRS) section

3. Preliminary Arrangements

The licensing project manager (LPM) arranges for the audit. The reviewer prepares the audit agenda, including specific areas of interest, and forwards it to the applicant at least 30 days before the date of the audit. The LPM should notify the appropriate Inspection & Enforcement Regional Office personnel, as well as any intervening parties, about the forthcoming audit.

4. Conduct of the Audit

A. Overview of the Plant Design

The applicant should present an overview of each of the key structures including a brief description, assumptions, modeling techniques, and technique features of design, as well as any deviations from those committed to in the final safety analysis report.

B. Audit of Design Calculations

The auditing personnel review the design calculations for the structures identified during the review of the applicant's design report. The participants in the audit should discuss and resolve any questions such as those regarding the structural modeling, analysis, proportioning of the members, and computer runs. If resolution of the questions requires additional engineering data from and further analysis by the applicant, the specific followup action items should be identified and noted in the meeting minutes for subsequent resolution.

5. <u>Exit Meeting</u>

An exit meeting is held at the conclusion of the audit to discuss and summarize the audit findings, generic issues pertaining to the design, specific action items, and the schedules for resolution of the action items.

6. <u>Minutes of the Audit</u>

The LPM is responsible for preparation of the audit minutes.

7. <u>After-Audit Meetings</u>

Review of the applicant's response to the action items may necessitate additional meeting(s) between the staff and the applicant to explain certain parts of the responses.

8. <u>Input to the Safety Evaluation Report (SER)</u>

The audit is an integral part of the review process. Resolution of the action items, together with appropriate consideration of other safety aspects, should constitute the major basis for the staff's preparation of the SER.

APPENDIX C TO DSRS SECTION 3.8.4

DESIGN REPORT

Seismic Category I Structures

I. OBJECTIVE

The primary objective of the design report provided by the applicant is to supply the reviewer with design and construction information more specific than that contained in the final safety analysis report. This information can assist the reviewer in planning and conducting a structural audit. For this review, the information must be in quantitative form representing the scope of the actual design computations and the final design results. The design report should also provide criteria for reconciliation between design and as-built conditions.

II. STRUCTURAL DESCRIPTION AND GEOMETRY

1. Structural Geometry and Dimensions

2. Key Structural Elements and Description

3. Floor Layout and Elevations

4. Conditions of Vicinity and Supports

5. Special Structural Features

III. STRUCTURAL MATERIAL REQUIREMENTS

1. Concrete

    A. Compressive Strength
    B. Modulus of Elasticity
    C. Shear Modulus
    D. Poisson's Ratio

2. Reinforcement

    A. Yield Stress
    B. Tensile Strength
    C. Elongation

3. Structural Steel

    A. Grade
    B. Ultimate Tensile Strength
    C. Yield Stress

4. Foundation Media

   A. General Description
   B. Unit Weight
   C. Shear Modulus
   D. Angle of Internal Friction
   E. Cohesion
   F. Bearing Capacity

5. Special Considerations

IV. STRUCTURAL LOADS

1. Live and Dead Load Floor Plans

2. Determination of Transient and Dynamic Loads

3. Manufacturer's Data on Equipment Loads

4. Environmental Loads

5. Torsional Effects

V. STRUCTURAL ANALYSIS AND DESIGN

1. Design Computations of Critical Elements

2. Stability Calculations

3. Engineering Drawings including Details of Connections and Joints

4. Discussion of Unique Features and Problem Resolution

VI. SUMMARY OF RESULTS

1. The Required Sections

2. The Provided Sections

3. Breakdown of Individual Load Contributions

4. Tabulation of Capacities of the Section Versus Capacities Required for Different Failure Modes (Bending, Shear, Axial Load)

5. Margins of Safety Provided

VII. CONCLUSIONS

APPENDIX D TO DSRS SECTION 3.8.4

GUIDANCE ON SPENT FUEL POOL RACKS

I. INTRODUCTION

Regulatory Guide 1.29, “Seismic Design Classification,” classifies spent fuel pool racks as seismic Category I structures. Spent fuel pool racks should be treated as safety-related components for determining quality assurance requirements (Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B) and periodic condition monitoring requirements (10 CFR 50.65 “Maintenance Rule”).

This appendix describes acceptance criteria for review of spent fuel pool racks and the associated structures which would meet the acceptance criteria specified in Subsection II of this design-specific review standard (DSRS) section. A secondary review responsibility would include the review of the material limits associated with the fuel assembly in the fuel storage racks and the effect of rack deformations on the coolability of the fuel assembly.

