U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

6.2.7 FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The reactor containment pressure boundary relates to the reactor containment system. The reactor containment system design must include the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products attendant to postulated accidents. This SRP section reviews fracture prevention of the reactor containment pressure boundary materials.

The reactor containment pressure boundary, as addressed in the NRC licensing review process, consists of those ferritic steel parts of the reactor containment system which sustain loading and provide a pressure boundary in the performance of the containment function under the operating, maintenance, testing and postulated accident conditions cited by General Design Criterion (GDC) 51. Within this context, typically reviewed are the ferritic materials of components such as freestanding containment vessels, equipment hatches, personnel airlocks, heads of primary containment drywells, tori, containment penetration sleeves, process pipes, end closure caps and flued heads, and penetrating-piping systems connecting to penetration process pipes and extending to and including the system isolation valves.

USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in the Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of the standard format have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) will be based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," until the SRP itself is updated.

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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The specific areas of review are as follows:

1. The containment vessel and all penetration assemblies or appurtenances attached to the containment vessel; all piping, pumps and valves attached to the containment vessel, or to penetration assemblies out to and including the pressure boundary materials of any valves required to isolate the system and provide a pressure boundary for the containment function.

2. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this SRP section is reviewed in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification." The staff recognizes that the review of ITAAC is performed after review of the rest of this portion of the application against acceptance criteria contained in this SRP section. Furthermore, the ITAAC are reviewed to assure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

3. COL Action Items and Certification Requirements and Restrictions. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

Review Interfaces

The listed SRP sections interface with this section as follows:

1. Review of the design of concrete containments will be performed under SRP Section 3.8.1, "Concrete Containment," and the design of steel containments will be performed under SRP Section 3.8.2, "Steel Containment."

2. Review of the adequacy of programs for assuring the integrity of bolting and threaded fasteners will be performed under SRP Section 3.13, "Threaded Fasteners."

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

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

1. GDC 1 (Reference 2), found in Appendix A to Part 50, as it relates to the quality standards for design and fabrication.

2. GDC 16 (Reference 3), as it relates to the prevention of the release of radioactivity to the environment.

3. GDC 51 (Reference 4), as it relates to the reactor containment pressure boundary being designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

4. 10 CFR 52.47(a)(1)(vi), as it relates to ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification.

5. 10 CFR 52.97(b)(1), as it relates to ITAAC (for combined licenses) sufficient to assure that the SSCs in this area of review have been constructed and will be operated in conformity with the license and the Commission's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in Subsection I of this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. To meet the requirements of GDC 1, 16 and 51, ferritic containment pressure boundary materials should meet the fracture toughness criteria and requirements for testing identified in Article NE-2300 of Section III, Division 1 or Article CC-2520 of Section III, Division 2 of the ASME Code (Reference 6 and 7) or, for materials that were not fracture toughness tested as discussed below, the fracture toughness criteria for Class 2 components identified in the Summer 1977 Addenda to Section III, Division 1, Subsection NC (Reference 8) of the ASME Code.

2. Mandatory fracture toughness testing of ASME Code Section III Class 2 materials was first identified in the Summer 1977 Addenda Code Class 2 rules. As a result, cases exist where Class 2 ferritic materials of the reactor containment pressure boundary were not fracture toughness tested, because the ASME Code Edition and Addenda in effect at the time the components were ordered, did not require that they be tested. The staff's assessment of the fracture toughness of materials that were not fracture toughness tested is based on the metallurgical characterization of these materials and fracture toughness data presented in NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," (Reference 9) and ASME Code Section III, Summer 1977 Addenda, Subsection NC. The metallurgical characterization of these materials, with respect to their fracture toughness, is developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addenda of the ASME Code Section III, provides the technical basis for the staff's evaluation of the compliance with Code Class 2 requirements of the materials which were not fracture toughness tested.

