U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

5.3.3 REACTOR VESSEL INTEGRITY

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of component integrity issues related to reactor vessels

Secondary - None

I. AREAS OF REVIEW

Integral, pressurized, small modular reactors (SMR) generally incorporate the reactor core, and the pressurizer inside the reactor vessel. One or more steam generators (and/or reactor coolant pumps) may be inside the reactor vessel or directly connected to the reactor vessel. For the purpose of this review, the applicant should provide an accurate definition of the reactor vessel.

The portions of the applicant's safety analysis report (SAR) listed below are reviewed. These portions are all related to the integrity of the reactor vessel. Although most of these areas are reviewed separately in accordance with other Design-Specific Review Standard (DSRS) sections, the integrity of the reactor vessel is of such importance that a special summary review of all factors relating to the integrity of the reactor vessel is warranted. The information in each area is reviewed to ensure that the information is complete, and that no inconsistencies in information or requirements exist that would reduce the certainty of vessel integrity.

The specific areas of review are as follows:

1. Design. The basic design of the reactor vessel is reviewed for compatibility of design with established quality standards for material properties and fabrication methods as described in DSRS Section 5.3.1, "Reactor Vessel Materials," and for compatibility with required inspections as described in DSRS Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice Inspection and Testing."

2. Materials of Construction. The materials of construction are each taken into consideration as described in Standard Review Plan (SRP) Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," and in DSRS Section 5.3.1.

3. Fabrication Methods. The processes used to fabricate the reactor vessel, including forming, welding, cladding, and machining, are reviewed as described in DSRS Section 5.3.1.

4. Inspection Requirements. The inspection test methods and requirements are reviewed as described in DSRS Section 5.3.1.

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5. Shipment and Installation. Protective measures taken during shipment of the reactor vessel and its installation at the site are reviewed to verify that the as-built characteristics of the reactor vessel are not degraded by improper handling.

6. Operating Conditions. All the operating conditions as they relate to the integrity of the reactor vessel are reviewed as described in DSRS Section 5.3.2, “Pressure-Temperature Limits and Pressurized Thermal Shock.”

7. Inservice Surveillance. Plans and provisions for inservice surveillance of the reactor vessel are reviewed as described in DSRS Section 5.3.1 and in DSRS Section 5.2.4.

8. Operational Program Description and Implementation. For a COL application, the staff reviews the Inservice Inspection and Reactor Vessel Material Surveillance Programs description and the proposed implementation milestones. The staff also reviews final safety analysis report (FSAR) the technical submittal Table [13.x] to ensure that the Inservice Inspection and Reactor Vessel Material Surveillance Programs and associated milestones are included.

9. Threaded Fasteners. The adequacy of programs for assuring the integrity of bolting and threaded fasteners is reviewed as described in SRP Section 3.13, “Threaded Fasteners - ASME Code Class 1, 2, and 3.”

10. 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, “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 Section 14.3.

    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:

This DSRS section involves the integrated review of DSRS Sections 5.2.4, 5.3.1, and 5.3.2 and SRP Section 5.2.3 as they relate to reactor vessel integrity.

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

1. Review of the reactor vessel design regarding compliance with Title 10 of the *Code of Federal Regulations* (CFR), Section 50.55a of 10 CFR Part 50 and regarding applicable Code Cases, as part of is performed under SRP Sections 5.2.1.1 and 5.2.1.2.

2. For COL reviews of operational programs, the review of the applicant’s implementation plan is performed under SRP Section 13.4, “Operational Programs.”

3. The review of the quality assurance program is performed under SRP Chapter 17.

4. Determination of SSC risk significance is performed under SRP Section 19.0.

II. ACCEPTANCE CRITERIA

Requirements

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

The basic acceptance criteria for each review area are covered by other SRP and DSRS sections, so they will be discussed here only in general terms. References are made to the DSRS sections that include detailed criteria. The acceptance criteria in these DSRS sections describe methods to meet the requirements of the following Commission regulations in 10 CFR Part 50: General Design Criteria (GDCs) 1, 4, 14, 30, 31, and 32 of Appendix A; Appendix B; 10 CFR 50.60 and associated Appendices G and H; 10 CFR 50.55a; and 10 CFR 50.61.