1. Description of the Spent Fuel Pool and Racks

The applicant should provide descriptive information including plans and sections showing the spent fuel pool in relation to other plant structures in order to define the primary structural aspects and elements relied on to perform the safety-related functions of the spent fuel pool, pool liner, and racks. The main safety function of the spent fuel pool, including the liner, and the racks is to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings (such as earthquakes) and impacts from drop of a spent fuel cask, drop of a spent fuel assembly, or drop of any other heavy object during routine spent fuel handling.

The following indicates the major structural elements reviewed and the extent of the descriptive information required:

A. Support of the Spent Fuel Racks—The applicant should describe the general arrangements and principal features of the horizontal and vertical supports to the spent fuel racks and indicate the methods of transferring the loads on the racks to the fuel pool wall and the foundation slab. All gaps (clearance or expansion allowance) and sliding contacts should be indicated. The discussion should cover the extent of interfacing between the rack system and the fuel pool walls and base slab (i.e., interface loads, response spectra, etc.).

   If connections of the racks are made to the base and to the side walls of the pool such that the pool liner may be perforated, the applicant should indicate the provisions for avoiding leakage of radioactive water from the pool.

B. Fuel Handling—The organization responsible for postulation of a drop accident and quantification of the drop parameters reviews the criteria related to fuel handling. The findings of the review are evaluated for the purpose of integrity of the racks and the fuel pool, including the fuel pool liner, in view of a postulated fuel-handling accident. The applicant should provide sketches and sufficient details of the fuel-handling system to facilitate this review.

2. Applicable Codes, Standards, and Specifications

Construction materials should conform to American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code (Code), Section III, Division 1, Subsection NF. All materials should be selected to be compatible with the fuel pool environment to minimize corrosion and galvanic effects.

Design, fabrication, and installation of spent fuel racks of stainless steel material may be performed based on ASME Code, Section III, Division 1, Subsection NF, requirements for Class 3 component supports.

3. Seismic and Impact Loads

Dynamic input data such as floor response spectra or ground response spectra are developed using the criteria described in DSRS Sections 3.7.1, 3.7.2, 3.7.3, and SRP Section 3.7.4..

For freestanding spent fuel pool racks, which are potentially subject to sliding, uplift, and impact between racks and with the pool walls, time-varying seismic excitation along three orthogonal directions (two horizontal and one vertical) should be imposed simultaneously.

For fully supported spent fuel pool racks, the response spectra analysis (RSA) method is acceptable. The peak response from each direction is combined in accordance with RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis." If response spectra are available for a vertical and horizontal direction only, the same horizontal response spectra may be applied along the other horizontal direction.

The effects of submergence in water need to be addressed in the spent fuel rack structural analysis. The effects of submergence are evaluated by the staff on a case-by-case basis.

Because of gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Additional loads resulting from this impact effect may be determined by estimating the kinetic energy of the fuel assembly. The maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly. Loads thus generated should be considered for local as well as overall effects on the walls of the rack and the supporting framework.

The applicant should demonstrate that the consequent loads on the fuel assembly do not lead to damage of the fuel. Damage of the fuel refers to structural elements of a fuel assembly (including the fuel rod cladding) which are stressed beyond the material allowable limits (established in terms of either strength or strain limits) such that the fuel rods are no longer able to provide confinement for contained radioactive fission materials.

An evaluation considering pertinent failure modes (such as buckling, etc.) should be performed to demonstrate that, when subject to the consequent loads resulting from the various load combinations described in Table 1, the structural elements of the fuel assembly will not exceed appropriate material allowable limits. Irradiation embrittlement effects, as well as pool temperature effects on the material properties, should be adequately accounted for in establishing the material allowable limits. Evaluations based on testing results to demonstrate structural integrity of the fuel assembly may also be acceptable, provided that the testing configurations and parameters are consistent with those for the fuel assembly being evaluated. To this end, the testing results are evaluated on a case-by-case basis in determining the structural integrity of the fuel assembly.

The evaluation should also confirm that any fuel assembly deformation resulting from the applicable load combinations does not degrade the coolable configuration of the fuel assembly to an unacceptable level.

Loads generated from other postulated impact events may be acceptable, if the total mass of the impacting missile, the maximum velocity at the time of impact, and the ductility ratio of the target material used to absorb the kinetic energy are described.