Technical Rationale

The technical rationale for application of these requirements to reviewing this SRP section is discussed in the following paragraphs:

1. GDC 1 requires that structures, systems and components be designed, fabricated, erected and tested commensurate with the importance of the safety functions to be performed. This SRP section evaluates the fracture toughness of the containment pressure boundary ferritic materials to ensure they are not subject to brittle fracture. ASME Code Section III, Division 1, Class MC (Metal Containment) or Section III, Division 2, Class CC (concrete containment) or Class 2 component criteria are used in the performance of this fracture toughness evaluation. The application of Code Class MC or CC component criteria for the evaluation metal containment components and the specific application of Subsection NE-2000 for evaluation of steel containment materials or CC-2520 for concrete containment liners, are established staff practices reflecting that Code Class MC or CC requirements for materials, design, fabrication, and testing are commensurate with the safety function of containment (see SRP Sections 3.8.1 and 3.8.2). The application of Code Class 2 criteria for materials that were not fracture toughness tested is consistent with the methodology for application of quality standards to pressure-retaining components commensurate with the importance of their safety functions as described in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants" (Reference 10). The consistency is developed in that the containment system is addressed in the licensing review process as an engineered safety feature, as is, for example, the emergency core cooling system. Regulatory Guide 1.26 does not explicitly discuss or classify the containment pressure boundary, but does assign a Quality Group B classification to the emergency core cooling system. Regulatory Guide 1.26 assigns correspondence between Quality Group B components and ASME Code Section III, Division 1 requirements for Class 2 components. The containment pressure boundary is one of the barriers that prevent the release of radioactivity to the environment in the event of an accident, and therefore fulfills a vital safety-related role. Use of appropriate design and fabrication standards in conjunction with Article NE-2300 or CC-2520 fracture toughness testing or evaluation of ferritic containment pressure boundary materials with respect to ASME Code Class 2 fracture toughness requirements provides assurance that containment will not fail due to brittle behavior and will thus be capable of preventing the release of radioactivity to the environment.

2. GDC 16 requires reactor containment and associated systems to be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Containment must be leak tight and withstand accidents because it is the final barrier against the release of radioactivity to the environment in the event of a LOCA. To ensure leak tightness, containment must not be subject to brittle fracture even under the most severe postulated conditions. Meeting GDC 16 provides assurance that containment will satisfactorily fulfill its safety role and that significant radioactivity will not be released to the environment.

3. GDC 51 requires, part, that the reactor containment boundary be designed with sufficient margin to assure that under operating, maintenance, testing and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. As the final barrier against the release of radioactivity to the environment, containment must not be subject to brittle failure or rapidly propagating fracture, either of which could cause a breach of containment integrity. Meeting GDC 51 will ensure that the containment pressure boundary remains intact during the harshest expected conditions, thereby precluding the release of radioactivity to the environment.

III. <u>REVIEW PROCEDURES</u>

The reviewer will select and emphasize material from the procedures described below, as may be appropriate for a particular case.

For each area of review specified in subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in Section II.

The licensing review process assesses the fracture toughness of the materials of the components of the reactor containment pressure boundary identified in Section I, within the context of compliance with the criteria of Article NE-2300 of Section III, Division 1 or Article CC-2520 of Section III, Division 2 of the ASME Code.

The reviewer addresses the information provided by the applicant for the materials of the components of interest. Such information should consist of construction drawings, piping system diagrams and related supplemental information, ASME Code data reports and certified material test reports.

1. For those ferritic materials for which fracture toughness data are unavailable, or are inappropriate, the reviewer addresses the applicant's assessment of their fracture toughness based on a metallurgical characterization developed from a review of how these materials were fabricated and the thermal history they experienced during fabrication. The reviewer addresses the applicant's correlation of this information with the fracture toughness data presented in NUREG-0577 and ASME Section III, Summer 1977 Addenda, Subsection NC. The reviewer addresses the applicant's justification of the acceptability of these materials within the context of the criteria for Class 2 materials as stated in the Summer 1977 Addenda, ASME Code Section III. The reviewer verifies that the Class 2 requirements of the Summer 1977 Addenda of ASME Section III Code have been met by the applicant.

2. For reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an ESP or other NRC-approved material, applications, and/or reports.

After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

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 safety evaluation report. The reviewer also states the bases for those conclusions.