1. 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 DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (NRC's) regulations;

2. 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 AEA, and the NRC's 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. Design. With regard to compatibility of design with material properties and fabrication methods, the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a are met by compliance with the provisions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter "the ASME Code"). The basic acceptance criteria for the design of the vessel are the requirements of Section III of the ASME Code. The design of the reactor vessel must be compatible with the properties of the materials used, and must permit construction by the use of standard and well proven fabrication methods. The design details should not include new or novel

concepts unless they are substantiated by a comprehensive justification showing that no aspects of the design will compromise the overall integrity of the vessel in any manner.

The design details must be adequate to permit all required inspections and to provide required access to all areas requiring inservice inspection in conformance with Section XI of the ASME Code, as detailed in DSRS Section 5.2.4. This satisfies the requirements of GDC 32 and 10 CFR 50.55a regarding inservice inspection.

If the procedures of Section IV.A of Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 do not indicate the existence of an equivalent safety margin, then Section IV.B allows the reactor vessel beltline to be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of 10 CFR 50.66, "Requirements for thermal annealing of the reactor pressure vessel." Annealing of the reactor vessel provides assurance that fracture toughness properties can be restored to satisfy the fracture toughness requirements of GDC 31.

2. Materials of Construction. The basic acceptance criteria for the materials used in the construction of the reactor vessel, and the regulations that they satisfy, are detailed in SRP Section 5.2.3 and DSRS Section 5.3.1. These criteria are the requirements of Appendix G, 10 CFR Part 50, as augmented by Sections III and IX of the ASME Code.

   The materials must be compatible with the design requirements in the GDC. Acceptability is based on standard practice and engineering judgement, with consideration being given to such factors as material form, size-related variations in properties, and nonisotropic characteristics.

   Although many materials are acceptable for reactor vessels according to Section III of the ASME Code, the special considerations relating to fracture toughness and radiation effects effectively limit the basic materials that are currently acceptable for most parts of reactor vessels to SA 533 Gr B C1 1, SA 508 C1 2, and SA 508 C1 3. Acceptability criteria for other grades will have to be developed before they can be used.

   Material compositions and expected neutron fluence must be compatible with the requirements for the material surveillance program. The reviewer uses published data to ensure that the predicted shift in toughness properties (RT_{NDT} and upper shelf energy) is conservative, based on actual material composition and predicted fluence. The predicted shift in toughness properties should be at least as conservative as that obtained by use of the most recent revision of Regulatory Guide (RG) 1.99. Acceptability of the material surveillance program, as specified in Appendix H, "Reactor Vessel Material Surveillance Program Requirements," of 10 CFR Part 50, depends on these relationships.

3. Fabrication Methods. Acceptance criteria for the basic fabrication processes and their qualification and control requirements, and the regulations satisfied by these criteria, are detailed in DSRS Section 5.3.1. These criteria are given in Sections III and IX of the Code.

   Although a particular fabrication process (such as multiple wire-high heat input welding) may be generally acceptable, it may not be suitable for reactor vessel fabrication for some materials without further justification or qualification. The reviewer uses "state-of-the-art" criteria and past practice to evaluate the acceptability of materials process combinations.

Because fabrication methods, materials, and the effectiveness of nondestructive evaluation methods are interrelated, the reviewer should rely on state-of-the-art knowledge and past practice to determine whether the proposed combinations are compatible and acceptable.

4. Inspection Requirements. The basic requirements for performing nondestructive inspections, the quality assurance criteria for the reactor vessel, and the regulations that all of these criteria satisfy, are detailed in DSRS Section 5.3.1. These requirements and criteria are contained in Section III of the ASME Code. Additional criteria are contained in Section V of the Code.

   Acceptance criteria for compatibility with materials and fabrication areas are discussed in previous sections.

   Very important relationships are those among in-process and final shop inspections, and the inservice inspection requirements of Section XI of the ASME Code. The reviewer should determine whether the methods of inspection, the sensitivity levels, and flaw evaluation criteria are compatible with Section XI, and whether the results of the preservice baseline inspection can be correlated with the results of later inservice inspections.