TABLE 1

Load Combination	Acceptance Limit
$D + L$ $D + L + T_o$ $D + L + T_o + E$	ASME Code Section III, Subsection NF, Level A service limits for Class 3
$D + L + T_a + E$ $D + L + T_o + P_f$	ASME Code Section III, Subsection NF, Level B service limits for Class 3
$D + L + T_a + E'$	ASME Code Section III, Subsection NF, Level D service limits for Class 3
$D + L + F_d$	The functional capability of the fuel racks should be demonstrated.
Limit Analysis	
Load Combination	**Acceptance Limit**

1.7 (D + L) 1.7 (D + L + T_o) 1.7 (D + L + E + T_o) 1.7 (D + L + E + T_a) 1.7 (D + L + T_o + P_f) 1.1 (D + L + T_a + E′)	ASME Code Section III, Subsection NF, paragraph 3340
Notes: 1. The abbreviations in the table above are those used in Subsection II.3 of DSRS 3.8.4 where each term is defined except for T_a, F_d, and P_f. T_a is defined here as the highest temperature associated with the postulated abnormal design conditions. F_d is the force caused by the accidental drop of the heaviest load from the maximum possible height. P_f is the upward force on the racks caused by a postulated stuck fuel assembly. 2. Deformation limits specified by the design specification limits should be satisfied, and such deformation limits should preclude damage to the fuel assemblies. 3. The provisions of ASME Code, Section III, Division 1, Subsection NF, were amended consistent with regulatory positions contained in RG 1.124, “Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports.”	

4. Loads and Load Combinations

Information pertaining to the applicable design loads and their various combinations should be provided. If applicable, any change in the temperature distribution resulting from a proposed modification to an existing spent fuel rack configuration should be identified. The temperature gradient across the rack structure that results from the differential heating effect between a full and an empty cell should be indicated and incorporated in the design of the rack structure. Maximum uplift forces available from the crane should be indicated and include consideration of these forces in the design of the racks and the analysis of the existing pool floor, if applicable.

The fuel pool racks and the fuel pool structure, including the pool slab and fuel pool liner, should be evaluated for accident load combinations which include the impact of the spent fuel cask, the heaviest postulated load drop, and/or accidental drop of the fuel assembly from the maximum height.

The review will evaluate the acceptable limits (strain or stress limits) on a case-by-case basis, but in general, the applicant is required to demonstrate that the functional capability and/or the structural integrity of each component is maintained.

The specific loads and load combinations are acceptable if they conform to the applicable portions of DSRS 3.8.4, Subsection II.3, and Table 1 provided in this appendix.

5. Design and Analysis Procedures

American National Standards Institute, N210-76, “Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants, Design,” provides general information regarding design of spent fuel pool racks.

Details of the mathematical model, including a description of how the important parameters are obtained, should be provided. The details should include the methods used to incorporate any gaps between the support systems and gaps between the fuel bundles and the guide tubes; the methods used to lump the masses of the fuel bundles and the guide tubes; the methods used to account for the effect of sloshing water on the pool walls; and the effect of submergence on the mass, the mass distribution, and the effective damping of the fuel bundle and the fuel racks.

Design and analysis procedures in accordance with DSRS 3.8.4, Subsection II, are acceptable. The effect on gaps, sloshing water, and increase of effective mass and damping resulting from submergence in water should be quantified.

If the spent fuel racks are designed to be freestanding (i.e., without connections to the pool walls/floor), then their response involves a complex combination of motions that includes sliding, rocking, and twisting and involves impacts between the fuel assemblies and the fuel cell walls, rack-to-rack, and rack-to-wall. In view of this, the seismic analysis of these fuel racks is typically performed using nonlinear dynamic time history analysis methods. For nonlinear seismic analysis of the racks, multiple time histories should be performed in accordance with the criteria for nonlinear analysis described in DSRS 3.7.1, unless otherwise justified. For the freestanding rack analyses, the entire range of the coefficient of friction for the rack material in water should be considered between the rack legs and the pool floor as well as the other contact surfaces (e.g., rack-to-rack impacts, rack-to-wall impacts). NUREG/CR-5912, “Review of the Technical Basis and Verification of Current Analysis Methods Used to Predict Seismic Response of Spent Fuel Storage Racks,” provides further guidance on the design and analysis of freestanding fuel racks.

