U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

4.5.2 REACTOR INTERNAL AND CORE SUPPORT STRUCTURE MATERIALS

REVIEW RESPONSIBILITIES

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

Secondary - Organization responsible for the review of materials engineering issues related to flaw evaluation and welding

I. AREAS OF REVIEW

Section 50.55a, “Codes and Standards,” of 10 CFR Part 50, and General Design Criterion (GDC) 1 of Appendix A to 10 CFR Part 50 require that structures, systems, and components (SSCs) important to safety be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed. The purpose of this standard review plan (SRP) section is to review and evaluate the adequacy of the materials selected for the construction of the reactor internal and core support structures, as defined in NG-1120 of Section III of the ASME Boiler and Pressure Vessel Code (hereafter “the Code”), and to assure that the reactor internal and core support structures meet these regulations. The reactor internal and core support structures reviewed under this SRP section include all structures and components within the pressure vessel other than the fuel and control assemblies, and instrumentation.

This SRP section covers the material, component design, fabrication and inspection to assure structural integrity in compliance with Section 50.55a and General Design Criterion 1 of Appendix A to 10 CFR Part 50.

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The following areas in the applicant's safety analysis report (SAR) relating to reactor internal and core support structure materials are reviewed; specific areas of review are as follows:

1. Materials. The review includes the acceptability of the materials, including weld materials, to be used for the reactor internals and core support structures.

   The adequacy and suitability of the materials specified for the reactor internals and core support structures are reviewed in terms of their fracture toughness, stress corrosion resistance, fabricability, and other mechanical properties.

2. Controls on Welding. The review includes the controls on welding for reactor internals and core support structures.

3. Nondestructive Examination. The review includes information submitted by the applicant on the nondestructive examination procedures used for inspection of each product form.

4. Austenitic Stainless Steel. Austenitic stainless steels are primarily used for the construction of the reactor internals and core support structures. These steels may be used in a variety of product forms, including several stabilized product forms. Unstabilized austenitic stainless steels, such as Types 304 and 316, are frequently specified.

   Since unstabilized compositions are susceptible to stress corrosion cracking when exposed to certain environmental conditions, process controls must be exercised during all stages of component manufacturing and reactor construction to avoid sensitization of the material, and to minimize exposure of the stainless steel to contaminants that lead to stress corrosion cracking. The review includes information submitted by the applicant in these areas, as described in SRP Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.”

5. Other Materials. Materials other than austenitic stainless steels are reviewed and evaluated in terms of their fracture toughness, corrosion resistance, fabricability, and other mechanical properties.

6. 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 SRP 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 SRP 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.

7. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address 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 sections interface with this section as follows:

1. The review of the adequacy of programs for assuring the integrity of bolting and threaded fasteners is performed under SRP Section 3.13, "Threaded Fasteners - ASME Code 1, 2, and 3."

2. The evaluation of corrosion and compatibility of reactor internals and core support structures materials with the expected environment during service is performed under SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials."

3. The review of acceptability of the reactor coolant chemistry and associated chemistry controls (including additives such as inhibitors) as it relates to corrosion control and compatibility with materials to be exposed to reactor coolant is performed under SRP Sections 5.4.8 "Reactor Water Cleanup System (BWR)" and 9.3.4 "Chemical and Volume Control System (PWR)."

4. The review of the adequacy of design fatigue curves for reactor internals and core support structures materials with respect to cumulative reactor service-related environmental and usage factor effects and consideration of each combination of loadings is performed under SRP Sections 3.9.1, "Special Topics for Mechanical Components," and 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures."

5. The review of the reactor internals and core support structures with respect to their mechanical design adequacy to withstand design and service loading combinations is performed under SRP Section 3.9.5, "Reactor Pressure Vessel Internals."

6. The review of the plant design, including the selection of materials to minimize activation products, to verify that occupational radiation exposures will be as low as is reasonably achievable (ALARA) is performed under SRP Section 12.1, "Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable."

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

II. ACCEPTANCE CRITERIA

Requirements

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

The design, fabrication, and testing of the materials used in the reactor internals and core support structures are reviewed and evaluated to meet codes and standards commensurate with the safety functions to be performed such that the relevant requirements of 10 CFR 50.55a and GDC 1 are met.

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

2. 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records," which requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. 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 or modified as necessary to assure a quality product in keeping with the required safety function. GDC 1 also requires that appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

3. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (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 plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

4. 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 combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

<u>SRP Acceptance Criteria</u>

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in 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. <u>Materials</u>. For core support structures and reactor internals, the permitted material specifications are those given in the ASME Code, Section III, Division 1, Sub-subarticle NG-2120. The properties of these materials are specified in Tables 2A, 2B and 4 of Section II of the Code.