The staff concludes that reasonable assurance has been provided that the materials of the reactor containment pressure boundary, under operating, maintenance, testing and postulated accident conditions, will not undergo brittle fracture, and that the probability of rapidly propagating fracture will be minimized, so that the requirements of General Design Criteria 1, 16, and 51 will be met. This conclusion is summarized as follows (provide the finding that applies):

> Based on its review, the staff finds that the ferritic materials of the reactor containment pressure boundary were (or will be where appropriate) acceptably tested and demonstrated to meet the fracture toughness requirements for Class MC components as specified in Article NE-2300 of ASME Code Section III, Division 1 or Class CC components as specified in Article CC-2520 of ASME Code Section III, Division 2.
>
> OR
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> For ferritic reactor containment pressure boundary materials that were not fracture toughness tested, based on the licensing process review of the applicant's available fracture toughness data, metallurgical characterizations of the materials of interest developed from their fabrication and thermal histories, and correlations of metallurgical histories with fracture toughness data presented in NUREG-0577 and ASME Code Section III, Summer 1977 Addenda, Subsection NC, the conclusion is made that the fracture toughness of the materials of the reactor containment pressure boundary meet the fracture toughness requirements invoked for ASME Code Section III Class 2 materials effective with the Summer 1977 Addenda.

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission’s regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section, unless superceded by a later revision.

VI. REFERENCES

1. 10 CFR Part 50, §50.55a, Codes and Standards.

2. 10 CFR Part 50, Appendix A, General Design Criterion 1, “Quality Standards and Records.”

3. 10 CFR Part 50, Appendix A, General Design Criterion 16, “Containment Design.”

4. 10 CFR Part 50, Appendix A, General Design Criterion 51, “Fracture Prevention of Containment Pressure Boundary.”

5. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

6. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, “Class MC Components,” American Society of Mechanical Engineers.

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

8. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Summer 1977 Addenda, Subsection NC, “Class 2 Components”, American Society of Mechanical Engineers.

9. NUREG-0577 Revision 1, “Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports,” USNRC, October 1983.

10. Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.”

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the draft Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in (Draft) Revision 1, dated April 1996 of this SRP. See ADAMS accession number ML052070465.

In addition, this SRP section was administratively updated in accordance with NRR Office Instruction, LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of the updated SRP section to prospective submittals by applicants pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 1, dated [Month] 2007:

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

1. Simplified introductory paragraphs by eliminating discussions of topics that are provided in other subsections of the SRP.

2. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC). This subsection was added for the purpose of addressing design certification and combined license reviews performed pursuant to 10 CFR Part 52.

Review Interfaces:

Added Review Interfaces paragraph to describe supporting reviews performed under other SRP sections. Added a review interface reflecting reviews of bolting and threaded fastener programs under new SRP Section 3.13.

II. ACCEPTANCE CRITERIA

1. Added acceptance criteria for 10 CFR 52.47(a)(1)(vi) in reference to ITAAC requirements for design certification reviews.

2. Added acceptance criteria for 10 CFR 52.97(b)(1) in reference to ITAAC requirements for combined license reviews.

3. The specific criterion was modified to refer to ASME Code Section III, Subsection NE-2000, or ASME Code Section III, Division 2, Class CC (Article NE-2300 or Article CC-2500 as the specific location of fracture toughness criteria) and to restore Class 2 component criteria for materials which were not fracture toughness tested.

4. Added Technical Rationale for GDCs 1, 16, and 51. Technical Rationale is a new section per the updated SRP format.

5. Discussion regarding Regulatory Guide 1.26 was moved to Subsection II, Acceptance Criteria, Technical Rationale 1. This discussion is not an acceptance criterion, but rather an explanation of why certain acceptance criteria exist. The discussion is more appropriate in the new Technical Rationale subsection. The text of the discussion was updated to reflect terminology used in the current version of Regulatory Guide 1.26.

III. REVIEW PROCEDURES

1. Added a paragraph to address the performance of design certification reviews and combined license reviews pursuant to 10 CFR Part 52.

IV. EVALUATION FINDINGS

1. Added a new item to the Evaluation Findings subsection to discuss findings pertinent to ASME Code Section III, Article NE-2300 and provided for a contingent finding based on whether materials were fracture toughness tested.

2. Added the last paragraph to address the performance of design certification reviews and combined license reviews pursuant to 10 CFR Part 52.

V. IMPLEMENTATION

1. Added boilerplate text to implementation subsection to incorporate 10 CFR Part 52 and to address applicability of the section to existing and future applications.

VI. REFERENCES

1. References updated references to reflect applicable regulations and guidance and renumbered per updated SRP format.