The seismic input motion to the racks should consider the spectra at the rack base and the wall of the spent fuel pool that typically is obtained from the overall seismic building soil structure interaction analysis. It is acceptable to envelop the seismic motion at these two locations for the input loading to the racks. This approach is also applicable to freestanding racks because seismic inertial loading can be transferred from the pool walls through the water in the pool to the racks. Alternative methods that may be used should be provided and reviewed on a case-by-case basis.

When pool walls are used to provide lateral restraint at higher elevations, the applicant should provide a determination of the flexibility of the pool walls and the capability of the walls to sustain such loads. If the pool walls are flexible (having a fundamental frequency less than 33 hertz (Hz)), the floor response spectra corresponding to the lateral restraint point at the higher elevation are likely to be greater than those at the base of the pool. To use the response spectrum approach in such a case, the following two separate analyses should be performed:

A. a spectrum analysis of the rack system using response spectra corresponding to the highest support elevation provided that there is not significant peak frequency shift between the response spectra at the lower and higher elevations

B. a static analysis of the rack system by subjecting it to the maximum relative support displacement

The resulting stresses from the two analyses above should be combined by the absolute sum method.

To determine the flexibility of the pool wall, it is acceptable for the applicant to use equivalent mass and stiffness properties obtained from calculations similar to those described in *Introduction to Structural Dynamics*, McGraw-Hill Book Co., New York, 1964, by John M. Biggs. Should the fundamental frequency of the pool wall model be higher than or equal to 33 Hz, it may be assumed that the response of the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

6. <u>Structural Acceptance Criteria</u>

Table 1 of this appendix provides the structural acceptance criteria, in accordance with ASME Code, Section III, Division 1, Subsection NF. When considering compression loads, Subsection NF, paragraph 3300, specifies additional criteria that must be satisfied to preclude buckling.

For impact loading, the ductility ratios used to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. In the consideration of the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modulus under all probable service conditions should be in accordance with DSRS Section 3.8.5, Subsection II.5. This position on factors of safety against sliding and tilting need not be met provided that the applicant meets either one of the following conditions:

A. Detailed nonlinear dynamic analyses show that the amplitudes of sliding motion are minimal and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the allowable values provided in DSRS Section 3.8.5, Subsection II.5.

B. Any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and any impact resulting from the clearances is incorporated.

The fuel pool structure should be designed for the loads imposed by the racks. The fuel pool liner leak-tight integrity should be maintained, or the functional capability of the fuel pool should be demonstrated.

7. Materials, Quality Control, and Special Construction Techniques

The applicant should describe materials, quality control procedures, and any special construction techniques; the sequence of installation of the new fuel racks; and the precautions to be taken to prevent damage to the stored fuel during reracking at an operating plant.

If connections between the rack and the pool liner are made by welding, the welder, as well as the welding procedure for the welding assembly, should be qualified in accordance with the applicable code.

For spent fuel pool racks fabricated from aluminum, American Society of Civil Engineers, Suggested Specification for Structures of Aluminum Alloys 6061-T6 and 6067-T6 and “Specification for Aluminum Structures” (issued by The Aluminum Association) contain the guidance regarding material properties.

II. REFERENCES

1. U.S. Nuclear Regulatory Commission, “Seismic Design Classification,” RG 1.29.

1. U.S. Nuclear Regulatory Commission, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” RG 1.92, Agencywide Documents Access and Management System (ADAMS) Accession No. ML053250475.

2. U.S. Nuclear Regulatory Commission, “Service Limits and Loading Combinations for Class 1 Linear-Type Supports,” Regulatory Guide 1.124, ADAMS Accession No. ML070160591.

3. American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, Section III, Division 1, “Rules for Construction of Nuclear Facility Components.”

4. American National Standards Institute, ANSI N210-76, “Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations.”

5. American Society of Civil Engineers, *Suggested Specification for Structures of Aluminum Alloys 6061-T6 and 6067-T6*.

6. The Aluminum Association, “Specification for Aluminum Structures.”

7. Biggs, J.M., *Introduction to Structural Dynamics*, McGraw-Hill Book Co., New York, 1964.

8. U.S. Nuclear Regulatory Commission, “Review of the Technical Basis and Verification of Current Analysis Methods Used to Predict Seismic Response of

Spent Fuel Storage Racks," NUREG/CR-5912, October 1992, ADAMS Accession No. ML13028A043.