Additional permitted materials and their applications are identified in ASME Code Cases approved for use as described in Regulatory Guide 1.84, “Design, Fabrication, and Material Code Case Acceptability, ASME, Section III.”

2. Controls on Welding. Methods and controls for core support structures and reactor internals welds shall be in accordance with ASME Code, Section III, Division 1, Article NG-4000. The examination requirements and acceptance criteria for these welds are specified in Article NG-5000.

3. Nondestructive Examination. Nondestructive examinations shall be in accordance with the requirements of ASME Code, Section III, Division 1, Subarticle NG-2500. The nondestructive examination acceptance criteria shall be in accordance with the requirements of ASME Code, Section III, Division 1, Subarticle NG-5300.

4. Austenitic Stainless Steels. The acceptance criteria for this area of review are given in SRP Section 5.2.3, subsections II.2 and II.4.a, b, d, and e.

   Regulatory Guide 1.44 provides acceptance criteria for preventing intergranular corrosion of stainless steel components. In conformance with this guide, furnace sensitized material should not be allowed. Methods described in this guide should be followed for cleaning and protecting austenitic stainless steel from contamination during handling, storage, testing, and fabrication, and for determining the degree of sensitization that occurs during welding.

5. Other Materials. All materials used for reactor internals and core support structures must be selected for compatibility with the reactor coolant, as specified in Subsubarticles NG-2160 and NG-3120 of Section III, Division 1 of the ASME Code. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardened stainless steels should be specified to provide assurance that these materials will not deteriorate in service. Acceptable heat treatment temperatures are 565°C - 595°C (1050°F - 1100°F) for aging of Type 17-4 PH and 565°C (1050°F) for tempering of Type 410 stainless steel.

   Other materials shall have similar appropriate heat treat and fabrication controls in accordance with strength and compatibility requirements.

Technical Rationale

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

GDC 1 and 10 CFR 50.55a require that SSCs be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. 10 CFR 50.55a also incorporates by reference the applicable editions and addenda of the ASME Boiler and Pressure Vessel Code. The reactor internals and core support structures include SSCs that perform safety functions and/or whose failure could affect the performance of safety functions by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the primary reactor coolant system). Application of 10 CFR 50.55a and GDC 1 to the materials of construction provides assurance that established standard practices of proven or demonstrated effectiveness for selecting materials, fabrication, and testing/ inspection of SSCs are used to achieve a high likelihood that these safety functions will be performed.

III. REVIEW PROCEDURES

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

These review procedures are based on the identified SRP 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. Materials. The list of the materials for reactor internals and core support structures that are exposed to the reactor coolant is reviewed.

   The materials identified for each component or part used in the reactor internals and core support structures are compared with the materials identified as being acceptable in Sections II and III of the ASME Code and/or acceptable ASME Code Cases identified in Regulatory Guide 1.84, as described in the acceptance criteria. The reviewer verifies that any exceptions to the ASME Code-specified materials are clearly identified. The reviewer evaluates the basis for the exceptions, taking into account precedents set in earlier cases, and determines the acceptability of such materials.

2. Controls on Welding. The reviewer verifies that welding methods and controls for the reactor internals and core support structures are in accordance with the procedures of ASME Code, Section III, Division 1, Article NG-4000. The reviewer verifies that welding controls submitted by the applicant are in conformance with the welding controls in SRP Section 5.2.3, which are also considered applicable to welding of reactor internals. The reviewer assures that any special welding processes or welding controls conform to the qualification requirements of ASME Code, Section IX, or that justification is made for any deviation.

3. Nondestructive Examination. The information submitted by the applicant is reviewed to determine methods used for nondestructive examination. The reviewer verifies that the nondestructive examination methods proposed by the applicant are in conformance with the examination methods specified by the ASME Code. Section III, Division 1, Subarticle NG-2500 of the ASME Code specifies that examination by either radiographic or ultrasonic examination plus surface examinations as required is acceptable.

4. Austenitic Stainless Steel. The materials and fabrication procedures used for reactor internals are reviewed. The areas of review and review procedures include those described in SRP Section 5.2.3. The reviewer verifies that environmental conditions are controlled and welding procedures are developed such that the probabilities of sensitization and microfissuring are minimized. SRP Section 4.5.1, Subsection III.2, identifies an acceptable alternate to the methods described in Regulatory Guide 1.44 for verifying the degree of sensitization that occurs during welding. In addition, the reviewer

verifies that materials are selected to assure compatibility with the compositions of the reactor coolant, and that the fabrication and cleaning controls imposed on stainless steel components are adequate to prevent contamination with chloride and fluoride ions.