5. Shipment and Installation. The basic acceptance criteria for procedures and care to maintain proper cleanliness and freedom from contamination during all stages of shipping, storage, and installation of the reactor vessel, and the regulations that these criteria satisfy, are given in SRP Section 5.2.3.

   The purpose of this area of review is to verify that the as-built characteristics and inspectability of the reactor vessel are not degraded by improper handling. Acceptability in these areas is assured for current designs and materials by compliance with the basic acceptance criteria. If nonstandard materials or designs are used, the reviewer should determine whether criteria will be adequate, based on current technology.

   If the basic criteria are not followed, either intentionally or through error, the reviewer should evaluate, on a case basis, whether the integrity of the reactor vessel is compromised, using current technology, past practice, and experience as applicable.

6. Operating Conditions. Acceptance criteria for operating limits for the reactor vessel, and the regulations that they satisfy, are detailed in DSRS Section 5.3.2. These acceptance criteria are given in Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 and for pressurized-water reactors (PWRs), 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

   The criterion for acceptable behavior is that the vessel remains leaktight enough to support adequate core cooling. The generally accepted principles and procedures of linear elastic fracture mechanics provide the basis for acceptance of analyses that support conformance with this criterion.

7. Inservice Surveillance. The acceptance criteria for adequacy of the reactor vessel materials surveillance program, and the regulations satisfied by the criteria, are detailed in DSRS Section 5.3.1. The criteria are based on the requirements of Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50.

   The SAR also provides information regarding the inservice inspections to be performed on the reactor vessel. The acceptance criteria for accessibility and inspection plan details, and the regulations that they satisfy, are detailed in DSRS Section 5.2.4. These criteria are those of Section XI of the ASME Code.

8. Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the Inservice Inspection and Reactor Vessel Material Surveillance Programs are reviewed under DSRS Sections 5.2.4 and 5.3.1 respectfully, in accordance with 10 CFR 50.55a(g), 10 CFR 50.60 and 10 CFR Part 50, Appendix H. The Reactor Vessel Material Surveillance Program and associated implementation milestone(s) are included within the license condition on operational program implementation.

9. 10 CFR 52.47(b)(1) requires that a DC application contain proposed ITAAC necessary and sufficient to assure the plant is built and will operate in accordance with the DC. 10 CFR 52.80(a) requires that the COL identify the ITAAC necessary and sufficient to assure that the facility has been constructed and will be operated in conformity with the license. SRP Section 14.3 provides guidance for reviewing the ITAAC. The requirements of 10 CFR 52.47(b)(1) and 10 CFR 52.80(a) will be met, in part, by identifying ITAAC of the top-level design features with respect to reactor vessel integrity in the DC and COL applications, respectively.

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:

This DSRS section involves the integrated review of reactor vessel integrity based on individual reviews performed for other DSRS sections and does not introduce any new or additional criteria. Technical rationale for the acceptance criteria described above are provided in DSRS Sections 5.2.4, 5.3.1, and 5.3.2 and SRP Section 5.2.3.

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.

Because the reviewer is familiar with the specific procedures used by current reactor vendors, he or she can readily pick out any differences related to iPWRs from past practice. The reviewer will evaluate these iPWR reactor vessel differences in detail, consulting with other staff as appropriate.

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. Operational Programs. The reviewer verifies that the Inservice Inspection and Reactor Vessel Material Surveillance Programs are fully described and that implementation milestones have been identified. The reviewer verifies that the program and implementation milestones are included in FSAR Table 13.x.

Implementation of this program will be inspected in accordance with NRC Inspection Manual Chapter IMC-2504, "Construction Inspection Program - Inspection of Construction and Operational Programs."

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 design control document (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 early site permit 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 and SRP Section 14.3.4 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 staff's technical 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.

For the reasons set forth above, the staff concludes that the structural integrity of the reactor vessel is acceptable and meets the requirements of GDCs 1, 4, 14, 30, 31, and 32 of Appendix A of 10 CFR Part 50; the requirements of 10 CFR Part 50, Appendix B; the requirements of 10 CFR 50.60 and associated Appendices G, and H; the requirements of 10 CFR 50.55a; and for PWRs, the requirements of 10 CFR 50.61. This conclusion is based on the staff's review of the SAR, conducted in accordance with the following SRP sections, and supplemented by the acceptance criteria of DSRS Section 5.3.3:

1. SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials."

2. DSRS Section 5.2.4, “Reactor Coolant Pressure Boundary Inservice Inspection and Testing.”

3. DSRS Section 5.3.1, “Reactor Vessel Materials.”

4. DSRS Section 5.3.2, “Pressure-Temperature Limits and Pressurized Thermal Shock.”

5. SRP Section 3.13, “Threaded Fasteners - ASME Code Class 1, 2, and 3.”

We have reviewed all factors contributing to the structural integrity of the reactor vessel and conclude there are no special considerations that make it necessary to consider potential reactor vessel failure for this plant. The bases for our conclusion are that the design, materials, fabrication, inspection, and quality assurance requirements for the plant will conform to applicable NRC regulations and RGs, and to the rules of the ASME Boiler and Pressure Vessel Code, Section III. The stringent fracture toughness requirements of the regulations and ASME Code Section III will be met, including requirements for surveillance of vessel material properties throughout service life, in accordance with Appendix H of 10 CFR Part 50. Also, operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G, “Protection Against Non-ductile Failure,” of ASME Code Section XI, Appendix G to 10 CFR Part 50, and 10 CFR 50.61.

The integrity of the reactor vessel is assured because the vessel:

1. will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and any pertinent Code Cases;

2. will be made from materials of controlled and demonstrated high quality;

3. will be subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fail because of material or fabrication deficiencies;

4. will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation, maintenance, testing, and anticipated operational occurrences;

5. will be subjected to periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions;

6. may be annealed to restore the material toughness properties if this becomes necessary; and

7. will be subjected to surveillance to account for neutron irradiation damage so that the operating limitations may be adjusted.

The applicant described the Inservice Inspection Program and its implementation in DSRS Section 5.2.4 in conformance with 10 CFR 50.55a(g). In addition, the applicant described the Reactor Vessel Material Surveillance program and its implementation in DSRS Section 5.3.1 in conformance with 10 CFR 50.60 and 10 CFR Part 50, Appendix H.

The reviewer ensures the Inservice Inspection and Reactor Vessel Material Surveillance Programs and their associated implementation milestones in DSRS Section 5.2.4 and DSRS Section 5.3.1 are included within the license condition on operational program implementation.

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 50.60, “Acceptance Criteria for Fracture Prevention Measures for Light-water Nuclear Power Reactors for Normal Operation.”

3. 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”

4. 10 CFR Part 50, Appendix A, GDC 1, “Quality Standards and Records.”

5. 10 CFR Part 50, Appendix A, GDC 4, “Environmental and Dynamic Effects Design Bases.”

6. 10 CFR Part 50, Appendix A, GDC 14, “Reactor Coolant Pressure Boundary.”

7. 10 CFR Part 50, Appendix A, GDC 30, “Quality of Reactor Coolant Pressure Boundary.”

8. 10 CFR Part 50, Appendix A, GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary.”

9. 10 CFR Part 50, Appendix A, GDC 32, “Inspection of Reactor Coolant Pressure Boundary.”

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

11. 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements.”

12. 10 CFR Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements.”

13. RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials.”

14. ASME Boiler and Pressure Vessel Code, Sections II, III, V, IX, and XI, American Society of Mechanical Engineers.

15. ASME Boiler and Pressure Vessel Code, Section III, Appendix G, “Protection Against Nonductile Failure,” American Society of Mechanical Engineers.

16. 10 CFR 50.66, “Requirements for thermal annealing of the reactor pressure vessel.”

17. NRC Inspection Manual Chapter IMC-2504, “Construction Inspection Program - Inspection of Construction and Operational Programs,” issued September 15, 2009.