Where cast austenitic stainless steels are proposed for use, the reviewer verifies that, under the expected environmental conditions, the selected material will provide adequate fracture toughness over its design life (e.g., considering thermal aging due to exposure to reactor coolant operating temperatures).

5. Other Materials. The reviewer verifies that the heat treatment and welding controls provided in the material specifications and fabrication procedures are appropriate for the material. The reviewer verifies that the fabrication and cleaning controls will preclude contamination of nickel-base alloys by chloride ions, fluoride ions, or lead.

   Operating experience has indicated that certain nickel-chromium-iron alloys (e.g. Alloy 600 and associated weld materials, Alloy 82 and 182) are susceptible to stress corrosion cracking, as documented in NUREG-1823 and NRC Generic Letter 97-01. Alloy 690, and associated weld materials Alloy 52 and 152, have improved corrosion resistance in comparison to Alloy 600 used in PWR reactor coolant pressure boundary (RCPB) applications. Where nickel-chromium-iron alloys are proposed for use in the PWR RCPB, use of Alloy 690 materials is preferred. If Alloy 600 material is proposed, the reviewer verifies that an acceptable technical basis is either identified (based upon demonstrated satisfactory use in similar applications) or presented by the applicant to support use of the material under the expected environmental conditions (e.g., exposure to the reactor coolant). In addition, the reviewer verifies that acceptable augmented inspection requirements have been proposed based on operating experience and service conditions. For all RCPB environments, particular review emphasis is placed upon the corrosion resistance and stress corrosion cracking resistance properties of the proposed nickel-chromium-iron alloy(s).

6. 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 final safety analysis report (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 (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

7. For review of both DC and COL applications, 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 safety evaluation report. The reviewer also states the bases for those conclusions.

1. The staff concludes that the materials used for the reactor internals and core support structures are acceptable and meet the requirements of 10 CFR 50.55a and 10 CFR Part 50, Appendix A, General Design Criterion 1. This conclusion is based upon the following considerations:

   The applicant has selected, and identified by specification, materials for the reactor internals and core support structures that satisfy the requirements of Subsubarticle NG-2120 of Section III, Division 1 and Tables 2A, 2B and 4 of Section II of the ASME Code. For materials not in accordance with ASME Code provisions, the applicant has selected materials of construction that are approved for use by NRC-accepted ASME Code Cases, as identified in Regulatory Guide 1.84, or that have otherwise been demonstrated acceptable for the application. As proven by extensive tests and satisfactory performance, the specified materials are compatible with the expected environment and corrosion is expected to be negligible.

   The applicant has demonstrated that the design, fabrication, and testing of the materials used in the reactor internals and core support structures are of high quality standards and are adequate to assure structural integrity. The controls imposed upon austenitic stainless steel components satisfy the positions of Regulatory Guide 1.44, “Control of the Use of Sensitized Stainless Steel,” and the related criteria provided in SRP Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.”

   The controls imposed on the reactor coolant chemistry provide reasonable assurance that the reactor internals and core support structures will be adequately protected during operation from conditions that could lead to stress corrosion of the materials and loss of component structural integrity.

   The material selection, fabrication practices, examination and testing procedures, and control practices provide reasonable assurance that the materials used for the reactor internals and core support structures will be in a metallurgical condition that will preclude inservice deterioration.

   Conformance with relevant requirements of the ASME Code, or accepted Code Cases, and the recommendations of Regulatory Guides 1.31 and 1.44 and the related criteria in SRP Section 5.2.3, constitutes an acceptable basis for meeting the relevant requirements of 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1.

2. 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 SRP section.

3. 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 following is intended to provide guidance to applicants and licensees regarding the staff's plan for implementing this section of the Standard Review Plan.

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 submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision. Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides.

VI. REFERENCES

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

2. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Plants," General Design Criterion 1, "Quality Standards and Records."

3. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

4. Regulatory Guide 1.84, "Design, Fabrication, and Material Code Case Acceptability, ASME Section III."

5. ASME Boiler and Pressure Vessel Code, Section II, "Materials," Tables 2A, 2B and 4; Section III, "Rules for Construction of Nuclear Facility Components," Division 1; and Section IX, "Welding and Brazing Qualifications." American Society of Mechanical Engineers.

6. NUREG-1823, "U.S. Plant Experience with Alloy 600 Cracking and Boric Acid Corrosion of Light-Water Reactor Pressure Vessel Materials." U.S. Nuclear Regulatory Commission. Washington, DC. April 2005.

7. NRC Letter to All Licensees of Pressurized Water Reactors (PWRs), "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations" (Generic Letter 97-01). April 1, 1997.
